NRC FORM 658
(9-1999)

U.S. NUCLEAR REGULATORY COMMISSION

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DATE OF MEETING
4/7/00

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Docket Number(s)	50-247
Plant/Facility Name	Indian Point 2
TAC Number(s) (if available)	MA 8449
Reference Meeting Notice	March 24, 2000
Purpose of Meeting (copy from meeting notice)	To Provide an opportunity for the petitioners and the licensee to provide additional information and clarifications concerning a 10 CFR 2.206 petition submitted on March 14, 2000 requesting that the NRC order that Indian Point 2 remain shutdown until the steam generators are replaced and resolution of related issues.

NAME OF PERSON WHO ISSUED MEETING NOTICE
L. A. Wiens

TITLE
Petition Manager

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MEETING ATTENDEES

INDIAN POINT 2 - 2.206 PETITION

April 7,2000

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Statement of Congresswoman Nita M. Lowey At April 7, 2000 NRC Hearing

I share the concerns of many environmental groups and residents regarding the status of Indian Point 2 nuclear power facility. I believe the owners and regulators of the facility have a responsibility to prevent the reopening of Indian Point 2 until several conditions are met to protect the health and safety of the residents of the surrounding communities.

As you know, the following conditions have been raised by several scientific and community advocacy organizations including the Pace Law School Energy Project, the Public Citizen's Critical Mass Energy Project, and the Union of Concerned Scientists. I expressed these concerns in a letter to Chairman Richard A. Meserve on March 22, 2000. Again, I join these organizations in urging that the plant remain closed until:

- **all four steam generators are replaced;**
- **Dr. Joram Hopenfeld's on-going safety concerns regarding the NRC's policies and procedures to a.) permit plants to operate with damaged steam generator tubes; b.) use the Westinghouse methodology to test the tubes; and c.) appoint a panel of NRC experts, instead of an independent panel, to evaluate his objections, are fully addressed and resolved; and**
- **potassium iodide (KI) tablets are distributed to residents and businesses within the 10-mile emergency planning zone and are stockpiled in the vicinity of the Indian Point 2 facility.**

I believe that implementation of these recommendations <u>before</u> Indian Point 2 is reopened will help maintain public health and safety while reviewing the efficiency and effectiveness of the facility.

PRINTED ON RECYCLED PAPER

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

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.

Nuclear Information & Resource Service
Public Citizen's Critical Mass Energy Project
PACE Law School Energy Project
Union of Concerned Scientists

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).

[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

[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.
>
> 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.

<u>**Radiation exposures to workers at IP2 will be reduced.**</u>

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 <u>actual</u> 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

[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 <u>**NOT**</u> 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.

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

Public Citizen

Buyers Up • Congress Watch • Critical Mass • Global Trade Watch • Health Research Group • Litigation Group
Joan Claybrook, President

Public Citizen's Comments
to
The U.S. Nuclear Regulatory Commission
on
The Petition Pursuant to 10 CFR 2.206
Concerning Indian Point Nuclear Generating Unit 2

April 7, 2000

Good afternoon, my name is James Riccio. I'm with Public Citizen's Critical Mass Energy Project. My task this afternoon is to brief you on the history of steam generator tube failures here in the U.S. and to demonstrate why we believe this history indicates that the NRC should not allow Indian Point 2 to restart unless and until it replaces the nuclear reactors steam generators.

While the economics of steam generator replacement are questionable and may place the future operation of Indian Point 2 in jeopardy, this should not be a concern of this panel or this agency. Steam Generator tube degradation has already contributed to the early retirement of several nuclear reactors including: Portland General Electric's Trojan reactor in Oregon, Maine Yankee Atomic Power Company's Maine Yankee reactor and Commonwealth Edison's Zion Units 1 & 2 in Illinois.

Although originally designed to last the life of the plant, steam generators have been replaced at nearly two dozen nuclear power plants since the 1980's. Steam generator tube degradation is not only a financial risk to the utility but more importantly a safety risk to the surrounding communities. When degraded steam generator tubes go undetected, they may break, initiating a potentially disastrous sequence of events. The rupture of as few as ten steam generator tubes could result in the meltdown of the reactor fuel rods, potentially releasing catastrophic amounts of radiation into the surrounding communities.

Unfortunately, the Nuclear Regulatory Commission (NRC) staff continues to find that cracks in steam generator tubes may go undetected 40 to 60% of the time. In a November 1992 memo, which had been withheld from public disclosure, the NRC's Director of Nuclear Reactor Regulation reported that **"steam generator tube rupture events appear to be unavoidable."** According to the NRC, spontaneous tube ruptures have occurred at a rate of approximately one every 2 years for the last 20 years, while tube failures that were incipient and self-identifying through excessive steam generator tube leakage just prior to rupture have occurred at a rate of approximately one per year.

Ralph Nader, Founder
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A review of the following history will show that NRC regulation has been unable to adequately address the issue of steam generator tube ruptures. The nuclear industry's efforts to detect potential tube ruptures have been ineffectual. The steam generator tube ruptures to date have shown that the myriad causes of steam generator tube degradation have gone unchecked, that inspection methods are insufficient to preclude further ruptures and that tube rupture is often accompanied with degradation to other tubes, raising the possibility of a multiple tube rupture.

A Brief History of Steam Generator Tube Problems at U.S. Nuclear Reactors

1975: The U.S. nuclear industry experienced its first steam generator tube rupture at Point Beach Unit 1 in Wisconsin. The steam generator tube rupture occurred in less than 5 years operation. Subsequent inspection revealed that 127 tubes had degraded wall thickness greater than 60%. The steam generators at Point beach were Westinghouse model 44's with mill-annealed Alloy 600 tubing and were replaced in 1984. These are the same steam generators that are still installed at Indian Point 2.

1976: Surry Unit 2 reactor located near Williamsburg, Virginia experiences the second steam generator tube rupture at a U.S. nuclear plant. The steam generator tube rupture occurred in less than 4 years operation. Subsequent inspection of nine U-bend sections of the steam generator tubes revealed a 4.5 inch crack, four of the other eight pulled tubes revealed cracking that was undetectable with inspection techniques available at the time.

1978: The NRC designates steam generator tube integrity as an Unresolved Safety Issue and plans were established to evaluate the safety significance of degradation in PWR steam generators.

1979: Prairie Island Unit 1 near Minneapolis, MN experienced a spontaneous tube rupture caused by a loose part in the steam generator. NRC issues Information Notice 79-27 Steam Generator Tube Ruptures at Two PWR Facilities, documenting the accident at Prairie Island as well as a similar accident at the Doel 2 nuclear reactor in Belgium. Both reactors were Westinghouse 2-loop plants. The Prairie Island accident released approximately 30 curies of radiation into the environment.

1982: The Ginna reactor located near Rochester, NY experienced a spontaneous tube rupture caused again caused by a loose part in the steam generator. The steam generator tube rupture occurred in less than 5 years of operation. Inspections in April of 1981 revealed eddy-current indications that were not interpreted as needing plugging. The Ginna accident released 90 curies of radiation into the environment. Had there been any damage to the core of the reactor, the bypass of the containment would have provided highly radioactive fission materials a direct pathway into the environment.

1982: Consolidated Edison sues Westinghouse over the Indian point 2 steam generators.

Every utility, with the exception of the Tennessee Valley Authority, that has purchased a Westinghouse reactor has subsequently sued the corporation over problems with their steam generators.

1984: Fort Calhoun, near Omaha, NE, experienced a spontaneous tube rupture. The ruptured tube had been included in the last steam generator tube inspection and occurred after less than 10 years of operation. Re-evaluation of the data from that inspection revealed a 99% through wall defect where the tube eventually ruptured.

1987: North Anna Unit 1 near Fredricksburg, VA experienced a spontaneous tube rupture. The rupture was caused by a 360 degree through wall crack and occurred after less than 10 years of operation. The plants technical specifications did not require much inspection of the area that eventually ruptured because it was on the cold-leg side of the steam generator. Thus in the previous inspection only 13% of the tubes in this area were inspected. The tube that eventually ruptured was not among the 13%.

1988: NRC Commissioner Kenneth Rogers acknowledges that multiple tube ruptures can lead to a meltdown of the nuclear reactor:

> The concern is with sudden multiple tube failures- common mode failures. For example, such failures could come about by having essentially uniform degradation of the tubes. Degradation would decrease the safety margins so that, in essence, we have a 'loaded gun,' an accident waiting to happen. Under those conditions, a pressure transient or a seismic event could rupture many tubes simultaneously. That could allow primary coolant to enter the secondary system and the resulting high pressure to lift the relief valves that are outside containment on the steam line, thus permitting primary water to by-pass containment and communicate with atmosphere directly, resulting in a LOCA (loss of coolant accident)

1988: NRC issues Information Notice 88-31: Steam Generator Tube Rupture Analysis Deficiency acknowledging that if the break location becomes uncovered, a direct path might exist for fission products contained in the primary coolant to be released to the atmosphere. The licensee further concluded that the offsite dose consequences exceeded those calculated in the Updated Final Safety Analysis Report (UFSAR) because tube uncovery could produce a direct path for fission product release.

1988: Indian Point 3, located 35 miles for New York City, experiences an incipient tube rupture after 13 years of operation. A 120 gallon/hour leak developed over a two and a half hour period. This amount of leakage was 7 times the technical specification limit. Subsequent inspection revealed a 250 degree circumferential crack.

1989: McGuire Unit 1, located near Charlotte, NC experienced a spontaneous steam generator tube rupture after less than 8 years of operation. The rupture was caused by stress corrosion cracking involving multiple sites along the tube. Prior to the rupture, primary to secondary leak rate had been low. The rupture released approximately 30 curies of radiation in to the environment.

1989: Beaver Valley Unit 2, located near Pittsburg, PA, experiences an incipient tube rupture due to wear caused by loose parts. The subsequent inspection revealed that the loose part had removed 97% of the tube wall. Three adjacent tubes were also damaged with wear 62 to 97% through the tube wall.

1990: Duke Power Company sues Westinghouse over the steam generators in the four reactors at Oconee and Catawba stations. Duke alleged that Westinghouse had hidden problems with Inconel or Alloy 600 since 1964. Internal Westinghouse memoranda were cited by Duke in support of their allegations:

> An August 17, 1964 Westinghouse memo stated, "Mr. Simpson was informed that he was not to inform anyone with the exception of his boss of the inconel corrosion problem, to prevent a hold on steam generator production."
>
> A June 11, 1968 Westinghouse memo contained the following hand written note: "What do we tell them at this stage? That the alloy is crumbling before our eyes or that service experience is so far good?

1990: Maine Yankee, located near Bath, ME experiences an incipient tube rupture. The licensees staff re-analyzed their steam generator data from 1988 and found the indication that may have been the precursor to the accident.

1991: An ACRS letter to NRC Chairman Selin states "(t)he sudden rupture of steam generator tubes due to a transient such as a steam line break or seismic event needs to be precluded."

1992: McGuire Unit 1 and Arkansas Nuclear One, Unit 2 both experience incipient tube ruptures. In both instances, inspections in the previous year missed indications of tube wear that exceed the 40% through wall threshold.

1992: Precipitating Portland General Electric's (PGE) decision to close the Trojan reactor in Oregon, Mr. Hopenfeld filed a differing professional opinion (DPO) regarding an NRC decision to allow the nuclear reactor to operate with seriously degraded steam generator tubes. The issue was that "a main steam line break (MSLB) outside containment could trigger a multiple steam generator tube failure which could then result in a core melt because of depletion of coolant inventory." NRC documents leaked to the Union of Concerned Scientists revealed that the risk of a meltdown at the Trojan reactor was 300 times greater than the NRC's Safety Goal standard. Trojan was eventually shutdown and PGE sued Westinghouse rather than replace the steam generators.

1992: In a memo, which had been withheld from public disclosure, the NRC's Director of Nuclear Reactor Regulation reported that "steam generator tube rupture (SGTR) events appear to be unavoidable." The memo also points out that NRC regulation is less stringent than other countries. "Regarding steam generator tube inspection programs, it is clear that the U.S. lags behind the major European countries in terms of scope of inspection.... Further, the leak rates allowed were reported to be consistently much lower than that allowed by U.S. Technical Specifications."

1993: Palo Verde Unit 2 located near Phoenix, AZ experienced a spontaneous tube rupture after less than seven years of operation. A month prior to the tube rupture the licensee had observed an increasing trend in radiation monitoring activity. NRC's Augmented Inspection Team later determined that the licensees monitoring method had been inaccurate and had caused the leak rate to be underestimated by a factor of ten.

1994: An ACRS letter to NRC's Executive director for Operations, James Taylor notes that Mr. Hopenfeld's Differing Professional Opinion "appears to warrant further consideration. This issue has not yet been resolved..."

1994: Maine Yankee was shut down in July due to steam generator tube cracks that had been present since 1990 but had gone undetected. The Maine Yankee Atomic Power Company claimed that, even with the circumferential cracks, the steam generator tubes could have withstood a worst-case-accident. Whether Maine Yankee's assertions were true or in fact it violated the NRC's requirements for steam generator tube integrity has never been determined. After attempting unsuccessfully to find a buyer for Maine Yankee, the utility retired the reactor.

1995: NRC issues Generic Letter 95-03: Circumferential Cracking Of Steam Generator Tubes to notify addressees about the recent steam generator tube inspection findings at Maine Yankee Atomic Power Station and their safety significance. Later that year the NRC issued Generic Letter 95-05, Voltage-Based Repair Criteria for Westinghouse Steam Generator Tubes Affected by Outside Diameter Stress Corrosion Cracking. The alternate repair criteria allows a greater number of tubes with crack indications to remain in service.

1995: Ms. Connie Hogarth filed a 2.206 petition with the U.S. Nuclear Regulatory Commission requesting that the operating licenses for Indian Point Nuclear Generating Units 2 and 3 be suspended until the licensees have completed the actions requested by Generic Letter 95-03.

1996: NRC denies Ms. Hogarth's 2.206 petition stating that due 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 condition before a tube ruptures."

The NRC acknowledged that stress corrosion cracking of the Indian Point Unit 2 steam generator tubes was first detected during the 1993 refueling outage. However, Unit 2 steam generator tubes that showed signs of circumferential cracking have been removed from service.

1996 Commonwealth Edison retires the Zion Unit 2 reactor rather than replace the steam generators. Zion Unit 1 is shut down five months later. The reactors operated for less than 24 years.

1997 NRC issues Information Notice 97-79: Potential Inconsistency In The Assessment Of The Radiological Consequences Of A Main Steam Line Break Associated With The Implementation Of Steam Generator Tube Voltage-Based Repair Criteria. The notice states that Commonwealth Edison's Braidwood reactor miscalculated the accident consequences of a main steam line break when it applied for license amendments to implement the new repair criteria. The notice goes on to acknowledge that other licensee had made the same mistake in their license amendment requests.

1998 NRC kills plans for a steam generator rulemaking and a proposed generic letter, instead deferring to the Nuclear Energy Institutes 97-06. The staff was leaning toward rulemaking and the generic letter because reactor technical specifications were not adequate to ensure safety from new, more severe from of steam generator tube degradation.

1999 NRC staff grants Indian Point 2 a license amendment that allows Consolidated Edison to forego steam generator tube inspections required by their technical specifications. This was supposed to allow a one time exemption from the 24 month inspection interval and removed the requirement that the NRC approve the Indian Point 2 steam generator inspection program.

2000 Indian Point Unit 2 is forced to shut down on February 15th due to a steam generator tube failure. NRC later acknowledged that both Con-Ed and the NRC staff had mishandled the 1997 steam generator tube inspection and that the 1999 license amendment was based upon faulty analysis.

How this agency can allow aged nuclear plants to forgo steam generator tube inspection is beyond me. I know that Indian Point 2 is not alone in this regard, the NRC has allowed many other reactors to skip inspections through several regulatory loop-holes. Does the NRC really want gamble with the prospect that the first multiple tube rupture at a U.S. nuclear reactor will occur 35 miles from New York City?

Considering the steep cost of steam generator replacement and the uncertainty of recouping the investment in a competitive market, Consolidated Edison may decide not to replace their steam generators and either attempt to sell or retire the reactor. However, the prospect of nuclear reactors limping along with seriously degraded steam generators is neither in the interest of the nuclear utilities nor in the interest of public health and safety. That is why I'm asking that the NRC not allow Consolidated Edison or any other owner to restart Indian Point 2 unless the steam generators have been replaced.

I thank you for your time and consideration of this most important issue.

TESTIMONY OF EDWARD A. SMELOFF
EXECUTIVE DIRECTOR, PACE LAW SCHOOL ENERGY PROJECT
APRIL 7, 2000 MEETING ON INDIAN POINT 2

I am Ed Smeloff, Executive Director, of the Pace University Law School Energy Project, based in White Plains, New York. The Energy Project is a non-profit research and advocacy organization supporting policies for sustainable energy solutions. Before joining the Energy Project I served for 11 years on the Board of Directors of the Sacramento Municipal Utility District which owns a nuclear power plant. In that capacity I was invited twice to testify before the Nuclear Regulatory Commission.

I have recently attended two public meeting organized by NRC staff in Westchester County. At those meetings NRC staff have repeatedly stated that a decision to replace the steam generators at Indian Point 2 is a business decision and that it up to the licensee, Consolidated Edison, to make that decision based on its own criteria. I will demonstrate to you in my following testimony that the economic regulatory policies in New York are perverse and create incentives that encourage risk taking in the operation of the nuclear facility. The structural perversity of economic regulatory policy in New York adds to the NRC's burden to assure that the licensee maintains the plant's structural integrity and operates the facility safely. Given the recent incident at the plant and clear indications of the initiation of stress corrosion cracking in steam generator tubes at Indian Point 2 the NRC must not permit Consolidated Edison to restart the plant unless the steam generators are replaced.

In New York, Con Ed's electric rates are governed by a restructuring settlement agreement entered into by the utility, the state Public Service Commission and several other parties. That agreement freezes the portion of electric rates associated with the recovery of Con Ed's investments in Indian Point 2 and other facilities. However, it does permit adjustments to Con Ed's rates based on the amounts and costs of power it purchases in the wholesale power market. This so-called fuel adjustment clause permits Con-Ed to pass through to ratepayers almost all of the costs of power purchased during forced outages at Indian Point 2. This set of incentives encourages Con Ed to defer capital investments and places the risk of forced outages at the plant on Con Ed's ratepayers. This is what I call heads-I-win, tales-you-lose regulation.

It could be argued that Con Ed is still at risk if it can be proved that they acted imprudently in deferring the replacement of the steam generators and that this decision resulted in the forced outage that caused the need to purchase large quantities of replacement power. However, Con Ed can argue that its actions are prudent because it has always acted with NRC oversight and that the NRC approved the decision not to replace the steam generators in 1997 and to waive an inspection of them in 1999.

Now let me show you specifically and concretely why the decision to defer replacing the steam generators in 1997 was in Con Ed's economic interests but not in the public's interest from an economic point of view. Con Ed has stated in its 1998 annual report that the cost of replacing Indian Point's steam generators with ones that have been in storage at the plant since 1987 is $100 million. For the purpose of this analysis I have assumed that as the capital cost that was deferred in 1997 when Con Ed decided not to replace the steam generators. To calculate the value of the deferral to Con Ed I have used an inflation rate of 2.75 percent and a discount rate of 10 percent. Using those assumptions the present value of a three-year deferral is $21.5 million. The present value of a seven-year deferral (until 2004) is $38 million. Please note that all the benefits of deferring this investment accrue to Con Ed shareholders.

The economic risk of the decision to defer the replacement of the steam generators in 1997 was that a steam generator tube would break and that Con Ed would have to buy replacement power to meet its customers' demand for electricity. As we know now that is, indeed, what happened. Con Ed has stated that the cost of replacement power is $600,000 per day. For a 90-day outage (that is, one lasting until May 15) the cost will be $54 million. Discounting that back to 1997 gives a present value of $43.3 million. That is the amount of money that would have had to be invested in 1997 at a 10 percent interest rate to be able to purchase 90 days worth of electricity at the wholesale rate in 2000.

This analysis shows that from a societal perspective (combining Con Ed's shareholder and ratepayer interests) a loss of $21.8 million, assuming a three year deferral in steam generator replacement and a $5.3 million loss from a seven-year deferral. What is perverse about this situation is that the benefits and losses associated with Con Ed's gamble are disproportionate. The shareholders reap the benefits of the deferral while the ratepayers are burdened with the loss associated with the cost of replacement power. Another way of looking at this is that in 1997 Con Ed flipped a coin knowing that if it came up heads they won and if came up tails the ratepayers lost. With the game rigged in this way Con Ed has a very strong incentive to defer replacing the steam generators for as long as it can get away with it.

This perverse set of economic incentives places an enormous burden and duty on the Nuclear Regulatory Commission. Because what we are talking about here is not just an economic game where tens of million of dollars are at stake. It is not just a business decision whether the steam generators at Indian Point 2 are replaced. What is at stake here is the public health of safety of the citizens of Westchester, Putnam, Rockland and Orange counties in New York.

I will end my portion of the presentation to you by pointing out that the elected officials of these counties understand the importance of replacing the steam generators. I would like to enter into the record the names and positions of 30 elected officials, republicans, democrats and independents who have called upon the NRC to assure that Indian Point 2 does not re-start until the steam generators are replaced.

Thank you.

Elected Officials Calling for Replacing Steam Generators at Con Edison's Indian Point 2 Nuclear Plant

Sandra Galef, Assemblywoman, 90^{th} AD
Vincent Leibell, NYS Senator, 37^{th} SD
Suzi Oppenheimer, NYS Senator, 36^{th} SD
Alex Gromack, NYS Assemblyman, 92 AD
Willis H. Stephens, Jr., NYS Assemblyman, 91^{st} AD
Michael Kaplowitz, Westchester County Legislator
Richard Wishnie, Westchester County Legislator
Bruce A. Burns, Village of Briarcliff Manor Trustee
Gary Bell, Village of Buchanan Trustee
Deborah Fay, Village of Buchanan Trustee
Jane Hitney, Village of Buchanan Trustee
Linda D. Puglisi, Town of Cortlandt Supervisor
Ann Lindau, Town of Cortlandt Councilwoman
Francis X. Farrell, Town of Cortlandt Councilman
John Sloan, Town of Cortlandt Councilman
Robert Elliott, Village of Croton-on-Hudson Mayor
Georgianna K. Grant, Village of Croton-on-Hudson Trustee
Deborah Yurchak McCarthy, Village of Croton-on-Hudson Trustee
Marion S. Sinek, Town of New Castle Supervisor
Richard Laster, Town of New Castle Deputy Supervisor
John Chervokas, Town of Ossining Supervisor
Francesca E. Connolly, Town of Ossining Councilwoman
Kathryn I. Penn, Town of Ossining Councilwoman
Thomas Cambariere, Village of Ossining Mayor
Miguel Hernandez, Village of Ossining Trustee
Melvin Bolden, City of Peekskill Councilman
Milagros Martinez, City of Peekskill Councilwoman
Al Undly, City of Peekskill Councilman
Mary Beth Murphy, Town of Somers Supervisor
Linda G. Cooper, Town of Yorktown Supervisor
Nicholas Bianco, Town of Yorktown Councilman

Economic Analysis of the Decision to Defer Replacing the Steam Generators at Indian Point 2

✓ Base Plan Is to Defer A $100 Million Investment In 1997 For Three Years (to Year 2000)

- 2.75% Inflation Rate, 10% cost of debt
- Present Value (base plan) = $100 Milllion * (1.0275 / 1.10) ^3
  = $100 Million * .815
  = $81.5 Million in Present Value (1997) dollars

✓ Savings From Postponement Is $100 Million (1997 Investment) MINUS $81.5 Million (Present Value Of Investment Made In 2000) EQUALS $21.5 Million SAVING FROM Postponement

✓ LOSS From the plant outage in the 1st Quarter of 2000

- 90 day outage
- $600,000 Per day
- $54 Million (nominal cost)
- $43.3 Million (PV)
- calculated as $54 Million * (1.0275 / 1.10) ^3.25
  = $54 Million * .801
  = $43.3 Mil in Present Value dollars

✓ TOTAL COST OF POSTPONING THE INVESTMENT IS

($43.3 Million)
+$21.5 Million

=($21.8 Million) IN ADDITIONAL COSTS

STATEMENT OF PETER CRANE
Commission Meeting on Indian Point 2.206 Petition
April 7, 2000

Good afternoon. My name is Peter Crane. I used to work here. I started at NRC 25 years ago this week, as legal assistant to Commissioner Marcus A. Rowden. From 1977 to 1999, except for a year spent as an administrative judge in the Central Pacific, I was a lawyer in OGC. I retired last year, and these days my profession is history.

I'll be brief. First, I'm not affiliated with any of the petitioner groups here today. I don't represent them nor they me. They have asked me to speak to the generic issue of potassium iodide for thyroid protection and about the DPO process at NRC. It's up to them to draw whatever links they see to their petition.

My views on KI have been reasonably consistent since I filed my DPO in June, 1989. Rather than reiterate them here, I've attached a copy of the statement I prepared for the Commission meeting of November 1997. The best part of that statement is the part I didn't write. It's from a letter to FEMA by Dr. Jacob Robbins, Scientist Emeritus at NIH and a world-renowned expert on thyroid cancer. He wrote:

> 1. The Chernobyl experience has shown us that thyroid cancer is indeed a major result of a large reactor accident, even when evacuation is carried out;
> 2. The Polish experience has shown us that large scale deployment of KI is safe;
> 3. The Three Mile Island experience has shown us that it is not easy to obtain a good supply of KI in an emergency;
> 4. The shelf life of properly packaged KI is extremely long;
> 5. The advantage of having a supply on hand for immediate use far outweighs its moderate cost;
> 6. The problems attendant on predistribution are immaterial for the matter of creating a stockpile;
> 7. No one questions the ability of KI to protect the thyroid from radio iodine;
> 8. Even though KI administration before any exposure is ideal, the Chernobyl experience also has shown us that the exposure can continue for days; institution of KI blockade at any time in this period is beneficial.

In the 21 years since Three Mile Island, the NRC staff and the Commission have repeatedly arrived at sound, well-reasoned positions on KI -- only to abandon them. Four come to mind:

(1) the Commission's announcement in 1979 that stockpiling of KI would be part of every emergency plan;

(2) the staff's proposal in 1982 that the Commission approve a federal policy statement strongly backing KI;

(3) the 1994 staff paper that said that "it appears prudent to stockpile KI for limited populations located close to the operating nuclear power plants," and proposed that the NRC recommend to states that they stockpile the drug;

(4) the Commission's 1997 decision to back the recommendation of a 17-agency committee that the federal government buy KI for any state that wanted it. In announcing the decision, the Commission said explicitly, "The NRC will provide the funding."[1]

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[1] See the NRC's website for this press release, dated July 1, 1997.

So no one should accuse me of wanton Commission-bashing or staff-bashing. I approve of each of those positions. It is the capricious discarding of those positions that I can't agree with.

The Commission's flip-flop in April 1999 bears special examination. It was not grounded on the supposed dangers of KI, or any of the similar twaddle that the staff put forward in NUREG-1633, the 40-page technical assessment that managed to leave out the fact that KI was a medicine found "safe and effective" by the FDA. Instead, the April 1999 reversal was based on the argument that while KI could be useful, NRC didn't have the money to divert to new initiatives.

That's a dangerous line of argument to pursue, for it opens a Pandora's box of comparisons. This has begun already. For example, Susan Hiatt objected to a $60,000 fee waiver to the owner of the Perry plant, saying that the same money would have paid for all the KI that Ohio had counted on getting from the NRC. The agency replied that the two were unconnected.

Perhaps so. But will that argument hold up indefinitely? When it becomes a matter of justifying this or that foreign trip, costly perk, or new set of plantings for NRC headquarters, are Commissioners prepared to argue to American parents, "It's not that we think it's *un*important to protect your children against thyroid cancer, we just think it's *more* important to use the money from your electric bills to pay for foreign travel, fancy mobile phones, and new shrubbery."? Like it or not, that's what the Commission's April 1999 position boils down to.

On the merits, all the most recent expert medical information tells us that the risks to the infant thyroid are even greater than previously thought. Look at the March 15 issue of "Cancer," published by the American Cancer Society, which talks about the extreme risks to children under two. Look at the revised guidance from WHO. It's no secret that FDA is revising its guidance on KI in the direction of more aggressive intervention with KI, aimed at protecting the youngest children.

The staff has done a brilliant job of putting off the day of decision, like Penelope with her weaving in the *Odyssey*. But after more than 20 years, enough is enough. In these two decades, the rest of the developed world has passed us by, leaving our children underprotected and the NRC's scientific reputation tarnished. I suggest to the NRC's new Chairman that the next time there is an international conference on radiation and thyroid cancer, like the one at Cambridge University in 1998, he should go there, and ask the public health experts and nuclear regulators of other countries what they think of the NRC's handling of the KI issue. If you can't go yourself, Mr. Chairman, send an assistant you trust.

Let me turn now to the DPO process. It was studied twice: in 1990, under the direction of Paul Bird, and in 1994, by a group headed by Guy Arlotto. Their reports are well worth reading, as are the statements that were submitted to them. Perhaps because it's more recent, I remember Guy's better. It was good. It found problems and made recommendations, if I recall correctly, but nothing changed.

The first lawsuit I ever handled in OGC involved the Bailly plant in Indiana, and among other things, its Mark II containment.[2] I mention the case because the DPO process reminds me a lot of the design of that containment. The Mark II operated on the principle that if something went wrong, and a pipe break released steam, the increased pressure would force the steam into large pipes leading down

[2] A wonderful old man named Charlie Horsky represented the utility. He did us the generous compliment of not filing a brief in the case. Instead, he wrote to the court that the NRC brief had said everything that needed to be said.

through the floor into a pool of cold water. There the steam would condense, reducing the pressure and protecting against any release. In much the same way, the DPO process contrives to pour cold water on the DPO and keep the problem safely inside the walls. The employee dutifully "works within the system," thinking his or her concerns are getting a fair shake. Perhaps sometimes they are, but you couldn't prove it by my experience.

The DPO process is supposed to provide a quick resolution of employees' safety concerns. My DPO on KI took five years and was never fully addressed. Mr. Hopenfeld, if *Inside NRC* is correct, went 11 years without even getting a DPO panel named. I had never heard his name, and for all I know, he had never heard mine. I wonder how many of us there were, each with a DPO on its own slow boat to China.

In what way was my DPO unaddressed? I had said that existing KI policy was based in part on inaccurate and incomplete information provided to the Commission and the public at a November 1983 briefing. It's an easy matter to establish whether that assertion was valid. Anyone who spends an hour reading the transcript will know the answer.

But the DPO panel would not answer the question, and so the Director of the Office of Research rejected the panel's report. Did the panel go back and complete the job? No.

Commissioner McGaffigan tried in a November 1997 Commission meeting to get an answer to the same question; he had no luck then, and I doubt he'd have any greater success today.

The staff is supposed to work for the Commission. If a Commissioner can't get a straight answer to a simple question, what hope is there that some troublemaking DPO filer is going to get the staff to concede it made a mistake?

The NRC staff's managerial class is about as enthusiastic to find error on the part of its members -- let alone deliberate error -- as the police force of Prince George's County, Maryland. On paper, the DPO process sounds like a civilian review board. The reality is quite different.

So how is the Commission to know when the staff has dropped, or hidden, the ball?

When people like Mark Rowden and Joe Hendrie ran the NRC in the 1970's, there were some checks and balances that don't now exist.

First, the Commissioners had the Office of Policy Evaluation to give them independent technical and policy advice. OPE wasn't infallible, but it often had valuable insights to contribute, and the very fact of its existence helped keep up the quality of the staff's work. As I recall, senior NRC staff management generally viewed OPE as a pain in the neck, with a habit of making comments and asking questions that the staff sometimes didn't want to hear. (You can see that, incidentally, on the transcript of the November 1983 KI meeting I referred to earlier. OPE Director Jack Zerbe tried to get the staff to explain why it had made a 180-degree turn on KI in the space of just three weeks, but he got no answer.)

OPE was abolished in the mid-80's -- to my mind, a grave mistake. Its last director was excellent. He is still around the agency, and could tell you more about the decision to eliminate the office.

Second, denials of 2.206 petitions were reviewable in court. That kept the staff on its toes, and it also meant that OGC was involved. The lawyer who would have to defend the case would insist that issues be addressed and questions answered. Since that time, however, the law has changed: as a general matter, 2.206 denials now are not reviewable, which for NRC has been a mixed blessing.

Third, the introduction of the SES bonus system has played a part, I believe, in making the NRC managerial class more loyal to one another and less answerable to the Commission. For Commissioners come and go, but the *nomenklatura* goes on forever. If you want the other members of the team to recommend you for bonuses, you'd better be viewed as a team player.

So I would respectfully suggest to the Commission, if and when it looks at what went wrong with Mr. Hopenfeld's DPO, or with mine, to take a broad view, and to consider systemic problems, not just their occasional symptoms. If I were in the Commission's shoes, I would not turn to the staff for a self-evaluation -- few of us are good judges of our own performance -- nor to the Inspector General. I would look instead to people like Guy Arlotto and Joe Scinto -- recent retirees from a lifetime of service with the NRC staff, people who understand the staff from the inside, know its strengths and its weaknesses, and have an impeccable reputation for integrity and objectivity.

I've sometimes been reproached for saying and writing critical things about the staff and the Commission. The charge, explicit or implicit, is that this shows a lack of loyalty on my part toward colleagues or the organization.

I would answer that two ways. The first is that in four years as a cancer patient at NIH, I spent a lot of time in waiting rooms with sick children and their parents. This may sound like bringing in the violins, but it's the simple truth: you don't go through some kinds of experiences unchanged -- for example, seeing the numb expression on the faces of parents checking a young child back into the hospital a couple of days *before* Christmas, when every child well enough to go home for a few days was being checked out. Likewise, I'll never forget having my blood drawn one morning and hearing a voice from the next booth: "Oh, take *that* vein! That's a *good* vein!" I turned around and saw that the speaker was a boy no older than five. He was already an old-timer.

You often hear the argument that thyroid cancer is usually curable. I don't want kids to *need* to be cured. I don't want them sick in the first place, not when prevention is so easy and cheap. I don't want them spending their childhoods going back and forth to hospitals, becoming veterans of phlebotomy and nuclear medicine departments. I don't want them worrying, every time some passing infection gives them swollen glands, whether this is a return of their cancer. I don't want the lives of their parents and siblings also blighted by fear and suffering. Finally, "usually" curable isn't the same as *always* curable.

There is enough childhood disease out there that we *can't* prevent that we shouldn't pinch pennies about a type of childhood cancer that we *can* prevent.

So that's where my first loyalties are: with the kids and the parents. Does that imply disloyalty to the NRC? Not at all. What do you think would happen to this agency if there were a major nuclear accident or act of terrorism in this country, and children's thyroids were harmed because the NRC had never implemented the recommendation of the Kemeny Commission from 1979? What would Congress and the public say when they learned that the NRC had spent millions of dollars fighting KI instead of buying it? Would the NRC be given another chance? Would there even be a five-member body called the Nuclear Regulatory Commission a year or two later? What do you think would happen to the reputations and careers of present and former Commissioners and senior staffers when the hunt began for those responsible? I think these are all questions worth considering.

In sum, I recommend this medicine because if an accident or act of nuclear terrorism occurred, it could save children. It might also have the incidental effect of saving the NRC.

Thank you.

STATEMENT OF PETER CRANE
before the Nuclear Regulatory Commission
Open Meeting on Potassium Iodide (KI)
November 5, 1997

I appreciate the opportunity to address the Commission at this meeting on the radiation antidote potassium iodide (KI). This is the first meeting that the Commission has held on the subject in 14 years, and it is long overdue. I am here strictly in my private capacity, as an interested citizen as the petitioner in a rulemaking, not in my official capacity as Counsel for Special Projects in the NRC's Office of General Counsel. This statement was written at home, on my own time, and I am on annual leave as I speak to you this morning.

Potassium iodide is an effective, safe, and cheap medicine, with a long shelf life. It prevents thyroid cancer and other thyroid diseases by blocking the absorption of inhaled or ingested radioactive iodine. We have seen from Chernobyl what happens when you don't have KI to give out in an emergency: aggressive childhood thyroid cancers, in large numbers, appearing only a few years later.

On July 1, the NRC issued a press release saying that the Commissioners had decided to back a new policy that would make supplies of KI available to states that ask for it. The press release never used the word "cancer." That's like announcing the availability of Sabin vaccine without using the word "polio."

The draft Federal Register notice announcing this policy, sent to the Commission by the staff in June, and not yet issued, uses the word "cancer" just once -- buried at page 8 of a 13-page notice. Moreover, that notice never tells states that stockpiling is a reasonable and prudent measure. If it isn't reasonable and prudent, why is the NRC offering it? And if it <u>is</u> reasonable and prudent, why not say so loud and clear?

The states are beginning to catch on to what the Federal Government hasn't been telling them. Maine held a public meeting in December and its Radiation Advisory Commission voted unanimously for stockpiling that same day. Ohio had its meeting last week, and New York's meeting will be in Albany on November 21. I spoke at the first two of those and I hope to speak at the third. I certainly plan to be there.

At that meeting in Ohio, which was called by state and local health and emergency authorities, I talked about the cancers in Belarus, Russia, and Ukraine. I mentioned that the photographs of the young victims show incisions running from ear to ear, because in these children, the cancer tends to spread rapidly to the lymph nodes. The representative of the NRC countered that these scars were bigger than they had to be, because of the quality of medicine in the former Soviet Union.

Maybe that's true; I'm not a doctor. Maybe it's true that if, God forbid, there should ever be a major accident in which American children unnecessarily develop thyroid cancer, because KI was not there when it was needed, their surgical scars will be smaller than those on the necks of the children in Belarus. But I don't want American children to have <u>any</u> scars, big or little, when we can prevent the disease for pennies.

The person who could answer these and other questions about thyroid cancer is sitting in the audience. He is Dr. Jacob Robbins of the National Institutes of Health, as distinguished an expert on radiation-associated thyroid cancer as there is anywhere in the world, with decades of experience on several continents. He could be sitting at the table today. But he isn't, because the Commission rejected the request of the American Thyroid Association (representing the nation's thyroid specialists) that he be allowed to speak for 15 minutes. For a Commission that has so often stressed the value of listening to the interested public, this was, I think, a sad day.

So today, just as at the last such meeting, in November 1983, the Commission will have to depend on its technical staff for expert medical advice. Let's hope they do better today than they did then. I described in my petition how the staff misled the Commissioners and the public in 1983, because I was making the point that the existing federal policy, adopted in 1985, is grounded in misinformation, and thus was defective from the start. I included lengthy portions of the transcript of the meeting, where the staff tells the Commissioners that "it's a relatively minor operation," involving "a few days off," and never even talks about cancer. The staff was arguing -- successfully, in the end -- that instead of spending pennies on prevention, society should put its resources into curing the thyroid disease if and when it occurred, because this was more cost-effective. This was taking the old adage about an ounce of prevention being worth a pound of cure and turning it upside down.

I was looking forward to seeing how the staff dealt with the misinformation issue in their paper analyzing my rulemaking petition, because they have dodged the issue for many years. They don't even mention it. As in the past, it's as though I had never said it.

I'll say it again, then. My petition had two bases. The first is that the wealth of new data from Chernobyl -- both about the health consequences of a major accident and the safety and efficacy of KI -- call for having KI stockpiling as a backup safety measure for use in the unlikely event of a major accident. The second is that existing U.S. policy is grounded in misinformation for the Commissioners and the public. The evidence of that misinformation is the publicly available transcript of the November 22, 1983, Commission meeting. That evidence is there for all to see, even if the NRC staff puts its fingers in its ears and refuses to hear.

What's the result? That today, children in other countries, from Japan to Poland and from Canada to Switzerland, have a protection that American children don't have. In the United States, believe it or not, we have KI to protect the sharks at Sea World but not the children who come to see them.[1]

All over the world, countries know that if you are serious about being prepared to protect the public in nuclear emergencies, you should have three arrows in your quiver. Those are: (1) evacuation, which is the ideal solution -- when it is feasible; (2) sheltering, which means taking cover; and (3) potassium iodide. Having all three options gives you the flexibility to choose among them, or use them in combination, depending on the particular circumstances.

If you can evacuate the entire population before the radioactivity arrives, and don't need to use KI, so much the better. But in the real world, bad weather, congested roads, or changing winds can make a full evacuation impossible. In that case, it's better to be safe than sorry.

The French, Germans, Swedes, Slovaks, etc., all know this, and they stockpile KI. It's cheap enough, at about 10 cents per person protected, that the Poles keep 90 million doses on hand. In fact, three years ago the NRC's technical staff calculated that it would be cheaper to buy a national stockpile of KI -- for a total of a few hundred thousand dollars total, or $1100 for the average plant -- than to go on studying whether to do so.

Isn't that the definition of a "no-brainer"? Only in Washington would we spend more money studying whether a medicine to protect our children is worth buying than the medicine itself would cost.

Today, at international conferences on the health effects of Chernobyl, you can hear American doctors lamenting that though the value of KI is the number one health lesson learned from that accident, the U.S. continues to lag behind other countries in protecting its children.

It's not as though experts in the U.S. haven't known better for a long time. Almost 20 years ago, during the Three Mile Island accident, federal and state authorities went looking for KI and discovered there was none to be had. An official at the Food and Drug Administration had to get on the phone to a pharmaceutical company executive in the middle of the night and beg him to start up the production line and rush the drug to Pennsylvania.

Afterwards, the Presidential Commission that investigated the accident was scathing in criticizing the Government's failure to stockpile KI. Stockpiling, it said, was long overdue. In its response to the Presidential Commission's report, the NRC agreed wholeheartedly, and it promised to require KI stockpiling in the vicinity of nuclear power plants. Later it reneged.

Ten years after the accident, the FDA official, Jerome Halperin, wrote an article for a medical journal in which he lamented that KI preparedness was still in a pre-TMI state. He could write the same article today.

Wholly apart from the question of what the Federal Government did before Chernobyl, after Chernobyl, and especially after the reports of widespread thyroid cancer among children in the former Soviet Union were

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[1] The 8-year-old daughter of Charles Pond, the director of Tennessee's program, having somehow learned that sharks in captivity require KI for their health, persuaded her father that as the state's KI reaches the end of its shelf life (5 years), it should be donated to Sea World, where it is added to the sharks' water. See her father's statement at p. 57 of the transcript of the public meeting on KI held at FEMA on June 27, 1996. Young Ms. Pond's accomplishment was written up in the "Kids Did It!" section of a recent issue of the children's magazine, "National Geographic World".

confirmed, there was no excuse to leave the Government's anti-KI policy in place. But it was left in place, and the Federal Government did nothing to alert the states and the public to the new data. In fact, the NRC staff keeps repeating, like a mantra, "No new data, no new data." That is simply untrue. We know much more than we did, both about induction of childhood thyroid cancer and about the use of KI in a major accident. But the person who is truly learned about those subjects is Dr. Robbins, and the Commission doesn't want to hear him.

Only three states currently stockpile KI: Tennessee, Alabama, and Maine. One reason for the states' hesitation is that 15 years of inaccurate and incomplete information from the Federal Government have left some of them with little understanding of the stakes involved. Last year, for example, in a public meeting on KI at the Federal Emergency Management Agency, officials of two states justified their refusal to consider KI by declaring, in writing, "Loss of the thyroid is not life-threatening."

Try telling that to Senator Tom Harkin of Iowa, who lost a brother to thyroid cancer last year.

In fact, thyroid cancer is curable -- usually. About 16,000 Americans are diagnosed with the disease each year, and it kills only about 1200. Those 1200, however, are just as dead as the people who die of usually fatal cancers. And because the hormones produced by the thyroid gland affect the whole body, even a non-fatal case of thyroid cancer can have significant impacts on the quality of life. Just ask some patients. They will tell you about surgery and a lifetime on medication, at the very least; about radiation treatments that require hospitalization in radiological isolation, because the patients themselves are giving off radioactivity; about changes of medication in preparation for tests and treatments that leave patients exhausted and chilled to the bone; and about the anxiety that goes with any cancer. Having had thyroid cancer myself -- first when I was in my twenties, with a recurrence 15 years later that required five hospitalizations over three years -- I know that it is nothing I would wish on my children, nor would you wish it on yours.

Fortunately, we don't have to be completely without the benefit of Dr. Robbins's expertise today, because he put the arguments for KI stockpiling very succinctly in a letter to FEMA in July 1996. There is more analysis of the real issues in his brief letter than in the decision paper given to the Commission last week (SECY-97-245), and unlike the NRC staff, it didn't take him 26 months to prepare it. Here is what he had to say:

> "1. The Chernobyl experience has shown us that thyroid cancer is indeed a major result of a large reactor accident, even when evacuation is carried out;
> 2. The Polish experience has shown us that large scale deployment of KI is safe;
> 3. The Three Mile Island experience has shown us that it is not easy to obtain a good supply of KI in an emergency;
> 4. The shelf life of properly packaged KI is extremely long;
> 5. The advantage of having a supply on hand for immediate use far outweighs its moderate cost;
> 6. The problems attendant on predistribution are immaterial for the matter of creating a stockpile;
> 7. No one questions the ability of KI to protect the thyroid from radio iodine;
> 8. Even though KI administration before any exposure is ideal, the Chernobyl experience also has shown us that the exposure can continue for days; institution of KI blockade at any time in this period is beneficial."

I urge the Commissioners to make the NRC staff tell you what is wrong with that analysis.

The case for KI stockpiling was also made in a 1994 letter sent jointly by Senator Joseph Lieberman, a Connecticut Democrat, and Senator Alan Simpson, a Wyoming Republican. Again, there is more real grappling with the issues in its two pages than in all the recent NRC staff papers on KI put together.

Let me emphasize that I am not an alarmist about nuclear power, any more than Senators Lieberman and Simpson. I have been defending NRC nuclear safety decisions in court for 20 years. I think that a major release is unlikely, because, generally speaking, our plants are well built and well run. But we have emergency planning because we know that accidents can happen, and that their consequences can be serious. If we are going to have emergency planning at all, it might as well be done right. I have often compared KI to the lifejackets on a ferryboat. Ferryboat accidents are very rare, and if one does occur, it is better to be evacuated in a lifeboat than to jump into the sea in a lifejacket. But in the real world, the unexpected happens, so we have lifeboats and lifejackets. We don't do fancy cost-benefit analyses, we don't study the issue for 15 years, we just do it, because it would be reckless and irresponsible not to.

The opponents of stockpiling can be expected to make the argument that because the Federal Government has recently decided to stockpile KI in 27 cities for acts of nuclear terrorism, that states and localities with nuclear power plants can safely forget about the issue. That major shift in U.S. policy on KI is a good thing -- no question about it. But we are talking about a medicine whose value is entirely dependent on time. Before the exposure to radiation is better than after, one hour after is better than two hours after, and so so. Thus it makes sense to have the drug on hand locally as well -- in schools, firehouses, and hospitals, for example, as the World Health Organization recommends -- with plans in place for its use. If there is one thing we know about emergencies, it is that planning is always preferable to improvised, ad hoc responses.

The real significance of the Federal Government's decision to stockpile KI for acts of terrorism is the recognition that the drug is useful in radiological emergencies. If it is valuable for emergencies caused by acts of terrorism, then it is also valuable for emergencies caused by accidents.

Some states worry that they could be held liable for side effects caused by KI. That is not a realistic concern. First, we know from the Polish experience during Chernobyl that side effects of KI are minimal. They gave out 18 million doses and two people were hospitalized briefly, both of whom men who had known iodine allergies and took the medicine against doctors' advice. There is also a study in the U.S. that reported on KI consumption in cough and cold medications. It said, in 1995, "for the most current data involving 38 million equivalent doses of KI consumed, there were <u>no</u> reports of adverse reactions." [Emphasis in the original.]

Second, KI would be given to the public only after the Federal Government advised that the emergency called for it. (See the Federal Radiological Emergency Response Plan, issued by FEMA in 1996.) If states want to worry about liability, they should think about the legal consequences of <u>not</u> having stockpiled, given all the information available to them.

Last December, when the Maine Advisory Commission on Radiation voted unanimously to support stockpiling, one of its members explained his vote in these words: "Ten years from now, if we have a release, I would rather say that we erred on the side of conservatism, knowing what we know."

"Knowing what we know" -- that is the crux of the issue. As the word filters down to the states, at long last, individual states will be having to decide on KI. Because if ever there is an accident or act of terrorism at a nuclear plant in which Americans are harmed because KI was not available, those states must expect their citizens to ask: "When you knew that the Federal Government was offering free supplies of this medicine, and that the thyroid doctors unanimously said we should have it, and that other countries were protecting <u>their</u> children with it, how could you nevertheless have decided to leave <u>our</u> children unprotected?"

I don't have to spell out the questions that will be asked of the NRC if that ever comes to pass.

The pity of it is that all this present mess was completely avoidable. The NRC staff had it absolutely right in March 1994, when they advised the Commission:

> "[I]t appears prudent to stockpile KI for limited populations located close to the operating nuclear power plants. This option represents an interoffice consensus and is recommended by the [NRC] staff. ... While NRC encourages the stockpiling of KI, the decision to stockpile, distribute, and use KI would be the responsibility of the individual States...

But the then Commissioners, on a deadlocked 2-2 vote, did not accept that sound advice.

I'd like to close by quoting Leo Tolstoy, who in 1896 described his proposal for solving the problems of Government:

> "to be honest, not to lie, to act and speak so that your motives for action are understandable to your loving seven-year-old son; to act so that your son doesn't say: 'Papa, why did you say that then, but now say and do something quite different?'"

Thank you.

VTK

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FOR IMMEDIATE RELEASE

NUCLEAR POWER PLANT AGING:
THE U.S. REGULATORY PERSPECTIVE

COMMISSIONER KENNETH C. ROGERS
U.S. NUCLEAR REGULATORY COMMISSION
PRESENTED AT THE
INTERNATIONAL SYMPOSIUM ON NUCLEAR POWER PLANT AGING
BETHESDA, MARYLAND
AUGUST 30, 1988

Thank you, Mr. Aggarwal. Good morning, ladies and gentlemen.

On behalf of the U.S. Nuclear Regulatory Commission, I am delighted to add my welcome to you. I too am particularly pleased to note the fine international attendance at this Symposium on an important and universal problem. Either the topic is of keen interest or Bethesda has an irresistible appeal. Chairman Zech personally asked me to convey to you his regrets at not being able to be here.

Before I address the main theme of the Symposium, I would like to recognize and thank the co-sponsors of this Symposium. They are: the American Nuclear Society, the American Society of Civil Engineers, the American Society of Mechanical Engineers, and the Institute of Electrical and Electronics Engineers. Without their support, this Symposium would not have been possible.

I will speak to you briefly this morning of my perspectives on the aging problem in nuclear power plants. These will primarily be those of a Commissioner of a U.S. regulatory agency, but also of one who, as an experimental physicist, has extensive training and experience with the behavior of a wide variety of materials and systems, and who understands, I think, some of the generic issues associated with aging.

Importance of the Aging Problem

It has been my observation, since coming to the Nuclear Regulatory Commission, that the aging of nuclear power plants is one of the most important issues facing the nuclear industry worldwide and the governmental bodies which provide safety oversight and regulation. As evidence I need only cite an example such as the main feedwater pipe break at Surry Unit 2 in December of 1986, in which there were four fatalities. The post-accident investigation showed that the pipe failed because of a thinned

EXHIBIT A
Declaration of James P. Riccio
Westinghouse v. Carolina Power
Civil Action No. 89-0826

wall, which probably resulted from a corrosion/erosion mechanism - one form of aging. These conclusions were rapidly communicated to the entire world community.

While this incident is particularly significant because of the loss of life involved, nuclear power plants have, over the years, experienced a number of component degradations, and even failures, resulting from a variety of aging processes.

Aging encompasses all forms of degradation to nuclear power plant materials, components and systems which result from exposure to environmental conditions in the power plant or from operational characteristics or procedures. The agents of aging range from radiation, high temperatures and pressures, steam conditions, and corrosive chemicals in the power plant environment to thermal and pressure cycling, component vibration and mechanical wear of operation, to the degradation induced by inspection processes or other such procedures. Virtually every component in a reactor is subject to some form of aging, from the pressure vessel itself, to the fuel and cladding, to the pipes, the pumps and valves, and the electrical systems that make up the rest of the plant. Aside from the corrosion/erosion experienced by the Surry plant, aging phenomena can, depending on the component and the conditions, include embrittlement, fatigue, corrosion, cracking, mechanical wear, creep, and dimensional instability.

While failures of individual components constitute an operational concern, and can be a safety concern, the more significant safety concern results not so much at a single component level but at the higher level of components aggregation because our key safety systems have been designed to accommodate single failures.

We have been particularly concerned about common mode failures. For example, several forms of aging degradation of steam generator tubing have been identified. The concern is not a single tube leaking or even failing. The concern is with sudden multiple tube failures - common mode failures. For example, such failures could come about by having essentially uniform degradation of the tubes. Degradation would decrease the safety margins so that, in essence, we have a "loaded gun," an accident waiting to happen. Under those conditions, a pressure transient or a seismic event could rupture many tubes simultaneously. That could allow primary coolant to enter the secondary system and the resulting high pressure to lift the relief valves that are outside containment on the steam line, thus permitting primary water to bypass containment and communicate with the atmosphere directly, resulting in a LOCA.

Another concern is loss of defense in depth. A basic tenet of reactor safety design has been the incorporation of redundant and diverse systems to prevent and mitigate accidents. For example the reactor protection system, the emergency core cooling system, the containment spray system, and the onsite power system are all designed to assure multiple options for mitigation of a breach in the reactor pressure boundary. However, if this engineered redundancy and diversity is gradually and unknowingly reduced because of aging, then safety margins will eventually be seriously reduced.

Plant aging is particularly important in the United States, which has an older plant inventory, on the average, than does the rest of the world. Whereas nearly two thirds of the power plants in the rest of the world are under 10 years old, in the U.S., about two thirds are over 10 years old.

However, the United States is certainly not alone in facing aging problems. Great Britain, for example, still operates reactors dating from the mid to late nineteen fifties, and a number of countries have plants that are 20 or more years old.

In the United States, we can also anticipate that the problems of aging power plants will be exacerbated by owner desires to continue to operate existing reactors beyond the somewhat arbitrary 40-year period for which most of them are presently licensed. This problem is also not unique to the U.S., but is likely to be particularly important here because of the very small amount of new electrical power production capacity of any type that has been installed in recent years. Given the long lead times and high costs typically required to bring any new plant on line, the continued operation of existing plants may be very important to meeting growing electrical power demands.

Thus, we can anticipate that, while we first and foremost need to be concerned about the continued safe performance of aging nuclear power plants in the near term, we will also likely have to address the question of how much longer than 40 years they can operate safely, and what is required to assure the safety of such operation. Other countries have different reactor licensing requirements, and may or may not have to address the question of extending a license for a fixed period of time.

Managing the Aging Problem

Fortunately, there are a variety of measures that can be taken to "manage" aging. By managing aging, I mean predicting or detecting when a component or system has degraded to the point where it becomes a potential safety hazard, and taking appropriate corrective measures.

There are a variety of detection techniques and remedial actions that I am sure are familiar to all of you, so I'll merely mention them here. Detection can include continuous or periodic monitoring (for example, of acoustic emissions for valve leak detection, or of water chemistry to minimize the potential for corrosion, corrosion-erosion, and intergranular stress corrosion cracking) visual and non-visual inspections of components, either while in service or during outages (for example, eddy current testing of steam generator tubes or ultrasonic tests for crack detection in pipes, welds and flanges) and periodic functional testing of equipment (for example, cycling valves open and closed in standby safety systems). Remedial measures can include component replacements or such techniques as in-situ annealing of reactor pressure vessels to restore required mechanical properties.

For all such management measures, questions that must be addressed are how much to inspect or test, how often to repair or replace, and what constitute signals of degradation that should alert us. These are very important and basic questions, and are at the center of what you will be discussing this week.

I want to emphasize as strongly as I can that aging problems cannot be addressed simply by building in what has been called "conservatism." In my view, our conservatism hasn't always been conservative. When we stitch one margin of safety on top of others for no sound technical reason, but only for the hope of ensuring that we have covered any possible but unanalyzed error or unknown, we often inadvertently establish a situation which is not conservative at all, and we don't even realize it. Because some testing is good, more is not necessarily better. In fact, more is sometimes less safe, because inspection and testing are not necessarily inherently benign. Tests of equipment may themselves put strains on the equipment. We have discovered this in the case of diesel generators, where requirements to test the equipment by rapid starting from a cold condition have been found to put significant wear and tear on the generators. Thus, testing may in fact, reduce safety if it should put a generator in a condition where it will fail just when it is needed. Similarly, inspections may introduce wear on equipment or increase the probability for damage or errors in reassembling or reinstalling, if disassembly was required for the inspection.

Another aspect of the management of aging is the need for a knowledge base of plant history in establishing residual safety margins for components and systems. Knowledge of the numbers of cycles components and systems have experienced, the maximum temperatures and pressures they have experienced, the total numbers of hours of operation, and the cumulative exposure to corrosive chemical elements can all be related directly to the amount of aging that components and systems have experienced. As the inventory of nuclear power plants gets older, and certain systems and components approach regulatory limits for operation, the need for good records to establish the plant history will increase. I can see this becoming part of the integrated maintenance data bases many utilities are pursuing.

Finally, I note the importance of taking a systems approach to the challenge of managing aging. That is, aging cannot be addressed in isolation. Aging and maintenance share a symbiotic relationship. Maintenance requirements are, of course, significantly shaped by needs to counteract the effects of aging. Conversely, maintenance procedures can produce aging. I have touched on how accident problems can be exacerbated by, or even produced by, aging. Aging clearly has an impact on the qualification of electrical equipment. Certain human actions may also be important to aging; for example, human error may produce additional stresses on the system, and thereby accelerate aging. Thus, we cannot think that we are addressing the aging problem simply by looking at environmental interactions with materials and components, rather we must consider both the impacts of various activities on aging, and the impacts of aging on the entire system.

The Need for a Research Program

While we have, based on past experience, identified a number of aging problems and developed appropriate ways of managing them, I do not believe we know yet all we need to know about these problems. Operational experience continues to produce evidence of situations we did not anticipate, and research on known problems has demonstrated that some of our initial assumptions were incorrect or incomplete. Thus, I believe that a very important element of aging management is a strong research program.

A strong research program yields several important benefits:

Research can, of course, be used to examine clearly identified problems and to develop appropriate ways of managing them. Such measures can include the development of advanced monitoring and repair techniques and, where appropriate, the technical bases for the reduction of unnecessarily restrictive management measures that may diminish safety. For example, better understanding of the relationship between physical measures of degradation and resulting impacts on reliability and performance, and better ways of predicting rates of degradation over time can allow one to set more precise requirements on the scope and frequency of inspection and maintenance.

Research can also be used to reveal potential new sources of degradation, not yet observed in practice, before they become problems in existing plants. Accelerated aging experiments, controlled tests, and the development of computer models to explore material and component performance in domains beyond the design basis, can enhance the understanding of the behavior, during abnormal conditions, of components with various degrees of aging.

Research can help develop a better understanding of other important safety concerns. For example, our analyses of severe accidents have, to date, suffered in that the probabilistic risk assessments made have not in general been able to recognize the effects of aging on the probabilities of certain kinds of events. Incorporating aging effects into PRAs will give deeper insight into the understanding of severe accidents, and help us to address that problem.

And finally, research can provide additional capabilities to answer new questions rapidly and effectively as they arise. Several times in the past NRC has been able to resolve significant new safety issues rapidly because it had an accumulated body of high quality research data and available

experts to examine the problem. One such case was the Pressurized Thermal Shock (PTS) problem, involving the question of whether overcooling transients occurring in conjunction with high coolant pressure could result in stresses causing preexisting cracks to propagate through the pressure vessel wall. Because NRC had conducted research for 15 years on vessel aging, it had available data on reactor vessel steel aging, together with crack propagation analyses that already had been validated by the age-related data. These data showed that even the most susceptible vessels had a safety margin sufficient to permit generic resolution through rulemaking, without plant shutdowns.

I will leave it to Eric Beckjord and others speaking at this conference to describe NRC's Nuclear Plant Aging Research (NPAR) program in detail. I think our program is a strong one and is making significant contributions to the management of aging and to the improvement of plant safety.

The NPARs broad scope has allowed it to address numerous aging problems requiring different expertise and diverse research facilities. The long-term and continuing investment NRC has made in this program has given us the background and tools to answer questions that were not even anticipated when the research was initiated or performed.

This breadth and stability must be maintained in the future. We can do so, in part, by continuing our commitment to cooperative research programs with U.S. industry and with foreign countries, and by carefully selecting areas for research with a long-term perspective in mind.

Domestic and international cooperative research efforts have proven one of the most important cost effective and productive elements of our research program. They provide opportunities for the synergism that can come from a broad range of views and approaches, and they effectively extend the investments of each research partner through the sharing of expensive facilities, the division of labor in experimentation and analysis, the elimination of duplicating work already done and the sharing of information.

This symposium, of course, is an excellent manifestation of international collaboration. Other efforts I could cite include the recently completed 5-year cooperative program on Steam Generator Integrity with Japan, Italy, France and EPRI; a new 3-year cooperative program on Piping Integrity (the IPIRG program) which includes France, the UK, Canada, Sweden, Switzerland, Japan, Taiwan and EPRI; a cooperative program with Germany and the UK for testing and evaluation of pressure vessel steel from the German Gundremmingen reactor; participation with 14 countries in the International Cyclic Crack Growth Rate cooperative program to study the environmental effects of reactors on cracking in steels used in LWR's; and membership in the newly formed international cooperative program on Irradiation Assisted Stress Corrosion Cracking of LWR core internals. Every one of these cooperative programs falls under the "Aging" umbrella.

The appropriate selection of research areas is itself an important activity. First, none of us have sufficient resources, facilities and manpower to address in detail every possible combination of aging phenomena and susceptible components. Therefore, we must focus effort on those items which have the greatest potential to impact safety. We must, of course be assured that the selection of priority research areas covers a broad range of types of components and types of problems, so that, should a new problem be identified, we are likely to have done research in a related area.

Based on these considerations, categories of components that we believe have a potential aging-related impact on plant safety include: mechanical components such as motor-operated, check and solenoid-operated valves, power-operated relief valves, snubbers, compressors, heat exchangers, and pumps; electric components such as cables, circuit breakers, relays, chargers, inverters, batteries, motors, bistables, transformers, connectors, and electric penetrations; and systems such as the high and low pressure emergency core cooling systems, the residual heat removal systems, and the auxiliary feedwater systems.

A key requirement in recent years has been the need to take advantage of special time-critical opportunities to obtain actual reactor samples. For example, we have put considerable priority on obtaining exposed samples and components from facilities being shut down or decommissioned, such as the Shippingport Nuclear Power Plant in Pennsylvania. As such reactors are entombed, dismantled and components disposed of, a unique opportunity to study their aged components disappears.

I would like to leave you with some thoughts as to how I believe an aging program might evolve in the future. Perhaps you can consider these during the course of this symposium and in forward planning within your own institutions.

We need to take fuller advantage of relevant experience from other industries. While there are, of course, some unique aspects to nuclear power plants, they still share many components, materials, and environmental conditions with petrochemical plants, fossil fueled plants, aircraft, ships and other complex systems. We should make maximum use of the experience of these industries where it is applicable. The concept of using data from other industries is not new. We have done so before, for example in acquiring data for probabilistic risk assessments. I believe joint research and coordination with other industries in selected areas could help to assure a mutual exchange of relevant information.

We need to be more creative and far-sighted in developing advanced instrumentation and control systems to monitor and inspect nuclear plant components. I believe there is much more to be done in this area.

We need to understand better the interrelationships between aging and other concerns to be able to implement the systems approach I spoke of earlier. I have already mentioned the mutual interfaces between aging and: severe accidents, maintenance, human factors, and equipment qualification. I also see a need to apply the experience we have gained on the aging of present reactor components to the design of the next generation of reactors.

Conclusions

If one thing impresses me more than anything else about the aging problem, it is the numerous linkages it has: to other industries, to other areas the NRC is addressing, to other domestic organizations and to other countries. I have tried to point these out in my comments. These linkages make study of the aging problem more difficult in some respects, but they also provide great opportunity. I think these linkages will shape our efforts to manage aging better and better in the years ahead.

Aging is clearly already a very important area of concern, and it will only grow more important as reactors get older, and as we face the question of operating reactors beyond the 40 years presently authorized. Not only can we expect to continue to experience those aging problems we have already seen and studied, but we will also, most assuredly, discover new aging effects at levels of exposure and conditions beyond those we have yet encountered. To assure continued safety under these conditions, we will need information from all relevant sources, continued strong research efforts, and as much international commitment and cooperation as possible.

Thus, I commend you all and I hope the next few days are productive and beneficial for the advancement of world-wide nuclear power plant safety.

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Westinghouse

Leaks and Lawsuits

by Jim Riccio

Public Citizen's Critical Mass Energy Project
June 2, 1994

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EXECUTIVE SUMMARY

Westinghouse is the preeminent manufacturer of nuclear power plants in the United States. One half of the world's nuclear reactors are based on Westinghouse technology. Yet, many of the owners of these nuclear power plants have sued Westinghouse alleging fraud and cover-up. The problem is the steam generator used in the Westinghouse design, or, more specifically, the steam generator tube. The material used for the tubing, Inconel-600, experiences rapid degradation when used in nuclear power plant steam generators. The degradation of steam generator tubes is both a safety and financial problem for the nuclear power plant.

Whether steam generators are repaired, replaced or the nuclear power plant is retired, the resolution of steam generator problems is an expensive one for the ratepayer. Steam generator repairs can run into the millions of dollars while replacement costs range between $150 and $200 million per reactor.

Steam generator tube leaks are not only a financial liability but also represent a risk to the communities which surround the nuclear reactor. Nuclear reactors are designed to handle the disruption caused by the rupture of a single steam generator tube. However, as nuclear power plants age and the steam generator tubes become more degraded, there is the possibility that more than one tube could rupture. As few as ten ruptured tubes could cause a loss of coolant accident resulting in the meltdown of the radioactive fuel. A differing professional opinion filed in 1992 by an NRC staff member questioned the Commission's decision to allow Portland General Electric's Trojan reactor in Oregon to operate with degraded steam generator tubes. The Union of Concerned Scientists estimated that due to steam generator tube degradation the chance of a meltdown at Trojan was 300 times greater than the NRC's standard allowed. Portland General Electric decided to close the nuclear reactor and sue Westinghouse rather than pay over $200 million in repairs. However, the NRC allowed four other nuclear reactors to operate with degraded steam generator tubes.

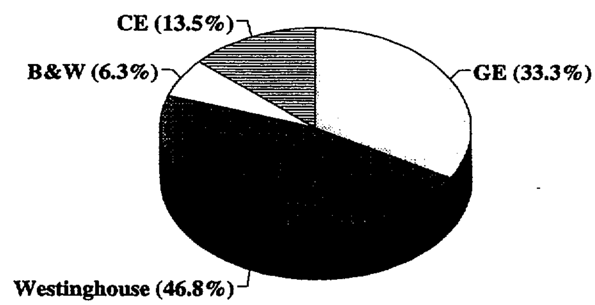
Steam generator problems have resulted in at least 14 lawsuits against Westinghouse. While the earliest suits were based on contractual claims, some of the later suits allege that Westinghouse had known about steam generator problems since 1964. However, not one of these cases has yet gone to trial. Many of the suits have been settled and Westinghouse has agreed with the utilities to seal important court documents. Westinghouse has gone to great lengths to keep information regarding steam generators away from other utilities and the public. However, citizens must bear the cost and risk of steam generator tube degradation. They should, at least, have the right to know about it.

WESTINGHOUSE LEAKS & LAWSUITS

Westinghouse Electric Corporation is a Pittsburgh-based multinational corporation. Westinghouse is the preeminent manufacturer of nuclear power plants in the United States. Westinghouse designs account for 50 operating reactors in the U.S., with two additional reactors under construction. Half of the world's nuclear reactors are based on Westinghouse technology.[1]

WESTINGHOUSE REACTORS IN THE UNITED STATES

Westinghouse designs pressurized water reactors (PWRs). PWRs incorporate a two loop design in which pressurized water in one loop carries heat from the radioactive core of the reactor into the steam generator. Steam generators contain thousands of tubes which carry the pressurized super-heated water. The heat from the radioactive loop causes the non-radioactive water on the opposite side of the tubing to flash to steam. This steam then drives the turbine to generate electricity.[2]



CE= Combustion Engineering GE= General Electric
B&W= Babcock & Wilcox

Pressurized water reactors (see adjacent diagram) have two or more steam generators depending on the plant design. Westinghouse-designed reactors incorporate recirculating steam generators. Operating experience has shown that recirculating steam generators are susceptible to a wide variety of age-related degradation.[3] This is primarily due to the fragility of the steam generator tubes made of Inconel or Alloy- 600. These tubes, which are typically .03 -.05 of an inch thick, are the boundary between the radioactive and non radioactive loop of the pressurized water reactor.

Steam generator tubes (see diagram on page 4) are susceptible to a host of aging problems. The U.S. Nuclear Regulatory Commission (NRC), Westinghouse and the utilities which own Westinghouse nuclear reactors have spent innumerable hours identifying and analyzing the many causes of steam generator tube degradation. Yet it appears that the root cause of the problem is the steam generator tube itself. The Inconel-600 is not a stable alloy when used as tubing for steam generators.[4] Although originally designed to last the life of the plant, steam generator replacement has been required at more than ten nuclear power plants since 1981.[5]

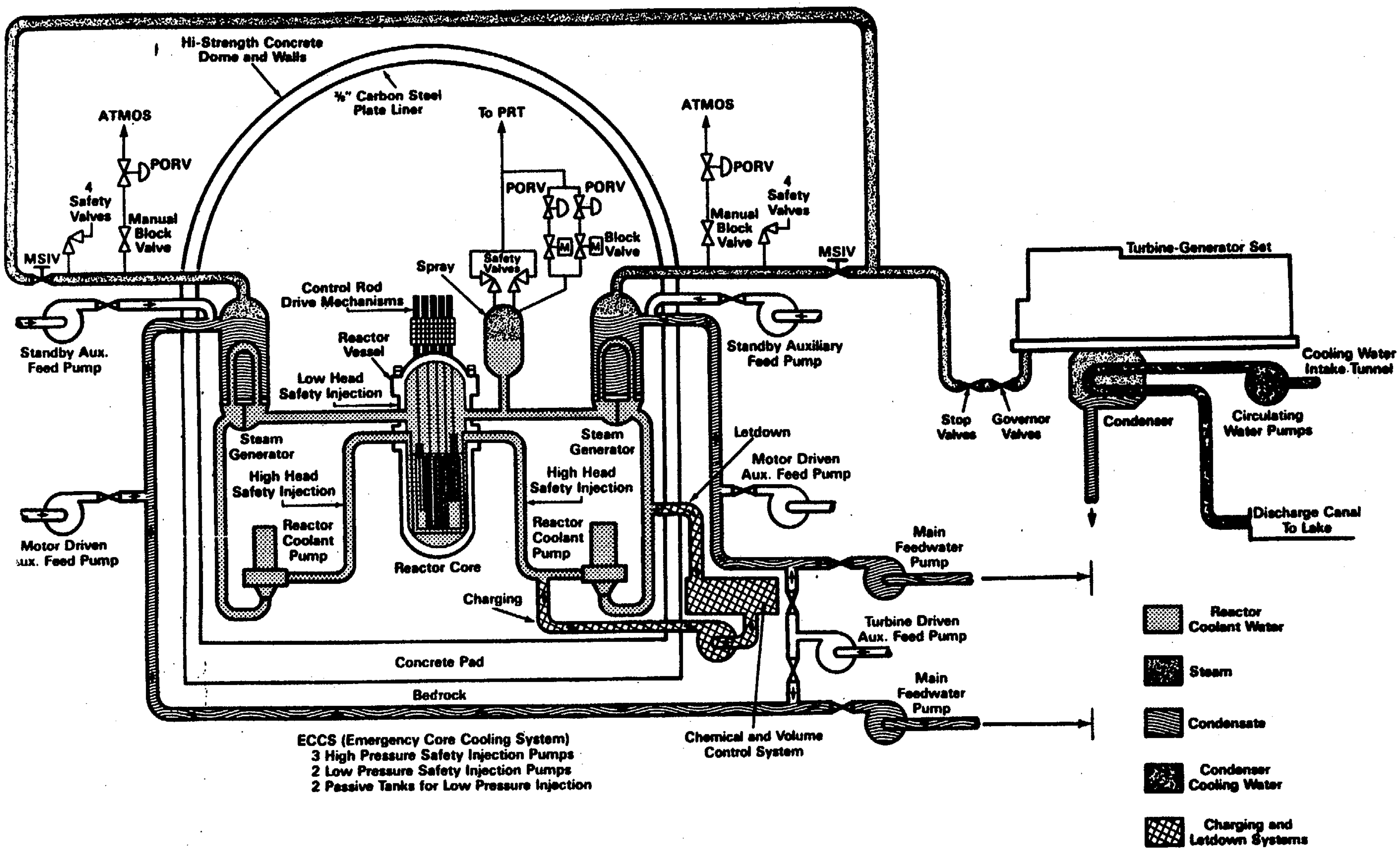
1 U.S.Nuclear Regulatory Commission, Information Digest, NUREG-1350, Vol.5, Appendix A, p. 77- 91, 1993

2 Id. at p.33

3 U.S. Nuclear Regulatory Commission,Steam Generator Operating Experience, Update for 1989-1990, NUREG/ CR-5796, December 1991. note: After this report the NRC stopped compiling steam generator data. This information is still submitted to NRC annually by each licensee.

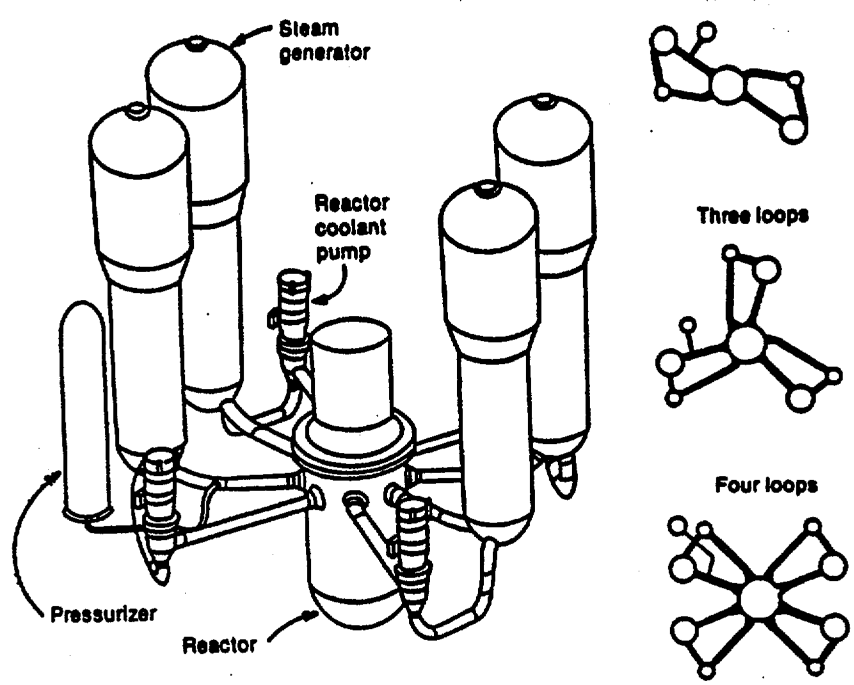
4 Memorandum for E. Beckjord, Director Office of Nuclear Reactor Research, From J. Hopenfeld, Reactor & Plant Safety Issues Branch, Division of Safety Issue Resolution, RES, Subject: A New Generic Issue: Multiple Steam Generator Leakage, March 27, 1992, p. 17.

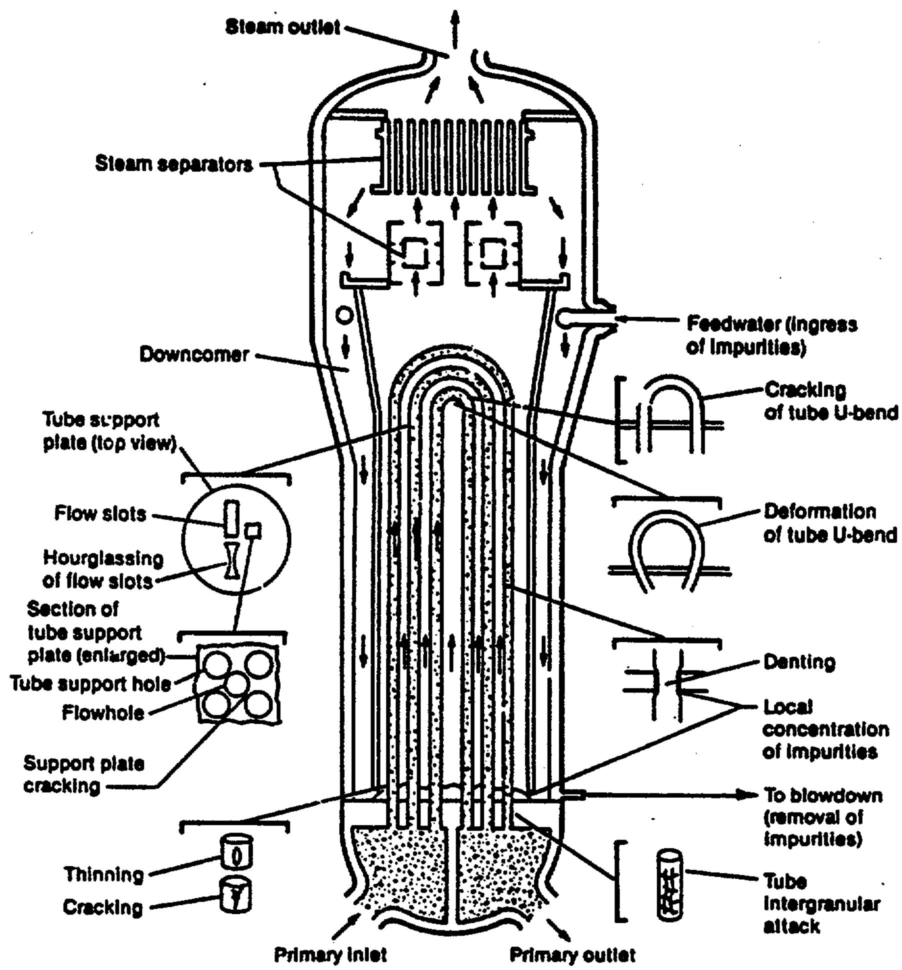
5 U.S. Congress Office of Technology Assessment, Aging Nuclear Power Plants: Managing Plant Life and Decommissioning, OTA-E-575, September 1993, p. 42



Schematic diagram of Westinghouse-designed pressurized water reactor

Source: *U.S. Nuclear Regulatory Commission, NRC Report on the January 25, 1982 Steam Generator Tube Rupture at R.E. Ginna Nuclear Power Plant, NUREG-0909, April 1982.*





Source: *U.S. Nuclear Regulatory Commission,* *Residual Life Assessment of Major Light Water Reactor Components, NUREG/CR-4731, Vol. I, June 1987*

When degraded steam generator tubes go undetected they may break, initiating a potentially disastrous sequence of events. The rupture of a steam generator tube is especially significant because it breaches the barrier between the radioactive and non -radioactive loops of the reactor. Such a breach can allow radioactive coolant to flow into the secondary system at a rate of several hundred gallons per minute. Unless plant operators respond correctly, pressure can increase in the secondary system, forcing relief valves to open and releasing radioactive gas into the environment. Such a situation occurred at the Westinghouse-designed Ginna reactor operated by Rochester Gas & Electric near Rochester, NY, in 1982. The release of 90 curies of radioactive gas could have been far worse. Had there been any damage to the core of the reactor, the bypass of the containment would have provided highly radioactive fission materials a direct pathway into the environment.[6]

Additionally, as the tubes continue to degrade there is the potential for a multiple tube rupture. Westinghouse reactors are designed to withstand the disruption caused by the rupture of only a single tube. Westinghouse reactors are not designed to withstand the rupture of multiple steam generator tubes. A multiple tube rupture is said to constitute a "beyond design basis" accident. The rupture of as few as ten steam generator tubes could result in the meltdown of the reactor fuel rods, releasing catastrophic amounts of radiation into the environment. As noted by NRC Commissioner Kenneth Rogers:

> The concern is with sudden multiple tube failures - common mode failures. For example, such failures could come about by having essentially uniform degradation of the tubes. Degradation would decrease the safety margins so that, in essence, we have a 'loaded gun,' an accident waiting to happen. Under those conditions, a pressure transient or a seismic event could rupture many tubes simultaneously. That could allow primary coolant to enter the secondary system and the resulting high pressure to lift the relief valves that are outside containment on the steam line, thus permitting primary water to bypass containment and communicate with atmosphere directly, resulting in a LOCA (loss of coolant accident).[7]

Unfortunately, the Commissioner's words proved prophetic. Precipitating Portland General Electric's (PGE) decision to close the Trojan nuclear power plant in 1992, a member of the NRC staff filed a differing professional opinion regarding the decision to allow the nuclear reactor to operate with seriously degraded steam generator tubes. The problem, according to the NRC staffer, was that "a main steam line break (MSLB) outside containment could trigger a multiple steam generator tube failure which could then result in a core melt because of depletion of coolant inventory."[8]

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6 U.S. Nuclear Regulatory Commission, NRC Report on the January 25, 1982 Steam Generator Tube Rupture at R.E. Ginna Nuclear Power Plant, NUREG-0909, April 1982.

7 Commissioner Kenneth C. Rogers, U.S. Nuclear Regulatory Commission, Nuclear Power Plant Aging: The U.S. Regulatory Perspective, International Symposium on Nuclear Power Plant Aging, August 30, 1988, p. 2

8 Memorandum for E. Beckjord, Director Office of Nuclear Reactor Research, From J. Hopenfeld, Reactor & Plant Safety Issues Branch, Division of Safety issue Resolution, RES, Subject: A New Generic Issue: Multiple Steam Generator Leakage, March 27, 1992.

NRC documents analyzed by the Union of Concerned Scientists (UCS) revealed that the risk of a meltdown at the Trojan reactor was 300 times greater than the NRC's Safety Goal standard. As the UCS letter to NRC Chairman Ivan Selin details:

> (t)he analysis conducted by the Office of Nuclear Reactor Research shows that operation with known flaws in the steam generator tubes can result in an accident in which several steam generator tubes rupture, leading to the melting of the reactor core and the release of radioactive material directly to the environment outside the reactor containment building. The staff has concluded that the probability of such an accident is over 300 times more likely than your (NRC) safety goal policy permits.[9]

Whether steam generator tubes are repaired or the steam generators are replaced altogether, the solution is expensive for the utility and ultimately the consumer. While the cost of steam generator tube repair is substantial--repairing microscopic cracks in Trojan's steam generators cost $37 million and kept the reactor shut down for a year--the cost of replacement is staggering. After fending off numerous voter referenda calling for the shutdown of Trojan, PGE decided to close the nuclear reactor and sue Westinghouse rather than replace the steam generators at a cost of at least $200 million.[10]

However, the NRC has allowed at least five other nuclear reactors to operate with degraded steam generators. These reactors include:

Joseph M. Farley Units 1 & 2
Operated by Alabama Power Co. (near Dothan, AL)

Donald C. Cook Unit 1
Operated by Indiana-Michigan Electric Co. (near Benton Harbor, MI)

Catawba Unit 1
Operated by Duke Power Co. (near Rock Hills, SC)

Braidwood Unit 1
Operated by Commonwealth Edison Company (near Joliet, IL)[11]

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9 Letter from Robert Pollard, Nuclear Safety Engineer, Union of Concerned Scientists To: Chairman Ivan Selin, James R. Curtiss, E. Gail de Planque, Forrest J. Remick, Kenneth C. Rogers, U.S. Nuclear Regulatory Commission, Subject: Multiple Steam Generator Tube Rupture Accidents, November 23, 1992, p. 1

10 "PGE Sues Westinghouse for 'Faulty Equipment' at Closed Trojan Reactor," Electric Utility Week, February 23, 1993, pp.7 - 8.

11 Hebert, Josef H. "Cracking Reactor Pipes" Associated Press, November 24, 1992; "NRC Staff Dispute Raises Questions About New Steam Generator Tube Criteria," Nucleonics Week, Nov. 26, 1992, p. 1 *[This information has been updated by conversation with U.S. NRC Public Affairs Office on May 12, 1994]*

UTILITY LAWSUITS AGAINST WESTINGHOUSE

Portland General Electric is not the only utility to sue Westinghouse over problems with steam generators. The premature aging of Westinghouse steam generators has resulted in 14 lawsuits against Westinghouse alleging that Westinghouse knew or should have known that the Inconel-600 alloy used for steam generator tubing was susceptible to degradation and would not last the 40 year life of the plant (see appendix II for complete list of lawsuits filed).

Not one of these suits has ever gone to trial and many of the court documents were filed under seal. Some details of the defects in steam generator tubes and Westinghouse actions to conceal these defects have emerged in public court documents, but broad protective orders keep important information secret. However, due to the sheer number of potential litigants, some information regarding Westinghouse steam generators has made its way into the public domain.

In 1990, when Duke Power Company sued Westinghouse over the four reactors at the McGuire and Catawba stations, the Duke complaint went well beyond the mere contractual claims alleged in earlier law suits. Duke Power Company alleged that Westinghouse had hidden problems with Iconel-600 since 1964.[12]

The Duke complaint refers to a number of internal Westinghouse memoranda in supporting its allegations of a Westinghouse cover-up including:

- "An August 17, 1964 internal memorandum on the 'Inconel corrosion problem' stated: 'Mr. Simpson was informed that he was not to inform anyone with the exception of his boss of the inconel corrosion problem, to prevent a possible hold on steam generator production.'"

- "A June 11, 1968 internal memorandum on the 'Inconel Stress Corrosion problem' in steam generator tubing has the following hand written notation by one researcher: "What do we tell them at this stage? That the alloy (Inconel) is crumbling before our eyes or that service experience is so far good?'"[3]

Like other Westinghouse steam generator lawsuits, the Duke suit was settled and many of the court documents sealed; thus denying the public access to evidence that supports or refutes Duke's allegations. The documents which are currently under seal in courts may represent the only opportunity to bring this information into the public domain. The Duke Power Company complaint also alleges that:

- "The Westinghouse Research Director-Power Systems made sure no customer learned the truth by labeling internal memorandum as follows: 'Strictly limited distribution. Cannot under any circumstances be distributed outside the company. Inside the company, recipient must have a

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12 "Duke Alleges Westinghouse Hid Inconel-600 Problems Since 1964," Nucleonics Week, April 6, 1990, p.1

13 Duke Power Company v. Westinghouse, Civ. Action No. 2:90-599-1, D.S.C., filed March 22, 1990, p. 10 &11.

14 Id. at p. 17

specific need for the information in the conduct of his assigned responsibilities. DESTROY BY BURNING OR SHREDDING."[14]

Westinghouse has gone to great lengths to keep this information from public dissemination. According to the **Legal Times,** they have even gone to the extreme of suing Shaw Pittman, Potts & Trowbridge, Newman & Holtzinger, and Chase, Rotchford, Drukker and Bogust, three law firms which represented utilities that sued Westinghouse. Westinghouse claimed that the excerpts cited above were taken from documents that were under seal and so the law firms were in breach of contract. The law firms for their part claim that the statements were evidence of a Westinghouse cover-up. The judge in the case threw out the Westinghouse complaint.[15]

When Westinghouse has not been forced into court, it has urged its disgruntled customers that it is in their best interests to resolve any steam generator disputes out of court. In a 1993 letter to the Tennessee Valley Authority, Westinghouse stated that:

> (t)his litigation is harmful to utilities, to Westinghouse and to the commercial nuclear power industry. For example, the Union of Concerned Scientists has used the litigation as a vehicle to incorrectly imply that steam generator issues pose health and safety risks to the public. This message has been communicated to the media and to legislators serving in the U.S. House of Representatives and the Senate. Although it is untrue, it is being heard, and will negatively affect us all.[16]

Westinghouse claims that the disputes over steam generators are commercial in nature and are thus better dealt with through commercial means. However, the following paragraph indicates that Westinghouse's desire to avoid trial is also based on keeping the dispute out of the public arena. The Westinghouse letter states that:

> (i)f the current litigation process proceeds through the public trial stage, we will have created a platform for those opposed to nuclear power to unfairly attack both the safety and economics of operating nuclear power plants. The public spectacle that steam generator trials will create will further threaten the nuclear power option for the future of our nation.[17]

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15 Greg Rushford, "Westinghouse Campaign To Hide Generator Flaws Shows Signs of Cracking," Legal Times, November 18,1991, p.1

16 Letter to Oliver Kingsley, President, Generation, Tennessee Valley Authority, From John B. Yasinsky, Group President, Westinghouse Electric Corporation, March 8, 1993, p.2 *(Attached as Appendix III)*

17 Id. at p. 2

18 Littlejohn v. BIC Corp., 851 F.2d 673, 677 -78 (3d Cir. 1988)

WESTINGHOUSE SETTLEMENTS AND THE PUBLIC'S RIGHT TO KNOW

The American legal tradition has long recognized the common law right of access to public records and documents including judicial documents. The existence of this right predates the Constitution and is legally beyond dispute.[18]

The public's right of access serves several functions with in our legal system and society. The ability of citizens to inspect and copy judicial records helps to foster confidence in the judicial system. As noted in a previous decision involving Westinghouse nuclear reactors, "the bright light cast upon the judicial process by public observation diminishes the possibilities for injustice, incompetence, perjury and fraud."[19]

However, in each of the Westinghouse steam generator lawsuits, many court documents were sealed. When utility companies settle their disputes with Westinghouse secrecy is obviously a bargaining chip. As demonstrated in its letter to TVA, Westinghouse's interest in avoiding court and sealing the court documents has little to do with trade secrets or other confidential business information. Westinghouse is merely attempting to keep its already tarnished image from suffering any further harm.

Utilities that own and operate Westinghouse nuclear reactors are as culpable as Westinghouse when it comes to keeping information from the public. While utilities can use the "spectacle" of a public steam generator trial to secure an expeditious and beneficial settlement from Westinghouse, they have no interest in drawing adverse attention to their own nuclear reactors. The two parties to the lawsuit agree to seal the court documents because it serves the interest of both Westinghouse and the utility.

The public has a right to know of any and all threats which nuclear reactors pose to their health, families, homes, and communities. If Westinghouse knowingly sold defective steam generators, the public has the right to know. The problems associated with steam generator tube degradation are more than merely commercial in nature. In a best case scenario, the utility will discover the degraded steam generators and decide whether to repair them, replace them or retire the reactor. In the worst case, degraded steam generator tubes can result in the meltdown of the reactor's fuel and a catastrophic release of radiation into the environment. In either case, citizens will bear the risk and cost of degraded steam generator tubes. They should, at least, have the right to know about it.

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19 Republic of the Philippines v. Westinghouse Electric Corporation, 949 F.2d 653, 660 (3rd Cir.1991)

APPENDIX 1
LISTING OF WESTINGHOUSE REACTORS

Reactor: Beaver Valley 1
Utility: Duquesne Light Co.
Location: 17 miles W of McCandless, PA
OL Date: July 2, 1976

Reactor: Beaver Valley 2
Utility: Duquesne Light Co.
Location: 17 miles W of McCandless, PA
OL Date: August 14, 1987

Reactor: Braidwood 1
Utility: Commonwealth Edison Co.
Location: 24 miles SSW of Joilet, IL
OL Date: July 2, 1987

Reactor: Braidwood 2
Utility: Commonwealth Edison Co.
Location: 24 miles SSW of Joilet, IL
OL Date: March 20, 1988

Reactor: Byron 1
Utility: Commonwealth Edison Co.
Location: 17 miles SW of Rockford, IL
OL Date: February 14, 1985

Reactor: Byron 2
Utility: Commonwealth Edison Co.
Location: 17 miles SW of Rockford, IL
OL Date: January 30, 1987

Reactor: Callaway
Utility: Union Electric Co.
Location: 10 miles SE of Fulton, MO
OL Date: October 18, 1984

Reactor: Catawba 1
Utility: Duke Power Co.
Location: 6 miles NNW of Rock Hill, SC
OL Date: January 17, 1985

Reactor: Catawba 2
Utility: Duke Power Co.
Location: 6 miles NNW of Rock Hill, SC
OL Date: May 15, 1986

Reactor: Comanche Peak 1
Utility: Texas Utilities Electric Co.
Location: 4 miles N of Glen Rose, TX
OL Date: April 17, 1990

Reactor: Comanche Peak 2
Utility: Texas Utilities Electric Co.
Location: 4 miles N of Glen Rose, TX
OL Date: April 6, 1993

Reactor: D.C. Cook 1
Utility: Indiana/Michigan Power Co.
Location: 11 miles S of Benton Harbor, MI
OL Date: October 25, 1974

Reactor: D.C. Cook 2
Utility: Indiana/Michigan Power Co.
Location: 11 miles S of Benton Harbor, MI
OL Date: December 23, 1977

Reactor: Diablo Canyon 1
Utility: Pacific Gas & Electric Co.
Location: 12 miles W of San Luis Obispo, CA
OL Date: November 2, 1984

Reactor: Diablo Canyon 2
Utility: Pacific Gas & Electric Co.
Location: 12 miles W of San Luis Obispo, CA
OL Date: August 26, 1985

Reactor: Ginna
Utility: Rochester Gas & Electric Co.
Location: 20 miles NE of Rochester, NY
OL Date: December 10, 1984

Reactor: Haddem Neck
Utility: CT Yankee Atomic Power Co.
Location: 13 miles E of Meriden, CT
OL Date: December 27, 1974

Reactor: H.B. Robinson 2
Utility: Carolina Power & Light
Location: 26 miles from Florence, SC
OL Date: September 23, 1970

Reactor: Indian Point 2
Utility: Consolidated Edison Co.
Location: 24 miles N of New York, NY
OL Date: September 28, 1973

Reactor: Indian Point 3
Utility: New York Power Authority
Location: 24 miles N of New York, NY
OL Date: April 6, 1976

Reactor: Joseph M. Farley 1
Utility: Southern Nuclear Operating Co.
Location: 18 miles SE of Dothan, AL
OL Date: June 25, 1977

Reactor: Joseph M. Farley 2
Utility: Southern Nuclear Operating Co.
Location: 18 miles SE of Dothan, AL
OL Date: March 31, 1981

Reactor: Kewaunee
Utility: Wisconsin Public Service Corp.
Location: 27 miles E of Green Bay, WI
OL Date: December 21, 1973

Reactor: McGuire 1
Utility: Duke Power Co.
Location: 17 miles S of Charlotte, NC
OL Date: July 8, 1981

Reactor: McGuire 2
Utility: Duke Power Co.
Location: 17 miles S of Charlotte, NC
OL Date: May 27, 1983

Reactor: Millstone 3
Utility: Northeast Nucler Energy Co.
Location: 3 miles E-NE of New London, CT
OL Date: January 31, 1986

Reactor: North Anna 1
Utility: Virginia Electric & Power Co.
Location: 40 miles NW of Richmond VA
OL Date: April 1, 1978

Reactor: North Anna 2
Utility: Virginia Electric & Power Co.
Location: 40 miles NW of Richmond VA
OL Date: February 19, 1971

Reactor: Point Beach 1
Utility: Wisconsin Electric Power Co.
Location: 13 miles N-NW of Manitowac, WI
OL Date: October 5, 1970

Reactor: Point Beach 2
Utility: Wisconsin Electric Power Co.
Location: 13 miles NNW of Manitowac, WI
OL Date: March 8, 1973

Reactor: Prairie Island 1
Utility: Northern States Power Co.
Location: 28 miles SE of Minnaepolis, MN
OL Date: April 5, 1974

Reactor: Prairie Island 2
Utility: Northern States Power Co.
Location: 28 miles SE of Minnaepolis, MN
OL Date: October 29, 1974

Reactor: Salem 1
Utility: Public Service Electric & Gas Co.
Location: 18 miles S of Wilmington, DE
OL Date: December 1, 1976

Reactor: Salem 2
Utility: Public Service Electric & Gas Co.
Location: 18 miles S of Wilmington, DE
OL Date: May 20, 1981

Reactor: San Onofre 1
Utility: Southern California Gas & Electric and San Diego Gas & Electric
Location: 4 miles SE of San Clemente, CA
OL Date: SHUT DOWN November 30, 1992

Reactor: Seabrook 1
Utility: North Atlantic Energy Service Co.
Location: 13 miles S of Portsmouth, NH
OL Date: March 15, 1990

Reactor: Sequoyah 1
Utility: Tennessee Valley Authority
Location: 10 miles NE of Chattanooga, TN
OL Date: September 17, 1980

Reactor: Sequoyah 2
Utility: Tennessee Valley Authority
Location: 10 miles NE of Chattanooga, TN
OL Date: September 15, 1981

Reactor: Shearon Harris
Utility: Carolina Power & Light
Location: 20 miles SW of Raliegh, NC
OL Date: January 12, 1987

Reactor: South Texas Project 1
Utility: Houston Light & Power Co.
Location: 12 miles SSW of Bay City, TX
OL Date: March 22, 1988

Reactor: South Texas Project 2
Utility: Houston Light & Power Co.
Location: 12 miles SSW of Bay City, TX
OL Date: March 28, 1989

Reactor: Summer
Utility: South Carolina Electric & Gas Co.
Location: 26 miles NW of Columbia, SC
OL Date: November 12, 1982

Reactor: Surry 1
Utility: Virginia Electric & Power Co.
Location: 17 miles NW of Newport News, VA
OL Date: May 25, 1972

Reactor: Surry 2
Utility: Virginia Electric & Power Co.
Location: 17 miles NW of Newport News, VA
OL Date: January 29, 1973

Reactor: Trojan
Utility: Portland General Electric Co.
Location: 32 miles N of Portland, OR
OL Date: SHUTDOWN November 9, 1992

Reactor: Turkey Point 3
Utility: Florida Power & Light Co.
Location: 25 miles S of Miami, FL
OL Date: July 19, 1972

Reactor: Turkey Point 4
Utility: Florida Power & Light Co.
Location: 25 miles S of Miami, FL
OL Date: April 10, 1973

Reactor: Vogtle 1
Utility: Georgia Power Co.
Location: 26 miles SE of Augusta, GA
OL Date: March 16, 1987

Reactor: Vogtle 2
Utility: Georgia Power Co.
Location: 26 miles SE of Augusta, GA
OL Date: March 31, 1989

Reactor: Watts Bar 1
Utility: Tennessee Valley Authority
Location: 10 miles S of Spring City, TN
OL Date: NO OPERATING LICENSE

Reactor: Watts Bar 2
Utility: Tennessee Valley Authority
Location: 10 miles S of Spring City, TN
OL Date: NO OPERATING LICENSE

Reactor: Wolf Creek 1
Utility: Wolf Creek Nuclear Operating Corp.
Location: 4 miles NE of Burlington, KS
OL Date: June 4, 1985

Reactor: Zion 1
Utility: Commonwealth Edison Co.
Location: 40 miles N of Chicago, IL
OL Date: October 19, 1973

Reactor: Zion 2
Utility: Commonwealth Edison Co.
Location: 40 miles N of Chicago, IL
OL Date: November 14, 1973

OL Date = Date of operating license

APPENDIX 2
LISTING OF WESTINGHOUSE LAWSUITS

Utility: Florida Power & Light Co.
Reactor: Turkey Point 3 & 4
Citation: Florida Power & Light Company v. Westinghouse Electric Corporation, No. 78-1896-CIV (S.D.Fla., filed May 5, 1978).

Utility: Consolidated Edison Co.
Reactor: Indian Point 2
Citation: Consolidated Edison Company of New York v. Westinghouse Electric Corporation, No. 82 Civ. 3504 (MEL) (S.D.N.Y., filed May 28, 1982).

Utility: Southern California Edison/San Diego Gas & Electric
Reactor: San Onofre 1
Citation: Southern California Edison Company v. Westinghouse Electric Corporation, No. 83-1985 (S.D.Cal., filed March 31, 1983).
San Diego Gas & Electric Co. V. Westinghouse Electric Corporation, No. 83-1986 (S.D.Cal., filed March 31, 1983).

Utility: Carolina Power & Light
Reactor: Shearon Harris
Citation: Carolina Power & Light Company v. Westinghouse Electric Corporation, No. 89-1383 (W. D. Pa., filed April 28, 1989).

Utility: Duke Power Co.
Reactor: Catawba 1 & 2, McGuire 1 & 2
Citation: Duke Power Company v. Westinghouse Electric Corporation, No. 2-90-599-1 (D.S.C., filed March 22, 1990).

Utility: South Carolina Electric & Gas Co.
Reactor: Summer
Citation: South Carolina Electric & Gas Company v. Westinghouse Electric Corporation, No. 2-90-598-1 (D.S.C., filed March 22, 1990).

Utility: Carolina Power & Light
Reactor: H.B. Robinson
Citation: Carolina Power & Light Company v. Westinghouse Electric Corporation, No. 2-90-636-1 (D.S.C., filed March 27, 1990).

Utility: Commonwealth Edison Co.
Reactor: Braidwood 1& 2, Byron 1 & 2, Zion 1 & 2
Citation: Commonwealth Edison Company v. Westinghouse Electric Corporation, No. 90-15993 (Cook County Circuit Court, filed Oct. 12, 1990).

Utility: Houston Light & Power Co., Cities of Austin & San Antonio, Texas
Reactor: South Texas Project 1 & 2
Citation: The City of San Antonio, Texas; The City of Austin, Texas; Central Power & Light Company and Houston Lighting & Power Company v. Westinghouse Electric Corporation, No. 90-SD-0684-C (D. Ct. of Matagorda County, Texas, filed Oct. 15, 1990).

Utility: Duquesne Light Co.
Reactor: Beaver Valley 1 & 2
Citation: Duquesne Light Company, The Cleveland Electric Illuminating Company, The Toledo Edison Company, Ohio Edison Company and Pennsylvania Power Company v. Westinghouse Electric Corporation, No. 91-720 (W.D.Pa., filed April 30, 1991).

Utility: Portland General Electric Co.
Reactor: Trojan
Citation: The City of Eugene, Oregon, Acting By and Through the Eugene Water & Electric Board v. Westinghouse Electric Corporation, No. 93-60 (D. Or., filed Feb. 15, 1993).[This suit has been consolidated in W.D.Pa with the one filed by Portland General Electric]

Utility: Northern States Power Co.
Reactor: Prairie Island 1 & 2
Citation: Northern States Power Company v. Westinghouse Electric Corporation, No. 4-93-680 (D. Minn., filed July 14, 1993).

Utility: Furnas Centrais Eletricas SA [Brazil]
Reactor: Angra 1
Citation: Furnas Centrais Electricas S.A. v. Westinghouse Electric Corporation, No. 87 Civ. 4934 (WK) (S.D.N.Y., filed July 10, 1987).

A Roll of the Dice

NRC's Efforts to Renew Nuclear Reactor Licenses

by Jim Riccio
with Michael Grynberg

Public Citizen

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Executive Summary

For the second time in four years, the U.S. Nuclear Regulatory Commission has issued a "final" rule that would allow utilities to operate nuclear power plants beyond the 40 year term of the current license. The nuclear industry's first attempt to renew a nuclear reactor license resulted in the shutdown of Yankee Atomic's Rowe reactor, a perennial industry leader. In the wake of the Rowe debacle, the nuclear industry pressed the Commission to change the license renewal rule. Acquiescing to industry demands, the NRC's rewrite of the rule concluded that the existing regulatory process will continue to mitigate the effects of aging to provide an acceptable level of safety in the period of extended operation. However, if that were actually the case, Yankee Rowe would still be splitting atoms rather than being decommissioned. The NRC's reliance on "current regulatory processes" to ensure the safety of nuclear reactors in the absence of any criteria, inspection or licensee submittal of a reactor's current licensing basis constitutes an abrogation of the Commission's statutory responsibility. Merely declaring that all reactors meet their current licensing basis does not make it so.

License renewal has proven to be a high stakes gamble. Nuclear utilities can gain 20 more years of operation, and the commensurate economic benefit, while shifting the risk of future operation from the stockholder to the ratepayer. The decision either to retire a nuclear reactor or renew its license will be based on a combination of economic, safety and political considerations. As the experience of Yankee Rowe demonstrated, the existing regulatory process may not mitigate the effects of aging and even the best run reactors may have difficulty proving they can operate safely for an additional 20 years. Furthermore, the Rowe experience demonstrates that examination of the licensing basis for extended operation could jeopardize the remaining years on the current license.

Yankee Atomic's efforts to extend Rowe's license raised the issue of the strength of the reactor pressure vessel, the steel crucible that holds the radioactive fuel rods. Bombarded by radiation, the reactor vessel becomes embrittled, which increases its susceptibility to cracking. If the vessel cracks a meltdown is virtually inevitable. The NRC has attempted to get ahead of the embrittlement curve by allowing reactors that are becoming severely embrittled to justify continued operation. While other utilities have pencil whipped their embrittlement calculations to conform with regulations, the public can take little solace in these computations. Although NRC estimates indicated that Rowe would not be in danger of embrittlement until 2029, the reactor is now being decommissioned because Yankee Atomic could not prove that its reactor pressure vessel was sound.

The reactor vessel is not the only component of a nuclear reactor susceptible to premature degradation. The steam generators used in Westinghouse and Combustion Engineering designs have experienced rapid degradation, necessitating lengthy reactor shut downs for repair and replacement. Steam generator tube failure is both a safety and economic problem for utilities. The rupture of as few as ten steam generator tubes could result in the meltdown of the reactor fuel rods, releasing catastrophic amounts of radiation into the environment. Steam generator tube degradation has led NRC Commissioner Kenneth Rogers to conclude that "in essence, we have a 'loaded gun,' an accident waiting to happen."

Whether steam generator tubes are sleeved, plugged or replaced altogether, the solution is an expensive one for the utility and ultimately the consumer. After fending off numerous voter referenda calling for the shutdown of the Trojan reactor in Oregon, the utility decided to close the nuclear reactor and sue Westinghouse rather than replace the steam generators at a cost of at least $200 million. Considering the steep cost of steam generator replacement and the uncertainty of recouping the investment, some utilities may decide not to replace their steam generators and forgo the opportunity to renew their operating licenses. However, the prospect of reactors limping along with degraded steam generators is neither in the interests of the nuclear utilities nor in the interests of public health and safety.

The NRC has long recognized that reactor vessels can crack precipitating a Chernobyl-like catastrophe. However, regulators have been slow to acknowledge that radiation-induced damage to other parts of the nuclear reactor could be just as catastrophic. More than a dozen General Electric designed nuclear reactors in the U.S. and abroad have evidence of cracking in the reactor core shroud—a metal cylinder surrounding the reactor fuel rods. The owners of these boiling water reactors have contended that the core shroud is of little safety significance. However, the NRC has acknowledged that cracking of the core shroud could damage the radioactive fuel rods, prohibit the insertion of control rods and lead to a meltdown of the reactor.

The problem of core shroud cracking is now believed to affect most, if not all, older General Electric reactors. However, "older" is a relative term. Cracking has been found in reactors that have operated for less than 10 years, only one quarter of a reactor's operating license. Replacement of the core shroud will cost millions of dollars and calls into question the economic viability of many of these nuclear reactors. While the issue of the core shroud cracking has not yet resulted in the permanent shutdown of a reactor, it does indicate that extended operation of boiling water reactors is anything but certain. It is doubtful whether any reactor could economically justify the two year down time estimated for core shroud replacement. Even if reactors can operate with hastily repaired core shrouds, the degradation of other reactor internals will pose both safety and economic problems. The degradation of reactor internals, including the core shroud, may lead boiling water reactors to shutdown prior to any renewal term.

The economics of license renewal are problematic at best. Increased competition in the wholesale electricity market is already placing serious economic pressure on nuclear utilities.

As nuclear reactors age, utilities will be forced to spend large amounts of capital to repair or replace components such as steam generators and core shrouds. Absent the ability to amortize or write off these large capital additions over an additional 20 year period, nuclear utilities will be forced by economics and safety to retire nuclear reactors prior to license expiration. Early shutdown of nuclear reactors may result in stranding large utility investments and under-funding utility commitments such as decommissioning and waste disposal. Utilities do not want to make their investors swallow these costs. Through license renewal, they can shift the risk of the bad investment in nuclear power from the investor to the ratepayer.

However, if utilities can not recoup their investments after nuclear reactors are retired, they may continue to operate unsafe and uneconomical reactors. If nuclear reactors are too expensive to operate but utilities are reluctant to close them down due to stranded investments, we could have an economic recipe for disaster. More than half of the nuclear reactors in the U.S. are already more expensive to operate than the cost of replacement power. If nuclear reactors can not compete in the current market, the prospects for license renewal would appear dim and fading. Even if nuclear utilities can bring O&M costs under control and reverse historic trends, the combination of cheap replacement power, large capital additions and a growing high-level waste problem will likely doom many renewal efforts.

Extending the licenses of operating reactors will only increase the amount of high level waste with which future generations will have to contend. Neither the nuclear industry nor the government has developed the means to isolate high-level radioactive waste from the environment for the duration of its hazardous life. Coping with the wastes that result from any proposed license extension looms as an unknown cost and possible taxpayer liability of the NRC's license renewal rule.

Serious doubt exists as to whether a geologic repository, once expected in 1998 but now envisioned by optimistic timetables in 2010, will ever be available. In fact, the feasibility of "disposing" of this nation's nuclear waste in an underground storage facility has never been more in doubt. Scientists at Los Alamos National Laboratory in New Mexico, fear that high level radioactive wastes stored in the proposed repository at Yucca Mountain could eventually explode. Even nuclear scientists are beginning to understand what environmentalists and public interest advocates have been arguing for decades, that one can not merely "dispose" of radioactive wastes that have a hazardous life of over 240,000 years.

In the absence of either a repository or interim facility, many utilities are running out of space in their spent fuel pools. As many as 14 reactor fuel pools will reach capacity by the year 2000. Faced with the politically unsavory task of attempting to site additional dry cask storage at the reactor site, utilities with reactors nearing the end their licenses may opt to avoid the political and economic costs and decide instead to retire the nuclear reactor.

No nuclear reactor has yet operated for the 40 year term of its operating license. It appears increasingly unlikely that older reactors will remain competitive with alternative sources of electricity. Given the myriad safety problems facing aging reactors, many nuclear power plants will have difficulty even lasting to the end of their current licenses. When one also considers the utilities' ever growing high-level radioactive waste problem and dismal economics, operation of nuclear reactors beyond 40 years seems like a pipe dream. The NRC's attempt to extend the operating licenses of nuclear reactors is little more than a regulatory "slight of hand." License renewal would allow utilities to shift the financial risk of nuclear power from the investor to the ratepayer by amortizing the costs of continued operation over an additional 20 year period.

The new license renewal rule has no foundation in safety. Merely relying upon the current regulatory process to protect the public while failing to require that reactors document compliance with the current licensing basis is an abdication of the Commission's responsibility. Absent any enforceable standard for renewal, the NRC's new license renewal rule appears to be little more than a rubber stamp. If the Commission were truly concerned with safety, it would ensure that aging, unsafe and uneconomical reactors are shut down. Rather than extending the operation of nuclear reactors, the NRC should develop objective criteria on which to base a decision to retire a nuclear reactor. Unfortunately, the Commission has never done so.

Introduction

The Nuclear Regulatory Commission is attempting to allow utilities to operate nuclear reactors beyond the 40 year term of the current license. License renewal has proven to be a high stakes gamble. Nuclear utilities can gain 20 more years of operation, and the commensurate economic benefit, while shifting the risk of future operation from the stockholder to the ratepayer. The downside, however, is that the review needed to renew a nuclear reactor license may cost the utility the remaining life on the current license.

The odds depend upon which nuclear reactor is rolling the dice. Experience has shown that even the best run nuclear reactors may have trouble proving they can operate safely for an additional 20 years. The nuclear industry's first attempt to renew a nuclear power plant license resulted in the shutdown of Yankee Atomic's Rowe reactor, once a perennial industry leader.

No commercial nuclear reactor has yet operated for 40 years. The oldest reactor, Big Rock Point in Michigan, is 32 years old and is already economically non-competitive with other sources of electricity.

Table 1

Shutdown	Reactor	State	Operation
11/30 92	San Onofre 1	CA	< 26 years
11/09/92	Trojan	OR	< 17 years
10/01/91	Yankee Rowe	MA	< 28 years
8/18/89	Fort St. Vrain	CO	< 16 years
6/28/89	Shoreham	NY	< 3 months
6/07/89	Rancho Seco	CA	< 15 years

Six nuclear reactors have been retired in the last 6 years (see Table 1 below).

On average, these reactors operated for less than half of the operating license. Even when excluding anomalies like the meltdown at Three Mile Island near Harrisburg, PA and the political meltdown that shut Shoreham in New York, retired reactors averaged only 20 years of operation. With increased competition in the electricity industry, nuclear reactors are facing a mid-life crisis and the NRC is attempting to give them a new lease on life.

The decision either to retire a nuclear reactor or renew its license will be based on a combination of economic, safety and political considerations. This report will examine a number of issues that have led or may soon lead to the early retirement of U.S. nuclear power plants. These issues include:

Steam Generator Tube Degradation
Reactor Pressure Vessel Embrittlement
Reactor Core Shroud Cracking
Economic Competitiveness
And
High Level Waste/Spent Fuel Storage

These factors, along with political considerations regarding the public perception of nuclear power, have led to the early demise of several reactors and threaten the viability of several more nuclear reactors across the country.

I. A Brief History of NRC's Attempts to Relicense Reactors

The First Rule: Two Principles of License Renewal

On December 13, 1991, the U.S. Nuclear Regulatory Commission published the "final" rule regarding renewal of nuclear power plant licenses.[1] At that time, the Commission premised its assumption that nuclear power plants could operate safely for an additional 20 years on two principles.

> "The first principle is that, with the exception of age-related degradation unique to license renewal and possibly some few other issues related to safety only during extended operation of nuclear power plants, the regulatory process is adequate to ensure that the licensing bases of all currently operating plants provide and maintain an acceptable level of safety so that operation will not be inimical to public health and safety or common defense and security. Continuing this regulatory process in the future will ensure that this principle remains valid during any renewal term if the process is modified to include age related degradation unique to license renewal.
>
> * * *
>
> The second and equally important principle is that each plant's current licensing basis must be maintained during the renewal term, in part, through a program of age-related degradation management for systems, structures and components that are important to license renewal as defined in the final rule."[2]

The Commission's original rule was premised on the assumption that the current licensing basis—those regulations, requirements and commitments which constitute the nuclear reactor's operating license—would be sufficient to protect the public health and safety so long as it was modified to account for age related degradation. Under the original renewal rule, members of the public could not challenge the sufficiency of or question the compliance with a reactor's current licensing basis. According to the Commission, the current licensing basis (CLB) for all reactors is sufficient and all reactors are in compliance with the CLB.

Under the license renewal rule the licensee need only compile a list of the documents that constitute the reactor's current licensing basis. The NRC is neither going to review these documents nor confirm that the reactor is in compliance with the regulations imposed under the current license. Yet, the NRC acknowledges that the current licensing basis for the nations nuclear power plants is "outdated and oftentimes poorly recorded."[3]

Advisory Committee Concerns

The myth that the current licensing basis is sufficient and that all plants are in compliance with the licensing basis fails the tests of both logic and reality. The NRC's assumption is based upon the specious argument that operating without a meltdown for a finite period of time means that safety is adequate. Hal Lewis, Subcommittee Chair of the Advisory Committee on Reactor Safeguards, recognized this fallacy when the ACRS took up the original license renewal rule. Mr. Lewis stated that:

> "the general argument that the fact that one has operated safely for a finite period of time proves that the safety level is adequate is just not statistically right, because there isn't that much history in the industry. And it's a trap. Because other agencies, for example, people have used the argument that they had 24 successful Shuttle flights, to show the level of safety was adequate. And in retrospect, after one disaster, it turned out not to be. The

Soviets, after Chernobyl, suddenly discovered that the level of safety they had before Chernobyl was not adequate. But the day before Chernobyl they would have said it was adequate on the basis of operating history.

So it is a general trap, a psychological trap, to believe that because something has not happened, you are doing just fine."[4]

Although the ACRS Commissioner's comments were directed at the original rule they apply to the current NRC proposal as well. Mr. Lewis continued his critique of the renewal rule noting that:

"**the Commission certainly doesn't know that its current regulatory process provides adequate protection to the public.** It has declared that it does, and it's the operating definition, but the Commission has also promulgated safety goals and the commission doesn't know that the current licensing basis will meet the safety goals, although it believes it to be the case."[5]

ACRS Commissioner Lewis recognized the arbitrary nature of the Commission's key assumption in license renewal. Unfortunately, the NRC continues to labor under this false assumption.

The General Design Criteria: Older Reactors Need Not Comply

The NRC will not verify licensee compliance with the current licensing basis, nor will the Commission require that 63 nuclear reactors conform to the minimum design standards necessary to protect the public health and safety. These standards, known as the General Design Criteria or GDC, established base-line requirements for nuclear reactor design and cover a range of topics including fire protection; inspection and testing of electrical power systems; containment design and testing; inspection and testing of the emergency core cooling system as well as fuel handling and storage requirements. According to the NRC, the General Design Criteria was:

necessary to provide reasonable assurance that the facility can be operated without undue risk to the public health and safety. The phrase 'without undue risk' represents the statutory requirements of Section 182 of the Atomic Energy Act for 'adequate protection of the public health and safety' The use of the statutory standard implies that the GDC represents the minimum standard for all licensees.[6]

However, the Commission decided that the General Design Criteria would not apply to nuclear reactors that received a construction permit prior to May 21, 1971.[7] The reactors listed in Table 2 (see page 4) have been exempted from the General Design Criteria.

The NRC has determined that for these 63 nuclear reactors the GDC will not apply. The Commission concluded that "current regulatory processes are sufficient to ensure that plants continue to be safe and comply with the intent of the GDC."[8] The NRC's reliance on "current regulatory processes" to ensure the safety of nuclear reactors in the absence of any criteria, inspection or licensee submittal of a reactor's current licensing basis constitutes an abrogation of the Commission's statutory responsibility. Merely declaring that all reactors meet their current licensing basis does not make it so.

NRC's Lead Plant Program: Yankee Rowe & Monticello

Yankee Atomic's Rowe reactor in Western Massachusetts and Northern States Power's Monticello reactor in Minnesota were to be the first reactors to submit license renewal requests to the NRC. Originally, the licensees were to submit applications in 1991 as part of the DOE-sponsored lead plant program.[9] By the end of 1992, however, Yankee Rowe had been permanently shut down and Monticello had postponed its license renewal application indefinitely.

While the NRC could merely declare that all reactors met their current licensing basis, Yankee Rowe had difficulty proving this point to the NRC staff. The issue concerning the staff revolved around the reactor pressure vessel (RPV). The vessel is the steel crucible that holds the nuclear fuel rods. The NRC staff was concerned that after years of being bombarded by neutrons from the fission reaction Yankee Rowe's vessel had become embrittled to the point where an accident could threaten the integrity of the vessel.

On September 5, 1990, NRC's senior metallurgist Dr. Pryor N. Randall told the Advisory Committee on Reactor Safeguards (ACRS) that he could not justify Yankee Rowe's continued operation.

> You know there are three ways the NRC handles this. If we've had a failure, you know, it's pretty simple. The industry knows how to deal with failures. About all we do is find out, well, is it generic? Is this an epidemic starting or not? If it is, we pass the word around to the utilities and so forth.
>
> The second approach that we have that there be some staff concern, raise it with the utility. They submit a report, we review it, we have questions, we ask more, they submit more reports, we want measurements. Sometimes we want long range things. They go along pretty cheerfully because they know that in most cases when it's all done, the questioner is exhausted and we'll say, it's alright.
>
> I'm afraid that we're into that here and I don't think we should, for safety reasons, on this plant. The third approach, which we don't use very often, is to declare a situation unacceptable. That's where I'm coming from.
>
> **I can not agree that restart is safe or justified.**
>
> I know it's a minority report but I have to show you what I have in mind.[10]

"Setting aside all of the PRA stuff," Randall continued, **"if we let them start up with this level, we're simply gambling**

Table 2

Reactor	State	Utility
Arkansas Nuclear 1	AR	Entergy
Beaver Valley 1	PA	Duquesne Light Company
Big Rock Point	MI	Consumers Power
Browns Ferry 1, 2 & 3	AL	Tennessee Valley Authority
Brunswick 1 & 2	NC	Carolina Power & Light
Calvert Cliffs 1 & 2	MD	Baltimore gas & Electric
Cooper	NE	Nebraska Public Power
Crystal River 3	FL	Florida Power Corp.
Davis Besse	OH	Toledo Edison
Cook 1 & 2	MI	Indiana/Michigan Power
Diablo Canyon 1 & 2	CA	Pacific Gas & Electric
Dresden 2 & 3	IL	Commonwealth Edison
Duane Arnold	IA	Iowa Electric Light & Power
Hatch 1	GA	Southern Nuclear Operating Co.
Fort Calhoun	NE	Omaha Public Power
Ginna	NY	Rochester Gas & Electric
Haddam Neck	CT	CT Yankee Atomic Power Co.
H.B. Robinson 2	SC	Carolina Power & Light
Indian Point 2 & 3	NY	Consolidated Edison
Fitzpatrick	NY	New York Power Authority
Kewaunee	WI	Wisconsin Public Service Corp.
Maine Yankee	ME	Maine Yankee Atomic Power Co.
Millstone 1 & 2	CT	Northeast Utilities
Monticello	MN	Northern States Power
Nine Mile Point 1	NY	Niagara Mohawk Power Co.
North Anna 1 & 2	VA	Virginia Electric & Power Co.
Oconee 1, 2 & 3	SC	Duke Power Co.
Oyster Creek	NJ	GPU Nuclear
Palisades	MI	Consumers Power Co.
Peach Bottom 2 & 3	PA	Philadelphia Electric Co.
Pilgrim	MA	Boston Edison Co.
Point Beach 1 & 2	WI	Wisconsin Electric Power Co.
Prairie Island 1 & 2	MN	Northern States Power
Quad Cities 1 & 2	IL	Commonwealth Edison
Salem 1 & 2	NJ	Public Service Electric & Gas
Sequoyah 1 & 2	TN	Tennessee Valley Authority
St. Lucie 1	FL	Florida Power Co.
Surry 1 & 2	VA	Virginia Electric & Power Co.
Three Mile Island 1	PA	GPU Nuclear
Turkey Point 3 & 4	FL	Florida Power & Light
Vermont Yankee	VT	VT Yankee Nuclear Power Corp.
Zion 1 & 2	IL	Commonwealth Edison

that this transient or worse will not occur in the next fuel cycle."[11] Unfortunately, two weeks earlier, while Randall was out of town, the NRC staff approved the restart of the Yankee Rowe reactor. Randall's ACRS testimony proved prescient as the NRC failed, time and time again, to declare the situation at Rowe unacceptable.

On June 4, 1991, the Union of Concerned Scientists and the New England Coalition on Nuclear Pollution filed a petition with the NRC asserting that Rowe's reactor vessel violated NRC requirements and requesting Rowe's immediate shutdown. Although UCS asked that the Commissioners exercise jurisdiction over the petition the Commission refused and the NRC staff denied the petition on June 25.

On July 11, 1991, UCS renewed and supplemented their petition. The Commission responded on July 31, 1991 by allowing Yankee Atomic to again restart the Rowe reactor while "uncertainties" regarding Rowe's vessel were resolved. This time, however, the Commissioners exercised their jurisdiction over the petition and gave Yankee a deadline, concluding that "[i]n no event will plant operation beyond April 15, 1992 be permitted, until these uncertainties have been resolved. . . ."[12]

On October 1, 1991, over a year after NRC's senior metallurgist had called for its closure, Yankee Atomic "voluntarily" shut down the Yankee Rowe reactor. After months of dispute and with the extent of vessel embrittlement still unresolved, the Yankee Atomic board voted to retire the reactor. On February 27, 1992, Yankee announced the permanent shutdown of the Rowe reactor citing "economic" reasons for its decision. The actual state of the reactor vessel has never been determined.[13]

Monticello & Northern States Power

In light of the events at Yankee Rowe, Northern States Power, owner of Monticello the industry's lead boiling water reactor, decided to delay the reactor's renewal application. The utility then submitted an analysis to NRC citing four reasons for its decision:

1) the uncertain resolution of the high level waste issue; 2) the uncertain resolution of the low level waste issue and rising cost associated with that uncertainty; 3) A need to demonstrate the ability to continue excellent operations while reducing costs; and 4) the regulatory uncertainties of the NRC license renewal process.[14]

The utility went on to note that the recent shutdowns all had three factors in common: the availability of inexpensive replacement power, high operations and maintenance costs and large capital expenditures. Since the cost of replacement power is beyond the control of the nuclear industry, NSP recognized that "[i]t is incumbent upon nuclear power plant owners, then, to control their operations and maintenance costs and capital expenditures such that nuclear power remains competitive with alternative energy supplies."[15] To accomplish this task at Monticello, **NSP concluded that it must keep O & M costs and capital additions significantly below historical trends.**

The Second License Renewal Rule: Has NRC Lost Its Principles?

In the wake of the Yankee Rowe debacle, the NRC reexamined the license renewal rule. In two papers presented to the Commission, the NRC staff attempted to finesse some of the issues that were the cause of the "regulatory uncertainties of the NRC license renewal process." The Staff determined that a new approach to license renewal could be implemented without changing the existing rule.[16]

However, in testimony before the NRC's Advisory Committee on Reactor Safeguards, industry representative William Rasin argued that the staff's new approach was in conflict with the original rule and could lead to a court challenge. Rasin stated that "[i]t was felt it would be very easy to show that the way we applied the rule was not the way the Commission stated, by reading the statement of considerations. Therefore we feel and we continue to express to the Commission, the fact that formal Commission action is necessary."[17]

The NRC eventually came around to the industry's point of view and decided to rewrite the license renewal rule. In promulgating the new license renewal provisions the NRC determined that "[p]ortions of the statements of consideration (SOC) accompanying the existing rule have been viewed to be inconsistent with the NRC staff guidance discussed in SECY-93-049 and SECY-93-113. In addition, **the industry does not believe that the existing rule provides a stable and predictable regulatory process for license renewal.**"[18]

The Commission's changes to license renewal alter the rule's original premise by deleting the concept of age-related degradation unique to license renewal (ARDUTLR). However, the ARDUTLR concept was the linchpin to the original NRC's rule. In fact, it was the only real issue!

> Moreover, consideration of the range of issues relevant only to extended operation has led the Commission to conclude that **there is likely only one real issue generally applicable to all plants—age-related degradation.** The renewal rule focuses the Commission's review on this one safety issue but provides leeway for the Commission to consider, on a case-by-case basis, other issues unique to extended operation.[19]

Absent the ARDUTLR concept, the license renewal rule has no foundation. The recognition that there is "Age Related Degradation Unique To License Renewal" and that it could be managed allowed the NRC to determine that the CLB could be carried into the renewal term without a reduction in safety. Furthermore, the inclusion of ARDUTLR concept limited the scope of the license renewal review. NRC Chairman Ivan Selin stated that:

> if the concept of age related degradation unique to license renewal were thrown out, even though its role there is fairly limited, but the theory, if that were thrown out, then we couldn't rely on the maintenance rule. We would basically have to take all parts of the CLB and at least look at that to say, is there any reason to believe that these will change in the next 20 years compared to the first 40? In other words, a much wider range of issues might have to dealt with at license renewal than is currently conceived.[20]

Without the ARDUTLR concept as a means of narrowing the scope of license renewal review, many utilities may find that the new license renewal process will still require extensive justification for the continued operation of aging reactor beyond 40 years.

The Commission's change of heart and resulting rule change has not been precipitated by any realization about nuclear reactor aging and safety. But, according to James Taylor, NRC Executive Director for Operations:

> because of the language of the rule, in particular the definition of age-related degradation unique to license renewal, the staff's proposed path would have unavoidably entailed a large amount of documentation on effective programs that would be drawn into NRC's regulatory system of formal documentation and change control. It was largely for this reason that the potential renewal applicants and the industry in general thought **the price of going down the staff's path was just too high.**[21]

The NRC's rewrite of the license renewal rule rationalizes the Commission's regulatory retreat by stating that:

> the Commission still believes that mitigation of the deleterious effects of aging resulting from operation beyond the initial license term should be the focus for license renewal. After further consideration and experience in implementing the current rule, the Commission has, however, determined that the requirements for carrying out the license renewal review can and should be simplified and clarified. The Commission has concluded that, for certain plant systems, structures, and components, **the existing regulatory process will continue to mitigate the effects of aging to provide an acceptable level of safety in the period of extended operation.**[22]

Unfortunately, the experience from Rowe does not bear out the Commission's conclusions. Rather, what the Rowe experience demonstrated is that the existing regulatory process may not mitigate the effects of aging and that even the best run nuclear plant may have difficulty proving it can operate safely in the renewal term. Furthermore, the Rowe experience demonstrated that examination of the licensing basis for extended operation could jeopardize the remaining years on the current license.

NRC Gambles On The Maintenance Rule

The Nuclear Regulatory Commission's rewrite of the license renewal rule relies heavily upon the maintenance rule. The Commission's first performance-based rule, the maintenance rule will not go into effect until 1996. The Commission's basis for the license renewal rests upon the as yet unproven record of licensee compliance with the maintenance rule that requires utilities to monitor and adequately maintain the operability of aging reactor systems, structures and components. The importance of this point has not been lost on the Commissioners. During a briefing on changes to the license renewal rule, the Commissioners discussed the relationship between the two rules.

CHAIRMAN SELIN: **I mean basically we're doubling our bet.** When we passed the maintenance rule we said we believe that this rule can be implemented through reg. guidance, inspection guidance to carry out its objectives. And now we're saying, assuming that can be done, one can make a second rule depend on that. But in the case of an individual program, we're going to have experience under the maintenance rule before they renew those.[23]

The Chairman's comments were seized upon by the NRC's Director of Nuclear Reactor Regulation, Thomas Murley. Murley recognized that in doubling their bet on the maintenance rule the Commission had created a double-edged sword.

DOCTOR MURLEY: I think you're right Mr. Chairman. To turn it around, though, let me just mention that, because this is such a cornerstone of our proposed approach, namely prospective reliance on the maintenance rule, we have to ask the question, suppose during a proceeding or during an application review we find problems in our inspection program where maintenance is not being done well? **Then the whole foundation of the rule comes under challenge for that particular application.**[24]

Dr. Murley's comments were not lost on the Chairman. Chairman Selin recognized that reliance on the maintenance rule could open a licensee up to a court challenge.

CHAIRMAN SELIN: I think what would come under challenge would be his application relying on his execution of the maintenance rule, in which his application wouldn't go through until he could satisfy us and, if necessary the courts, that we had been thorough in doing that and that's why it's so desirable that there be a timely renewal process in the rule.[25]

The Commission's concern regarding reliance on the maintenance rule to justify license renewal led Commissioner Remick, a major proponent of performance-based regulation, to address the issue.

COMMISSIONER REMICK: I agree with Tom that there are indications out there that from time to time there's poor maintenance and some equipment loses its functionality and I hope, however, we're maintaining the current licensing basis today which I think we are. With a maintenance rule, the maintenance rule won't be perfect. Hopefully it might improve some of the maintenance problems, but it won't be perfect and with the maintenance rule some equipment will lose functionality even in good programs.

So, I think we have to be careful we aren't thinking of something magical about a maintenance rule that's going to assure these things in the future. It's going to help but it's—and so, I still say that before there was ever something called the maintenance rule, there was maintenance, some good, some bad. I think that has maintained the current licensing basis. It better have or we'd better take action.[26]

Unfortunately, the industry's performance in the area of maintenance mirrors the Commissioner's remarks, "some good, some bad." However, Commissioner Remick is overly optimistic in assessing the impact of poor maintenance on the current licensing basis.

In December 1993, internal nuclear industry documents obtained by Public Citizen revealed marked disparities between what the nuclear industry was telling the Nuclear Regulatory Commission and what the Nuclear Regulatory Commission was then telling the public.

The secret documents were prepared by the Institute of Nuclear Power Operations (INPO), an Atlanta-based industry group established in the wake of the meltdown at Three Mile Island in order to avoid heightened government regulation. Public Citizen's analysis compared the INPO reports with the NRC's Systematic Assessment of License Performance or SALP report, the NRC's report card on nuclear power plant operations. The NRC uses the assessments to determine whether to increase, decrease or provide the same levels of inspection at each of the nuclear reactors.

The Public Citizen report revealed that the NRC consistently failed to address issues raised in all eight areas evaluated by the INPO reports, including maintenance.

INPO assessed each nuclear utility's maintenance program to ensure that maintenance was effectively and efficiently conducted. Furthermore, the INPO evaluations assess whether maintenance activities result in safe and reliable nuclear power plant operation. Of the 56 nuclear power plants covered in the INPO reports leaked to Public Citizen, 40 plants were cited for maintenance problems.

The INPO findings were not all of equal significance, but represented a range of problems. However, at five reactors, deficiencies in the area of maintenance were so severe that they contributed to plant events (see Table 3 below).[27]

The NRC's SALP reports only addressed this problem at two of the five reactors identified by the INPO reports. The NRC, responding to Senate inquiries, has claimed that many of the omissions were actually addressed in the underlying inspection reports.[28] Since the SALP reports are supposed to incorporate the significant information contained in the inspection reports, the public is left to wonder why the negative information in the inspection reports was not included in the SALP.

Although not all maintenance problems rise to this level of significance, they are nonetheless important to reactor safety. In a regulatory analysis accompanying the Commission's maintenance rule, the NRC claimed that safety system failure rates for the best maintained plants were roughly a factor of two to three lower than for those plants that were more poorly maintained.[29]

Table 3

Reactor	Utility	State
Braidwood	Commonwealth Edison	IL
Browns Ferry	Tennessee Valley Authority	AL
Dresden	Commonwealth Edison	IL
Perry	Cleveland Electric	OH
South Texas	Houston Light & Power	TX

II. Can Nuclear Power Plants Operate for 20 More Years When Safety & Economics Threaten Current Operating Licenses

Embrittlement of Reactor Pressure Vessels

As evidenced by the shutdown of Yankee Rowe, the issue of the toughness of the reactor pressure vessel may well determine the operating life of a nuclear reactor.

Reactor Pressure Vessels (RPV) are the steel crucibles that hold the reactor's radioactive core. After years of exposure to the harsh environment, the vessel steel becomes embrittled. Embrittlement is the loss of ductility, i.e., the ability of the pressure vessel metals to withstand stress without cracking. Embrittlement is caused by neutron bombardment and is contingent upon the extent of exposure and the amount of copper and nickel in the metal. The extent of reactor pressure vessel embrittlement is thus dependent upon the unique operating history of each reactor.

The significance of reactor pressure vessel embrittlement is the increased susceptibility to pressurized thermal shock (PTS). Pressurized thermal shock is caused by the rapid cooling and repressurization of the RPV. In establishing requirements for reactor vessels the NRC found that:

> [t]he operating records for pressurized water reactors (PWRs) in the 1970's and early 1980's contained several pressurized thermal shock (PTS) events in which rapid cooldown from operating temperature was followed immediately by repressurization. **The combined thermal and pressure stresses were high enough to induce fracture in a reactor vessel** if it contained a flaw in the beltline and if the event had occurred later in life when the vessel was significantly embrittled by neutron radiation.[30]

If a reactor vessel were to crack, a meltdown of the radioactive fuel rods would be virtually inevitable.

The NRC regulations governing embrittlement of reactor vessels establish two methods for measuring the state of radiation damage to the reactor vessel. One, Charpy upper shelf energy, measures the strength of the vessel welds in foot/lbs. NRC regulations require that strength of the vessel be above 75 foot/lbs prior to licensing and remain above 50 foot/lbs for the life of the reactor.[31] The second, RT-NDT (reference temperature for nil ductility transition), measures the change in the mechanical properties of the metal from ductile to brittle.[32] In other words, it measures the shift in vessel strength as it is bombarded by neutrons. Since embrittlement is time-dependent, it is possible to chart when a reactor will reach its screening criteria for pressurized thermal shock (see Appendix A, page 31).

The original pressurized thermal shock rule, adopted in July 1985, established a point beyond which reactors could not operate without further safety analysis. This point, known as the screening criteria, was established for every PWR. However, the method for calculating the toughness of the reactor pressure vessel failed to account for the copper and nickel in the vessel.[33]

The NRC later amended the rule to bring it into accord with the current state of knowledge regarding vessel embrittlement as reflected in the NRC's regulatory guide 1.99 revision 2. However, the rule change failed to make a concomitant change in the screening criteria for pressurized thermal shock.

On March 23, 1989, Dr. Pryor N. Randall, who later challenged the NRC's restart of the Yankee Rowe reactor due to embrittlement concerns, presented the NRC's case to the Advisory Committee on Reactor Safeguards (ACRS). Discussions between Dr. Chester Seiss of the ACRS and Dr. Randall are most revealing. During the course of his presentation to the ACRS, Dr. Randall raised questions concerning the screening criteria stating, "shouldn't we also change the screening criterion because the formula used to derive the probabilities in the original rule was the old PTS formula. And now we're going to change the formula."[34]

The discussion between Dr. Seiss and Dr. Randall continued:

> DR. RANDALL: The first reason I don't want to change the screening criterion is that if we do so,we will have to reopen the whole issue of PTS.
>
> DR. SEISS: This is a non-technical reason.
>
> DR. RANDALL: I can't —
>
> DR. SEISS: I mean —
>
> DR. RANDALL: Characterize it any way you like. I can't believe I can sell the idea, you know, that this was a neat mathematical process where we've picked an acceptable probability and we've read off a screening criterion, and how if we calculated a different probability, we're just making a delta change in the screening criteria, and are happy.[35]

While the revised PTS rule recognizes that reactors are becoming embrittled at an accelerated rate, it fails to adjust the screening criteria to ensure that reactors are tested at an earlier date. The resulting incongruity will mean that many reactors will operate longer, with more severely embrittled reactor vessels, prior to testing the strength & ductility of the RPV. When utilities finally do test the reactor vessels they may be in for the same rude awakening that resulted in the shutdown of Yankee Rowe.

The attempt to re-license Yankee Rowe failed due to the inability of the licensee to prove that the reactor pressure vessel was sound. According to the NRC's pressurized thermal shock rule governing RPV embrittlement, Yankee Rowe's vessel should not have reached its screening criterion until 2025. However, as early as 1987, NRC knew that the strength of the Yankee Rowe vessel was cause for concern.

In response to a request from then-Director of Nuclear Reactor Regulation Thomas E. Murley, NRC staff developed a list of reactors where the Charpy upper shelf energy could fall below 50 ft-lbs and thus be in violation of fracture toughness requirements. The NRC staff found that 17 reactors were at risk of violating requirements and, as of January 1, 1986, four reactors actually were below the 50 ft-lb threshold. The information in Table 4 (below) accompanied the NRC memo.

Table 4: Reactor Vessels at/or below 50 FT/LBS

PWR Plant	End of License	Charpy USE at End of License (ft-lb)	Charpy USE on January 1, 1986 (ft-lb)
Point Beach 2	2013	34	39
Point Beach 1	2010	38	43
Turkey Point 3	2007	40	44*
Turkey Point 4	2007	40	44*
Ginna	2006	42	47
Arkansas 1	2008	44	>50**
Rancho Seco	2008	44	>50**
Crystal River	2008	44	>50**
TMI 1	2008	44	>50**
Oconee 1	2013	44	>50**
Oconee 3	2014	44	>50**
Surry 2	2008	46	51
Zion 1	2008	47	>50***
Zion 2	2008	49	>50***
Oconee 2	2013	49	>50**
Surry 1	2008	53	57
Davis Besse	2011	56	>50**
Yankee Rowe	1997	Low Copper Welds	
Byron 1	2024	Low Copper Welds	

*Florida Power & Light Company has provided analyses to demonstrate that the weld metal in these reactor vessels have adequate margin.

**B&W Owners Group has provided analysis to demonstrate that the Charpy USE for these reactor vessel welds will reach 50 ft-lb no earlier than 1997.

***Commonwealth Edison Co. has provided analyses to demonstrate that the Charpy USE for these reactor vessel welds will reach 50 ft-lb no earlier then 1994.[36]

The NRC staff theorized that Yankee Rowe and Byron 1 were not at risk of falling below the 50 ft-lb threshold because they used low copper metals in their welds. According to the NRC, "they are less susceptible to having their Charpy USE reduced by neutron irradiation than the other PWR reactor vessels."[37]

Less than five years later Yankee Rowe would shut down because the utility could not prove to the NRC that the Charpy upper shelf energy was above 50 ft-lbs or that lower values of upper shelf energy would provide margins of safety equivalent to those required by Appendix G.

On the heels of the Rowe shutdown, the NRC realized that it had to get ahead of the curve on the embrittlement issue. The NRC issued a generic letter requiring reactors to address the state of RPV embrittlement. From the information submitted, the NRC developed with a list of reactors at risk of embrittlement. "We're trying to avoid any more surprises like we had with Yankee," said NRC's Director of Nuclear Reactor Regulation.[38]

The NRC's 1993 list identified many of those reactors first mentioned in the 1987 memo. There were, however, several notable additions and omissions from the earlier list. Virginia Power's Surry Units 1 & 2 near Williamsburg, VA, Toledo Edison Company's Davis Besse reactor in Ohio and Commonwealth Edison's Byron nuclear plant in Illinois were not included in the 1993 analysis. The NRC required or would later require analysis of the Tennessee Valley Authority's Watts Bar 1 in Tennessee, Northeast Utilities' Millstone 2 in Connecticut and Carolina Power & Light's H.B. Robinson in South Carolina. In addition to identifying these pressurized water reactors, the NRC acknowledged for the first time that embrittlement of the reactor vessel may also be a problem for boiling water reactors by singling out Niagara Mohawk's Nine Mile Point 1 in New York and Public Service Electric & Gas' Oyster Creek in New Jersey for further analysis.

The NRC's list of reactors susceptible to embrittlement has fluctuated over time as the regulator has allowed the licensees to recalculate the extent of embrittlement or justify how more severely embrittled reactors would provide the same level of safety. In October 1994, the NRC issued a proposed rule altering the fracture toughness requirements for reactor pressure vessel embrittlement. The proposed amendments would clarify the pressurized thermal shock (PTS) requirements, make changes to the Fracture Toughness Requirements and the Reactor Vessel Material Surveillance Program Requirements, and provide new requirements for thermal annealing of a reactor pressure vessel. The proposed rule change basically allows the licensees more "flexibility" in proving that the RPV can meet fracture toughness requirements.

Regardless of how the NRC allows utilities to "pencil whip" their analyses, the vessels of these reactors are not getting any stronger. Each time a utility finds additional operating margin in their embrittlement calculations it is likely to come at the expense of the safety margin. The NRC's proposed revision of fracture toughness requirements states that:

> The modification would permit a licensee to develop plant-specific data. Generally, plant-specific data would result in a reduction in the margin applied to the fracture toughness data, to reflect the reduction in uncertainties due to the availability of plant-specific data. However, this must be evaluated on a case-by-case basis.[40]

The NRC's most recent "final" rule on fracture toughness requirements is not due until September 1995.[41] However, even if the NRC can provide utilities with more "wiggle room" in regard to their embrittlement calculations, operation for the additional 20 years contemplated under a renewed license is unlikely (see table 5, page 12).[42]

The NRC has stated that at this point only Palisades is at risk of violating embrittlement standards during its current operating license. The utility is now confronted with the choice of annealing the vessel, replacing it or shutting down the reactor entirely. No nuclear power plant has ever replaced its pressure vessel. Replacement cost estimates approach $100,000,000 and thus appear prohibitive.

Shutdown or Anneal

On January 5, 1995, Consumers Power Company informed its employees that the Palisades reactor would reach its PTS screening criteria limit by as early as 1996. Consumers Power says it plans to anneal the Palisades vessel by the year 2000.[43] Annealing is the process of heating the vessel to approximately 850 degrees F for a week, which will allow the crystalline structure of the metal to regain its ductility. Cost estimates for vessel annealing are around $10 million. However, the outcome of annealing is uncertain. According to NRC, if the embrittled area is a weld, annealing will be less effective and the rate of re-embrittlement will be increased.[44]

If the vessel can actually last until 2000 and Consumers Power goes ahead with the annealing, Palisades will be the first commercial reactor to go through the process. As was the case with Yankee Rowe, Consumers may find that retirement is a better option than the annealing alternative. The duration of the outage required for annealing the vessel combined with the need to justify continued operation may result in the shutdown of the Palisades reactor prior to the expiration of its license in 2007. If Rowe is any indication, the decision to retire the reactor will be left in the hands of the utility while the regulators vacillate.

Steam Generator Tube Degradation

Pressurized Water Reactors (PWRs) incorporate a two-loop design in which pressurized water in one loop carries heat from the reactor's radioactive core into the steam generator. Steam generators contain thousands of tubes which carry the pressurized super-heated water. The heat from the radioactive loop causes the non-radioactive water on the opposite side of the tubing to flash to steam. This steam then drives the turbine to generate electricity[45] (see Diagram A, page 13).

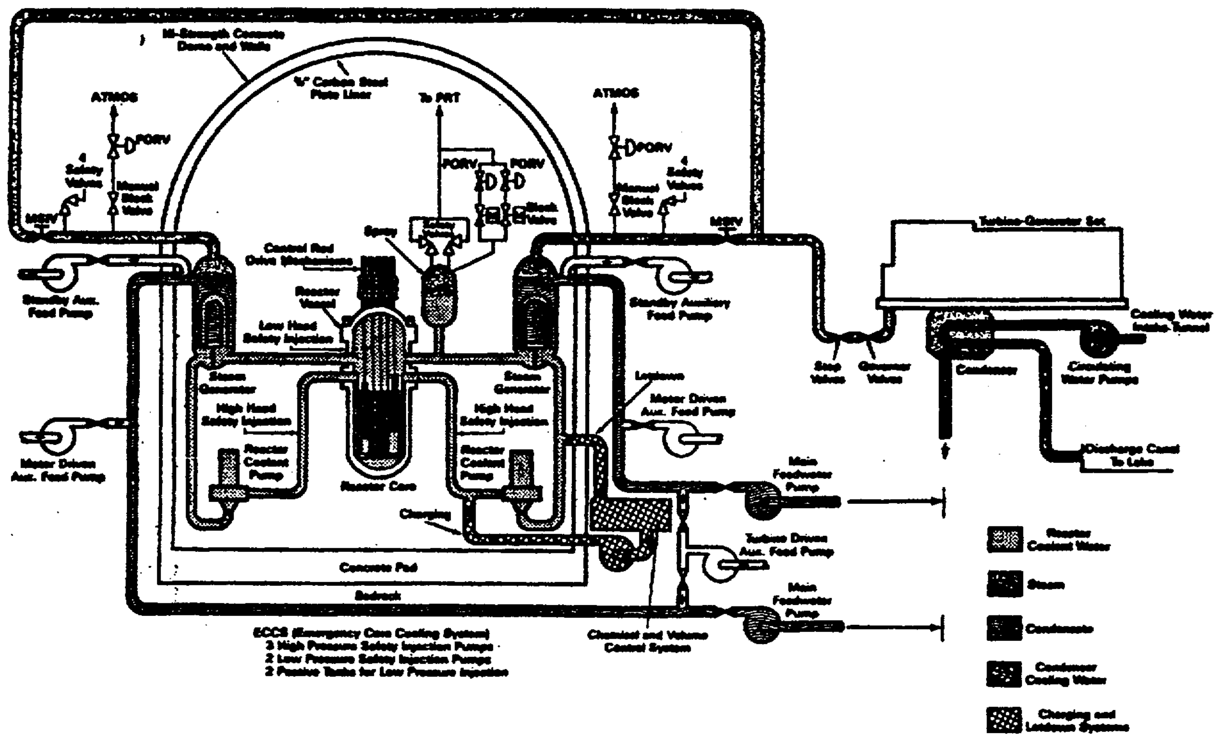
Pressurized water reactors have two or more steam generators depending on the plant design. Westinghouse and Combustion Engineering designed reactors incorporate recirculating steam generators. Operating experience has shown that recirculating steam generators are susceptible to a wide variety of age related degradation.[46] This is primarily due to the fragility of the steam generator tubes made of Inconel or Alloy-600. These tubes, which are typically .03 -.05 of an inch thick, are the boundary between the radioactive and non-radioactive loops of the pressurized water reactor (see Diagram B, page 14).

Steam generator tubes are susceptible to a host of aging problems. The NRC, Westinghouse and the utilities that own Westinghouse and Combustion Engineering nuclear reactors have spent innumerable hours identifying and analyzing the many causes of steam generator tube degradation. Yet it appears that the root cause of the problem is the steam generator tube itself. The Inconel-600 is not a stable alloy when used as tubing for

Table 5: Reactors at Risk of RPV Embrittlement

PTS Date As of 1994/Rule	Reactor	State	Utility	End of License
~1996/1992	Palisades	MI	Consumers Power	2007
2006/2004	Kewaunee	WI	Wisconsin Public Service	2013
>2005/1997	Calvert Cliffs 1	MD	Baltimore Gas & Electric	2014
>2010/2011	Point Beach 1	WI	Wisconsin Electric Power	2010
2011/2011	Zion 1	IL	Commonwealth Edison	2008
>2012/2019	Surry 1	VA	Virginia Power Company	2012
2013/1993	Fort Calhoun	NE	Omaha Public Power	2008
>2013/2008	Point Beach 2	WI	Wisconsin Electric Power	2013
2014/2048	Beaver Valley 1	PA	Duquesne Light Company	2016
2019/2019	Oconee 2	SC	Duke Power Company	2013
2020/2020	Salem 1	NJ	Public Service	2008
2023/2023	Zion 2	IL	Commonwealth Edison	2008
2026/2026	Ginna	NY	Rochester Gas & electric	2006
2034/2008	Diablo Canyon 1	CA	Pacific Gas & Electric	2008
2037/2037	Cook 1	MI	Indiana/Michigan Power	2009
>2050/2050	Farley 1	AL	Southern Nuclear	2007
>2050/2050	St. Lucie 1	FL	Florida Power & Light	2016

Diagram A



Schematic diagram of Westinghouse-designed pressurized water reactor

Source: U.S. Nuclear Regulatory Commission, U.S. NRC Report on the January 25, 1982 Steam Generator Tube Rupture at R.E. Ginna Nuclear Power Plant, *NUREG-0909, April 1982.*

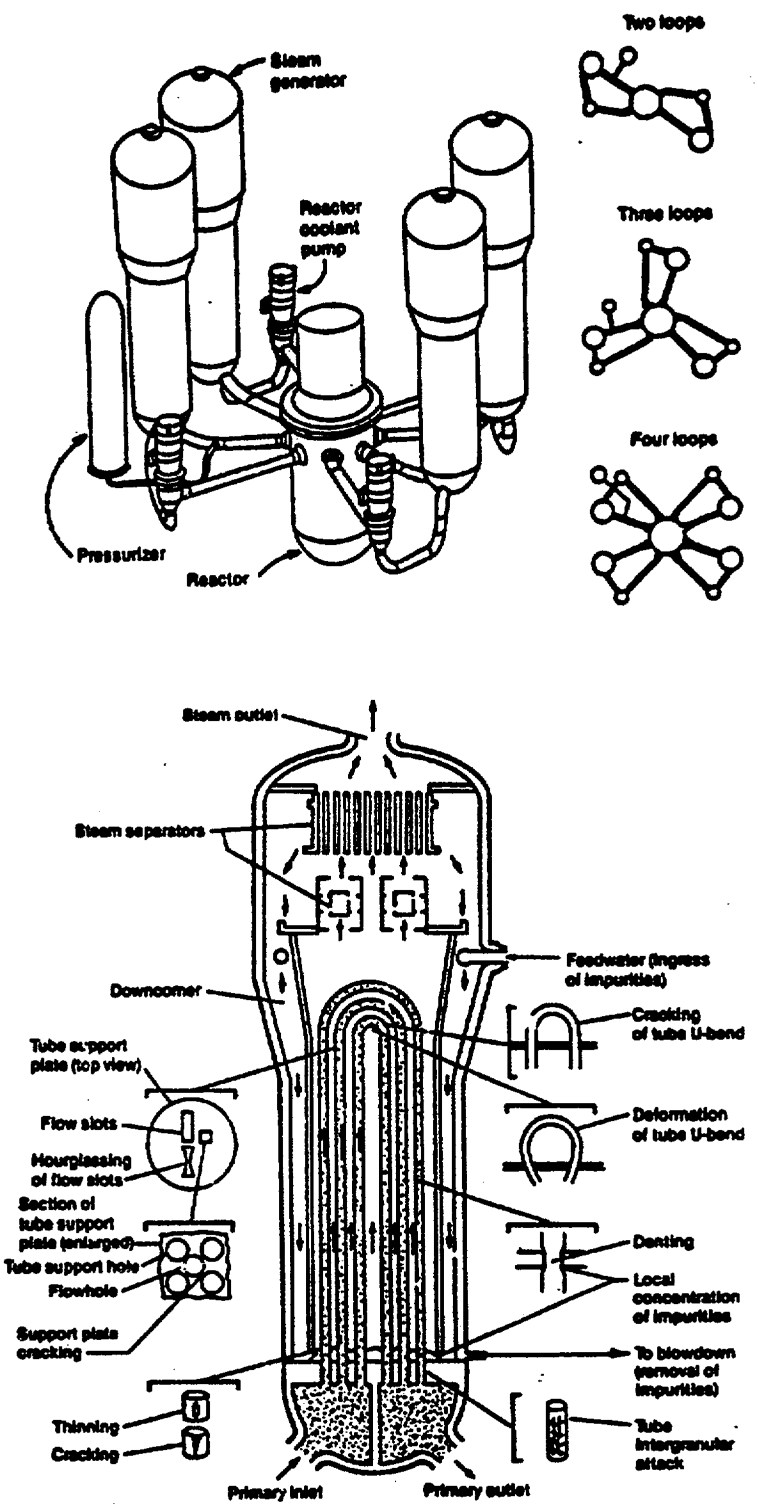
steam generators.[47] Although originally designed to last the life of the plant, steam generators have been replaced at more than ten nuclear power plants since 1981.[48]

When degraded steam generator tubes go undetected, they may break, initiating a potentially disastrous sequence of events. The rupture of a steam generator tube is especially significant because it breaches the barrier between the radioactive and non-radioactive loops of the reactor. Such a breach can allow radioactive coolant to flow into the secondary system at a rate of several hundred gallons per minute. Unless plant operators respond correctly, pressure can increase in the secondary system forcing relief valves to open and releasing radioactive gas into the environment. Such a situation occurred at the Ginna reactor operated by Rochester Gas & Electric near Rochester, NY in 1982. The release of 90 curies of radioactive gas could have been far worse. Had there been any damage to the core of the reactor, the bypass of the containment would have provided highly radioactive fission materials a direct pathway into the environment.[49]

Additionally, as the tubes continue to degrade there is the potential for a multiple tube rupture. Reactors are designed to withstand the disruption caused by the rupture of only a single tube. A multiple tube rupture is said to constitute a "beyond design basis" accident. The rupture of as few as ten steam generator tubes could result in the meltdown of the reactor fuel rods, releasing catastrophic amounts of radiation into the environment. As noted by NRC Commissioner Kenneth Rogers:

> The concern is with sudden multiple tube failures common mode failures. For example, such failures could come about by having essentially uniform degradation of the tubes. Degradation would decrease the safety margins so that, **in essence, we have a 'loaded gun,' an accident waiting to happen.** Under those conditions, a pres-

Diagram B



Source: U.S. Nuclear Regulatory Commision, Residual Life Assessment of Major Light Water Reactor Components, *NUREG/CR-4731, vol. 1, June 1987*

sure transient or a seismic event could rupture many tubes simultaneously. That could allow primary coolant to enter the secondary system and the resulting high pressure to lift the relief valves that are outside containment on the steam line, thus permitting primary water to by-pass containment and communicate with atmosphere directly, resulting in a LOCA (loss of coolant accident) [50]

Unfortunately, the commissioner's words proved prophetic. Precipitating Portland General Electric's (PGE) decision to close the Trojan reactor in Oregon, a member of the NRC staff filed a differing professional opinion (DPO) regarding an NRC decision to allow the nuclear reactor to operate with seriously degraded steam generator tubes. The problem according to the NRC staffer was that "a main steam line break (MSLB) outside containment could trigger a **multiple steam generator tube failure which could then result in a core melt because of depletion of coolant inventory.**"[51]

NRC documents leaked to the Union of Concerned Scientists (UCS) revealed that the risk of a meltdown at the Trojan reactor was 300 times greater than the NRC's Safety Goal standard. As the UCS' letter to NRC Chairman Ivan Selin details:

> [t]he analysis conducted by the Office of Nuclear Reactor Research shows that operation with known flaws in the steam generator tubes can result in an accident in which several steam generator tubes rupture, leading to

the melting of the reactor core and the release of radioactive material directly to the environment out side the reactor containment building. The staff has concluded that the probability of such an accident is over 300 times more likely than your (NRC) safety goal policy permits.[52]

Whether steam generator tubes are sleeved, plugged or replaced altogether, the solution is an expensive one for the utility and ultimately the consumer. While the cost of steam generator tube repair is substantial—repairing microscopic cracks in Trojan's steam generators cost $37 million and kept the reactor shut down for a year—the cost of replacement can prove so prohibitive as to result in the shutdown of the reactor. After fending off numerous voter referenda calling for the shutdown of Trojan, PGE decided to close the nuclear reactor and sue Westinghouse rather than replace the steam generators at a cost of at least $200 million.[53]

However, the NRC allowed at least five other nuclear reactors to operate with similarly degraded steam generators.
These reactors include:

Table 6

Utility Reactor	State
Southern Nuclear Joseph M. Farley Units 1 & 2	AL
Michigan/Indiana Power Donald C. Cook Unit 1	MI
Duke Power Company Catawaba Unit 1	SC
Commonwealth Edison Braidwood Unit 1	IL[54]

In a November 1992 memo, which had until recently been withheld from public disclosure, the NRC's Director of Nuclear Reactor Regulation reported that "**steam generator tube rupture (SGTR) events appear to be unavoidable.**"[55] The memo also points out that NRC regulation is less stringent than other countries. "Regarding steam generator tube inspection programs, it is clear that **the U.S. lags behind the major European countries in terms of scope of inspection. . . . Further, the leak rates allowed were reported to be consistently much lower than that allowed by U.S. Technical Specifications.**"[56]

While the NRC knowingly provides less protection than its European counterparts, the Commission continues to ignore the advice of its own engineers here at home. On July 13, 1994, the same NRC staffer who filed the DPO in the Trojan case again challenged the NRC's lax enforcement of steam generator requirements. In a memorandum to NRC Executive Director for Operations James M. Taylor, the engineer explained that the newly proposed generic letter allowing Westinghouse reactors to operate with degraded stream generator tubes would increase the probability of a core melt accident and the likelihood of a serious release radiation to the environment. The engineer concluded that:

> **The lack of prompt disposition of safety concerns that I have brought to RES management attention has been systematic and pervasive.** For example in 1987 I predicted certain complex SG degradations (11). When RES failed to act I raised the issue with the Commission which promptly issued an inquiry. Several years later the degradation occurred almost exactly as predicted at San Onofre and Maine Yankee. I believe that if prompt action had been taken in 1987 unnecessary risk to plant and costly outages could have been avoided.
>
> **In summary public health and safety can be best protected by replacing the affected steam generator units and not by the institution of easily tunable plugging criteria.**[57]

The NRC engineer went on to conclude that NRC's generic letter would result in reactors violating regulatory limits for potential radiation dose rates and that Commonwealth Edison's Braidwood unit 1 in Illinois already exceeded the requirements. Unfortunately, the NRC has thus far ignored the warnings of its own engineers.

Serious questions were raised again this year as to whether NRC's inspection criteria were sufficient to detect cracking in the steam generator tubes. Maine Yankee was shut down in July 1994 due to steam generator tube cracks that had been present since 1990 but had gone undetected. The Maine Yankee Atomic Power Company claims that, even with the circumferential cracks, the steam generator tubes could have withstood a worst-case-accident. Whether Maine Yankee violated NRC's requirements for steam generator tube integrity remains unclear.[58] What is becoming increasingly clear, however, is that Maine Yankee may soon have to replace its steam generators or retire the reactor.

While many reactors have replaced degraded steam generators, others have decided that replacement is not economically justified. Given NRC's lax enforcement, nuclear reactors will continue to operate with seriously degraded steam generator tubes until the reactor is forced to shut down.

The nuclear reactors in Wisconsin are a good case in point. WEPCO replaced the steam generators at Point Beach Unit 1 during the 1980s and has already ordered replacements for the generators in unit 2. However, WEPCO's proposal to replace the unit 2 steam generators next year at a cost of $120,000,000 has been put on hold by state regulators.[59]

Wisconsin Public Service has decided not to replace the steam generators at Kewaunee. "When we did our steam generator analysis, I think we had a rude awakening," said Clark Steinhardt, senior vice president for nuclear operations. The analysis showed that it would be cheaper to retire Kewaunee in 1998 and replace it with a combined cycle gas facility than it would to replace the steam generators and continue operating the reactor.[60]

Since the life of the steam generator will determine the remaining life of the reactor, the utility is attempting to extend the service life of the steam generator tubes. **Kewaunee is even attempting to unplug tubes that have previously been taken out of service.**[61] Since Kewaunee's license doesn't expire until 2013, the reactor appears to be another candidate for early retirement. Even if the steam generator problems do not shut down the reactor, Kewaunee will face embrittlement concerns prior to license expiration. The only real question remaining is whether the reactor will be shut down by the NRC, by the utility or by an accident.

While steam generator replacement costs have already led to the shutdown of PGE's Trojan reactor, the quick fixes undertaken by utilities to avoid replacement costs are proving problematic. The plugs used to remove cracking steam generator tubes from service are themselves cracking. The risk is that a cracked tube plug could act like a bullet and shoot through the tube bundle resulting in a multiple tube rupture. Westinghouse has recommended that, during the next refueling outage, utilities repair or replace 2,400 plugs in as many as 26 nuclear reactors.[62] Westinghouse acknowledged that there may not be any plugs that have life estimates later than 1994. Westinghouse and NRC have refused to identify those reactors that are at risk, yet only half of the 26 reactors have outages scheduled prior to June 1995.[63] The Commission has abdicated its responsibility on this issue to Westinghouse and in the process obfuscated the truth from public disclosure. The NRC knows which nuclear reactors have the defective tube plugs; they recognize that "in essence, we have a 'loaded gun,' an accident waiting to happen." However, the nuclear bureaucrats refuse to require a timely response to the threat (see Table 7, page 17).

Considering the steep cost of steam generator replacement and the uncertainty of recouping the investment, some utilities may decide not to replace their steam generators and forgo the opportunity to renew their operating licenses. However, the prospect of reactors limping along with degraded steam generators is neither in the interest of the nuclear utilities nor in the interest of public health and safety.

Cracking In Boiling Water Reactors

As nuclear power plants split atoms, the intense radiation and the harsh environs of the reactor core weaken and begin to crack the metal components of the nuclear reactor. The Nuclear Regulatory Commission has long recognized that the reactor vessels—the crucibles that hold the radioactive fuel—can crack precipitating a Chernobyl-like catastrophe. However, the regulators have been slow to acknowledge that radiation-induced damage to other parts of the nuclear reactor could be just as catastrophic. (See Diagram C, page 18.)

More than a dozen General Electric designed nuclear reactors in the U.S. and abroad have evidence of cracking in the reactor core shroud—a metal cylinder surrounding the reactor fuel rods. The owners of these boiling water reactors have contended that the core shroud is of little safety significance. However, the NRC has acknowledged that cracking of the core shroud could damage the radioactive fuel rods, prohibit the insertion of control rods and lead to a meltdown of the reactor.[65]

If this scenario were not frightening enough, another design flaw in the General Electric reactors virtually ensures that the radiation from a meltdown would be released directly into the environment and surrounding communities. As early as 1971, government regulators knew that the public's last line of defense against radiation, the containment, was worthless yet licensed the GE reactors anyway. When staff members suggested that this type of containment be banned, the Commission's deputy director for technical review responded that it "**could well be the end of nuclear power. It would throw into question the continued operation of licensed plants, could make unlicensable the GE and Westinghouse ice condenser plants now in review** and would generally create more turmoil than I can think about."[66] Only years later when documents were released to Public Citizen via the Freedom Of Information Act did the public learn that GE designed reactor containments have a 90% probability of failure in the event of a meltdown.[67] The NRC's *Reactor Risk Reference Document*, a report that studied both GE Mark I and Mark III containments, found that "[i]n general, these data indicate that early containment failure (during a severe accident) can not be ruled out with high confidence for any of the plants."[68] In the event of a meltdown, GE-designed reactors leave the public virtually defenseless.

On July 25, 1994 the NRC issued a generic letter regarding the cracking of core shrouds in boiling water reactors.[69]

The NRC acknowledged that cracking had been observed in nine U.S. reactors. The NRC letter required all BWR owners, with the exception of Big Rock Point, which does not have a core shroud, to inspect their reactors for cracking no later than the next refueling outage and to perform a safety analysis supporting continued operation until the inspections were completed.[70]

Since the issuance of the generic letter several

Table 7: Anticipated Steam Generator Replacements

1993	Indian Point 2	NY	Consolidated Edison (**DELAYED**)
1995	North Anna 2	VA	Virginia Power
1996	Ginna	NY	Rochester Gas & Electric
	Point Beach 2	WI	Wisconsin Electric Power
	Zion 1	IL	Commonwealth Edison
	McGuire 1	NC	Duke Power Company
1997	Kewaunee	WI	Wisconsin Public Service Co.
1998	Catawba 1 *	SC	Duke Power Company
	Farley 2	AL	Alabama Power Company
	Braidwood 1	IL	Commonwealth Edison
	Byron 1	IL	Commonwealth Edison
1999	Maine Yankee*	ME	Maine Yankee Atomic Power
	Fort Calhoun*	NE	Omaha Public Power District
2000	Cook 1*	MI	American Electric Power
	St. Lucie 2*	FL	Florida Power & Light
	San Onofre 2 & 3*	CA	Southern California Edison
	Calvert Cliffs 1 & 2*	MD	Baltimore Gas & Electric
	McGuire 2*	NC	Duke Power Company
	~ Shearon Harris	NC	Carolina Power & Light
	> Farley 2	AL	Alabama Power Company
2001	St. Lucie 1*	FL	Florida Power & light
	Beaver Valley 1 & 2*	PA	Duquesne Light Company
	Zion 2*	IL	Commonwealth Edison
2002	Salem 1	NJ	Public Service Electric & Gas
2006	Salem 2	NJ	Public Service Electric & Gas
	Prairie Island 1 & 2	MN	Northern States Power

* denotes approximate date provided by vendor[64]

Diagram C

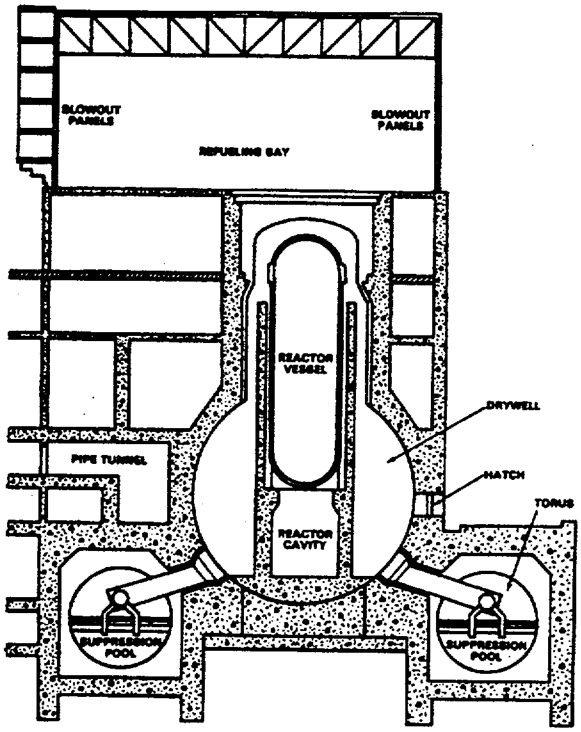
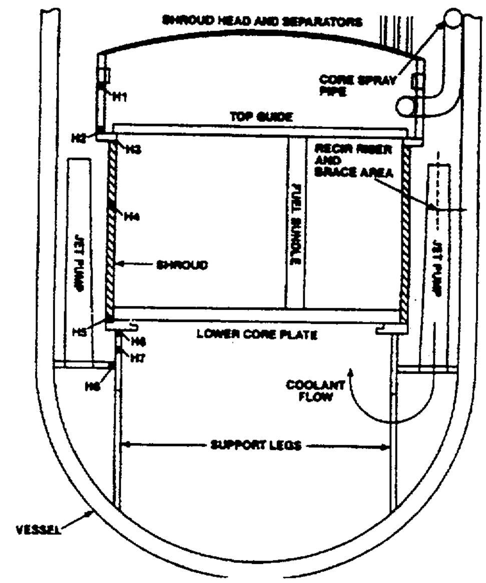


FIGURE 1

CORE SHORUD STRUCTURAL CONFIGURATION



other reactors have turned up cracks in their core shrouds. The following chart lists U.S. BWRs and the year they were licensed. Those reactors which appear in bold have experienced cracking while other reactors are planning preemptive fixes on the core shroud (see Table 8, page 19).[71]

The problem of core shroud cracking is now believed to affect most, if not all, older General Electric reactors. However, "older" is a relative term. Cracking has been found in reactors that have operated for less than 10 years, only one quarter of reactor's operating license. Replacement of the core shroud will cost millions of dollars and calls into question the economic viability of many of these nuclear reactors.[72]

While U.S. utilities are scrambling to repair the reactor core shrouds, German and Swiss owners of boiling water reactors are looking at replacement. PreussenElektra, owner of Germany's oldest BWR, Wurgassen, said that replacement of the core shroud would take 18 months and could cost as much as $120 million.[73] The utility is only prepared to pay $64 million for the replacements, placing the future of the German BWR in doubt.

Faced with an 18-month shutdown and replacement costs potentially running into the hundreds of millions of dollars, U.S. nuclear utilities are attempting to come up with quick fixes for the core shrouds. General Electric has recommended, and Hatch 1 and Oyster Creek have already installed, modifications designed to prevent the shroud from shifting in the event of an accident. The quick fix is slated to cost between $2 and $3 million and take six to ten weeks to install.[74]

Although the concern so far has been that a lateral shift of the core shroud could damage the fuel rods and prevent insertion of control rods in the event of an accident, it appears that the core shroud is not the only cracking reactor component. In fact, cracking in the core shroud appears to be an indicator that other reactor internals are experiencing similar degradation.[75] The NRC has identified at least 25 BWR internals that are susceptible to degradation. The NRC concluded that "[f]ailures of internals could create conditions that may challenge the integrity of the reactor primary containment system, but they do not affect the effectiveness of primary containment systems. However, aging related failures may require extensive shutdown time for repair work."[76]

While the issue of core shroud cracking has not yet resulted in the permanent shutdown of a U.S. reactor, extended operation of boiling water reactors is anything but certain. It is doubtful whether any reactor could economically justify the two-year down time estimated for core shroud replacement. Even if reactors can operate with hastily repaired core shrouds, the degradation of other reactor internals will pose both safety and economic problems which make license renewal improbable.

Table 8: Boiling Water Reactors

Year Licensed	Nuclear Reactor	State
1969	Dresden 2	IL
	Nine Mile Point 1	NY
	Oyster Creek	**NJ**
1970	**Millstone 1**	**CT**
1971	**Dresden 3**	**IL**
	Monticello	MN
1972	Pilgrim	MA
	Quad Cities 1	**IL**
	Quad Cities 2	IL
1973	Browns Ferry 1	AL
	Peach Bottom 2	PA
	Vermont Yankee	VT
1974	Browns Ferry 2	AL
	Brunswick 2	**NC**
	Cooper	NE
	Duane Arnold	IA
	Hatch 1	GA
	Fitzpatrick	**NY**
	Peach Bottom 3	**PA**
1976	**Browns Ferry 3**	**AL**
	Brunswick 1	**NC**
1978	**Hatch 2**	**GA**
1982	LaSalle 1	IL
	Susquehanna 1	PA
1984	Grand Gulf	MS
	LaSalle 2	IL
	Susquehanna 2	PA
	Washington Nuclear 2	WA
1985	**Fermi 2**	**MI**
	Limerick 1	PA
	River Bend	LA
1986	Hope Creek	NJ
	Perry	OH
1987	Clinton	IL
	Nine Mile Point 2	NY
1989	Limerick 2	PA

The Economics Of Current Operation Threaten License Renewal

The economics of license renewal are problematic at best. Increased competition in the wholesale electricity market is already placing serious economic pressure on nuclear utilities. When Northern States Power (NSP) removed Monticello from NRC's lead plant program, the utility recognized that the recent shutdowns of Yankee Rowe, San Onofre Unit 1 and Trojan "demonstrate the changing economic climate which nuclear operations must adjust to if they are going to remain viable options."[77] NSP concluded that there were three factors that were common to each of the shutdown reactors: 1) Alternative energy costs, 2) Operations and maintenance costs and 3) capital expenditures.[78]

In March 1992, the Shearson Lehman Brothers investment firm brought together three former NRC Commissioners as well as the president of Yankee Atomic Electric Company to discuss the economic viability of the nuclear industry.[79] The conference, spurred on by the shutdowns of Yankee Rowe and San Onofre Unit 1, recognized that the nuclear industry is facing a mid-life crisis. The panel acknowledged that "there are increasing prospects that **a number of additional plants will be permanently shutdown while large unamortized investment remains on their owners' books.**"[80]

Peter Bradford, former NRC Commissioner and then Chairman of the New York State Public Service Commission, highlighted the problem posed by early retirement of reactors.

> As to the unamortized portion of the power plant itself, that which is left in the rate base—that's assuming its prudent—and there's no reason to think that if its been around as long as Yankee Rowe unchallenged that it wouldn't be—that has to be recoverable. Because otherwise the utility doesn't have the right incentive to make the sensible going forward decision. That is if it does the right thing by the

ratepayers in not pursuing relicensing, and thereby risks taking an $80 million disallowance from its rate base because it's not using the plant anymore, then you create a discrepancy between the customer's interests and the shareholder's interests which—its that type of discrepancy—its in a lot of different forms—that did so much to get this industry in trouble in the 70's and 80's. So I have no difficulty in it.[81]

The problem of stranded investment, detailed by Mr. Bradford, clearly illustrates the dilemma faced by many nuclear utilities. While many reactors are non-competitive sources of electricity, the utilities fear that they will be unable to recover their sunk costs if they retire these reactors. The situation is further exacerbated when the reactor in question is a poor performer. Although Mr. Bradford contends that Yankee Rowe was a prudent investment, few if any nuclear reactors can match Rowe's performance history. Thus for many reactors, the prudence of the nuclear investment will likely come into question when state regulators determine the amount of stranded investment, if any, the utility will be allowed to recoup after closing the plant.

While some analysts contend that the government has an obligation to the investor, a number of states will disallow recovery of costs from plants that are not "used and useful." Whether the government has an obligation to the shareholders or to the ratepayer is a matter of debate. However, if utilities can not recoup their investments after a nuclear reactor is retired, they may continue to operate unsafe and uneconomical reactors. Last September, NRC Chairman Ivan Selin acknowledged that economic pressure was providing utilities with an "incentive to cut corners."[82] If nuclear reactors are too expensive to operate but utilities are reluctant to close them down due to stranded investments, we could have an economic recipe for disaster.

In March 1994, *Nucleonics Week* reported on Edison Electric Institute (EEI) studies which concluded that "(o)nly a quarter of U.S. nuclear plants produced power more cheaply than the average replacement power available in their power pools as of January 1993. . . ."[83] Remarkably, the EEI study found that half of U.S. nuclear plants would not be competitive even if utilities were able to reduce Operations and Maintenance (O&M) costs by 20%.[84]

While the EEI study has not been publicly released, the methodology which produced these shocking results can be repeated by comparing each nuclear plants O&M costs per kilowatt hour (KWH) with the average cost of replacement power. The following chart shows those reactors which are under the most competitive pressure by comparing O&M and replacement power costs. Results for each nuclear power plant are included in the appendices.

As Table 9 on the following page illustrates, more than half of the nuclear reactors in the U.S. are more expensive to operate than the cost of replacement power. While the analysis is not dispositive, it illustrates the dire state of nuclear economics. If nuclear reactors can not compete in the current market, the prospects for license renewal would appear dim and fading. Even if nuclear utilities can bring O&M costs under control and reverse historic trends, the combination of cheap replacement power, large capital additions and a growing high-level waste problem will likely doom many renewal efforts.

High-Level Radioactive Waste/ Spent Fuel Storage

Of commercial nuclear energy's many dangers, few promise to vex humanity for as long as high-level nuclear waste. Extending the licenses of operating reactors will only increase the amount of waste with which future generations will have to contend. Neither the nuclear industry nor the government has developed the means to isolate high-level radioactive waste for the duration of its hazardous life. Coping with the wastes that result from any proposed license extension looms as an unknown cost and possible taxpayer liability of the NRC's license renewal rule.

High-level waste is defined as irradiated (or spent) nuclear fuel or the wastes that result from reprocessing such materials. U.S. commercial nuclear power plants have generated and

Table 9: Reactors with Operating and Maintenance Costs Above Replacement Power[85]

	Utility	Reactor	O+M Costs (Mills/KWH /KWH)	Replacement Cost* (Mills	Margin
1	Houston Lighting and Power Company	South Texas-1	79.76	20.7	59.06
1	Houston Lighting and Power Company	South Texas-2	79.76	20.7	59.06
3	Consumers Power Company	Big Rock Point-1	65.99	22.9	43.09
4	Tennessee Valley Authority	Browns Ferry-2	48.64	8.5	40.14
4	Tennessee Valley Authority	Browns Ferry-3	48.64	8.5	40.14
6	Gulf States Utilities Company	River Bend-1	48.38	12	36.38
7	Carolina Power and Light Company	Brunswick-1	52.81	19.6	33.21
7	Carolina Power and Light Company	Brunswick-2	52.81	19.6	33.21
9	Florida Power and Light Company	Turkey Point-3	52.63	25.4	27.23
9	Florida Power and Light Company	Turkey Point-4	52.63	25.4	27.23
11	Cleveland Electric Illuminating Company	Perry-1	36.08	10.2	25.88
12	Omaha Public Power District	Fort Calhoun-1	33.38	10.9	22.48
13	Illinois Power Company	Clinton-1	27.59	9.1	18.49
14	New York Power Authority	Indian Point-3	48.03	33.2	14.83
15	Duquesne Light Company	Beaver Valley-1	26.95	12.4	14.55
15	Duquesne Light Company	Beaver Valley-2	26.95	12.4	14.55
17	GPU Nuclear Corporation	Oyster Creek-1	36.59	22.2	14.39
18	Toledo Edison Company	Davis-Besse-1	24.99	11.7	13.29
19	Tennessee Valley Authority	Sequoyah-1	21.87	8.7	13.17
19	Tennessee Valley Authority	Sequoyah-2	21.87	8.7	13.17
21	Northeast Utilities Service Company	Millstone-1	36.02	22.9	13.12
22	Northeast Utilities Service Company	Millstone-2	36.02	23.3	12.72
23	Indiana Michigan Power Company	Cook-1	23.33	11.8	11.53
23	Indiana Michigan Power Company	Cook-2	23.33	11.8	11.53
25	Iowa Electric Light and Power Company	Duane Arnold	22.67	12.1	10.57
26	Northeast Utilities Service Company	Haddam Neck	30.88	21.8	9.08
27	Detroit Edison Company	Fermi-2	26.91	18	8.91
28	System Energy Resources, Inc.	Grand Gulf-1	20.58	12.7	7.88
29	Nebraska Public Power District	Cooper Station	20.08	12.8	7.28
30	Northern States Power Company	Monticello	20.06	13.8	6.26
31	Washington Public Power Supply System	Wash. Nuclear-2	25.01	18.8	6.21
32	Northeast Utilities Service Company	Millstone-3	28.99	22.8	6.19
33	Philadelphia Electric Company	Peach Bottom-3	27.47	21.3	6.17
34	Philadelphia Electric Company	Peach Bottom-2	27.47	21.4	6.07
35	Arkansas Power and Light Company	Arkansas-1	20.46	15.7	4.76
35	Arkansas Power and Light Company	Arkansas-2	20.46	15.7	4.76

continued on next page

Table 9, continued

	Utility	Reactor	O+M Costs (Mills/KWH)	Replacement Cost* (Mills /KWH)	Margin
37	Carolina Power and Light Company	Robinson-2	23.35	19.7	3.65
38	Boston Edison Company	Pilgrim-1	28.93	25.4	3.53
39	Union Electric Company	Callaway-1	16.31	13	3.31
40	TU Electric	Comanche Peak-1	22.91	19.8	3.11
40	TU Electric	Comanche Peak-2	22.91	19.8	3.11
42	Louisiana Power and Light Company	Waterford-3	18.27	15.8	2.47
43	New York Power Authority	Fitzpatrick	34	31.7	2.3
44	Public Service Electric and Gas Company	Salem-1	25.47	23.7	1.77
44	Public Service Electric and Gas Company	Salem-2	25.47	23.7	1.77
46	Commonwealth Edison Company	Dresden-2	29.48	28	1.48
47	Arizona Public Service Company	Palo Verde-1	22.14	20.8	1.34
47	Arizona Public Service Company	Palo Verde-2	22.14	20.8	1.34
47	Arizona Public Service Company	Palo Verde-3	22.14	20.8	1.34
50	Commonwealth Edison Company	Quad Cities-1	25.64	24.4	1.24
51	Northern States Power Company	Prairie Island-2	14.92	13.8	1.12
52	Northern States Power Company	Prairie Island-1	14.92	13.9	1.02
53	Commonwealth Edison Company	Dresden-3	29.48	28.5	0.98
54	Georgia Power Company	Hatch-2	21.79	21	0.79
55	Georgia Power Company	Hatch-1	21.79	21.1	0.69
56	Commonwealth Edison Company	Quad Cities-2	25.64	25	0.64
57	Consumers Power Company	Palisades	23.42	22.9	0.52

*Replacement costs are for the winter of 1993-4

Sources: U.S. Nuclear Regulatory Commission, Replacement Energy costs for Nuclear Electricity—Generating Units in the United States: 1992-1996, NUREG/CR-4012, October 1992, p. 79-190; Inside NRC, June 13, 1994, p. 1-4

discharged over 25,000 metric tons high-level waste in the form of irradiated fuel. The Department of Energy projects that this figure will rise to 84,300 by the year 2030.[86]

The toxicity and longevity of high-level nuclear waste present unique challenges. Irradiated fuel rods contain some of the deadliest substances known to humanity. An individual standing three feet away from unshielded irradiated fuel would receive a lethal dose in 10 seconds.[87] One large nuclear power plant generates annually as much long-lasting radioactivity as a thousand Hiroshima-type atomic bombs.[88] While irradiated fuel constitutes, by volume, a small portion of the over 5 million cubic meters of nuclear waste in the nation, reactor discharges to date contain over 95 percent of the radioactivity.[89]

The government's track record for dealing with the high-level waste problem does not engender confidence. Speaking before the Institute of Nuclear Materials Management, NRC Chairman Ivan Selin acknowledged that:

> The history of spent fuel management in this country has taken several turns, with a final solution still out of reach. Several repository programs have started, stalled and stopped. The latest effort at Yucca

Mountain is proceeding but, at best is years away from the early phases of licensing, much less the actual underground disposal of spent fuel. A monitored retrievable storage (MRS) facility was expected to start accepting commercial spent fuel beginning in 1998, but no such facility is clearly on the horizon. All of these recent developments have changed the circumstances that we face with spent fuel management.[90]

The Chairman noted that both operating and retired nuclear reactors would have to provide additional on-site spent fuel storage for a longer period than originally planned. "But, the dry storage option has triggered an unprecedented amount of local opposition at many sites, further taxing NRC and industry resources."[91]

For obvious reasons, license extensions will exacerbate the challenges of high-level nuclear waste disposal. Current DOE projections, which, as noted above, predict the metric tonnage of irradiated fuel to approach 85,000 by 2030, assume that current reactors will operate to the end of their licenses and that no licenses will be extended or new reactors ordered. When these calculations are amended to assume that half of the nation's operating reactors receive 20-year license extensions, the projection rises to 103,000 metric tons. The amount of anticipated accumulated radioactivity would double, rising from 25,000,000,000 to 53,400,000,000 curies.[92]

Disposal of High-Level Radioactive Waste

Compounding the dangers of high-level radioactive waste is humanity's current inability to guarantee either safe disposal or long-term isolation of highly radioactive materials. The current plan for coping with high-level waste calls for burial in a geologic repository. In 1987, Congress, in a move dictated by political expedience rather than scientific consensus, designated Yucca Mountain in Nevada as the sole candidate for a permanent repository. Site characterization studies have proceeded in the face of vigorous state opposition, Native American claims to the land in question, cost overruns, and significant scientific uncertainty about the Yucca Mountain site in particular, and the notion of geologic disposal in general. As a result, serious doubt exists as to whether a geologic repository, once expected in 1998 but now envisioned by optimistic timetables in 2010, will ever be available.

Many of Yucca Mountain's geologic features cast the site's feasibility as a repository in doubt. For example, one important requirement to ensure waste isolation is keeping water away from containers. One of Yucca Mountain's supposed advantages is slow travel time of the water through the ground. Studies suggest, however, that water may move through the mountain at rates faster than once thought.[93] Conditions of the water table beneath the site may also pose a risk. Over 30 seismic faults cross Yucca Mountain's area. Critics of the repository program fear that an earthquake could raise the water table and flood the repository.[94] Uncertainties about volcanic activity are another problem. A volcano 20 kilometers away from the site appears to have erupted within the last 20,000 years, rather than 270,000 as once thought.[95] When one remembers that the wastes to be deposited will be highly toxic for over 250,000 years, Yucca Mountain's long-term stability becomes a serious concern. Indeed, considering nuclear toxins' longevity, permanent isolation may never be a certainty.

In fact, the feasibility of "disposing" of high-level nuclear waste in an underground storage facility has never been more in doubt. Scientists at Los Alamos National Laboratory in New Mexico fear that high-level radioactive wastes stored in the proposed repository at Yucca Mountain could eventually explode. The scientists fear that plutonium could escape from disposal canisters into the surrounding rock, which possesses physical properties that might aid a spontaneous chain reaction and explosion.[96]

The possibility that radioactive wastes buried below Yucca mountain could detonate in a nuclear blast was first raised last year by Dr. Charles D. Bowman and Dr. Fancesco Venneri. **"We think there is a generic problem with putting fissile materials underground,"** said Dr. Bowman. Since Plutonium-239 has a half life of 24,360 years, significant amounts would still be present long after the metal of the radioactive waste container had disintegrated. Even if the postulated precursors to the explosion do not occur for thousands of years, Plutonium-239 decays into Uranium-235, which contains the same explosive potential as plutonium but takes millions of years to decay. [97] Scientists at the DOE facility at Savannah River have endorsed Bowman's thesis.[98]

Site feasibility studies at Yucca Mountain have already cost over $1.7 billion. Even if scientists eventually disprove Dr. Bowman's thesis, the seriousness of the current dispute so late in the process threaten plans for a repository at Yucca mountain. The Department of Energy projects that a repository, which would still cost at least $15 billion to construct, could open in 2010 if the site is deemed suitable and a license is granted.

The nuclear industry argues that the government has a responsibility to take the high-level radioactive waste it generated at the 109 licensed nuclear reactors in 1998 and that the study of the Yucca mountain site should continue. "We're concerned that this not be used as an excuse by the opponents of waste solutions to stop the scientific analysis of the mountain," said Cathy Roche, vice president of communications for Nuclear Energy Institute (NEI), lobbyists and propagandists for the nuclear industry.[99]

Burying high-level radioactive waste and hoping that Yucca Mountain will not erupt in a nuclear blast hardly seems an appropriate solution to the high-level waste problem. Even nuclear scientists are beginning to understand what environmentalists and public interest advocates have been arguing for decades, that one can not merely "dispose" of radioactive wastes that have a hazardous life of over 240,000 years.

Even if Yucca Mountain eventually opens as a repository, radioactive waste disposition costs will remain high. For one thing, the Nuclear Waste Policy Act calls for a permanent repository to hold no more than 70,000 metric tons of irradiated fuel. As noted above, over 80,000 tons are expected by the year 2030. Demands upon space will also be made by 7,000 tons of defense wastes that are currently earmarked for Yucca Mountain.[100] If the nation intends to continue to pursue geologic storage, more sites will clearly be needed, with all the risks, costs, and citizen opposition that implies.

Finally, serious doubts exist as to whether sufficient funds exist to finance the disposal program without imposing high costs to taxpayers. As part of the Nuclear Waste Policy Act of 1982, Congress created the Nuclear Waste Fund to finance a permanent solution to the high-level waste problem. The Waste Fund's money comes from a tenth of a cent per kilowatt hour fee assessed on commercial nuclear power plants. The fee has not changed since the fund's establishment in 1983. In the intervening 12 years, inflation eroded the fee's buying power by 40 percent.[101] Although the fund has collected over $8 billion, over $4 billion has already been spent.[102] Demands upon the fund, however, are extensive. Characterization of Yucca Mountain alone, once estimated in the hundreds of millions, could cost over $6 billion.[103] In addition to site characterization, the Waste Fund is the intended source of money for construction of the repository and interim storage facility, as well as technical assistance and training funds for communities affected by radioactive waste transport.

The potential for a budget shortfall is compounded by the Department of Energy's management of the Yucca Mountain Program. Critics have faulted the DOE for running overbudget, neglecting important scientific studies, and failing to conduct an honest evaluation of Yucca Mountain's shortcomings. Independent bodies like the General Accounting Office and the Nuclear Waste Technical Review Board have consistently called for an independent review of the Yucca Mountain project and, in the case of the GAO, of the nation's entire nuclear waste policy. "Without a comprehensive independent review of the disposal program and its policies," warns the GAO, "millions—if not billions—of dollars could be wasted."[104]

Interim Storage of High-Level Radioactive Waste

In the meantime, waste continues to accumulate at reactors around the nation. Without a so-called permanent solution for the high-level waste dilemma, interim measures are necessary. One option strongly favored by the nuclear utilities is the establishment of a centralized interim storage facility.

In addition to being expensive, an interim facility would raise risks and costs to the public. If a centralized away-from-reactor storage facility opens, transportation of high-level nuclear waste would proceed on an unprecedented scale. A dump at Yucca Mountain, for example, would require 43 states to bear the risks associated with transport.[105]

Transporting nuclear materials poses significant risks to populations along the route. With an increasing number of shipments, the likelihood of a serious accident involving a cask of irradiated fuel waste increases. Adequate steps have not been taken to ensure cask integrity in the event of such a mishap. Cask safety standards fail to incorporate the full range of trauma to which a container may be exposed in an accident. For example, temperature tolerance standards are lower than the temperatures that a cask might experience in a fire that results from an accident. Furthermore, regulations do not even require that testing be performed on full-scale models to ensure that the containers meet regulatory standards. Even accident-free transport causes radiation exposures along routes, as casks are unable to fully contain radiation.[106]

The political difficulties attendant on transporting radioactive waste mean that any interim facility will likely become a *de facto* repository. Unlike a repository, however, an interim dump will not have been selected with long-term isolation of high-level nuclear wastes in mind.

On-Site Storage of High-Level Radioactive Waste

When Northern States Power dropped out of the license renewal process it concluded that one of the major impediments to renewal was the high level radioactive waste problem. The utility realized that:

> Waste is *the* issue in the political and public arena and must be resolved before the nuclear industry can expect to advance to extended operation of nuclear power plants. Until a satisfactory long term solution to the high and low level waste issues is realized, as perceived by the public, the contention that nuclear power is environmentally benign power source will never be accepted by the public, or state and local governmental and policy leaders. **The basic issue for extended operation is how can a nuclear power plant owner be allowed to continue to generate waste for an additional twenty years if no solution for dealing with the waste generated under its current 40 year license is at hand or at least is moving forward.**[107]

Northern States Power's comments were not mere speculation or rhetoric. Within a year, NSP was facing a challenge to the continued operation of its Prairie Island reactors due to the inability to store additional high level radioactive waste in the reactor's spent fuel pool. While the situation in Minnesota, where a state law prohibited the siting of a high-level waste dump without legislative approval, was unique, the predicament faced by Prairie Island is not. The chart on the opposite page shows those reactors that will lose their ability to operate absent some form of additional storage of high level radioactive waste (see Table 10, page 26).

In the absence of either a repository or interim facility, many utilities are running out of space in their spent fuel pools for discharged assemblies. As many as 14 reactors anticipate this condition by the year 2000.[108]

Another option for interim storage of irradiated fuel is using dry casks at the reactor site. Dry casks offer several advantages to continued reliance upon fuel pools. Casks do not require water and therefore present less of a criticality risk than fuel pools. Furthermore, casks do not rely upon systems that may break down and do not generate wastes by their operation. Finally, use of dry cask on-site storage does not present the transportation risks attendant upon shipping waste to a repository or interim facility.

Dry casks do, however, present risks. Past decisions to resort to usage of the containers

Table 10: Reactor Operability Due to Loss of Irradiated Fuel Storage Capacity

Utility	Reactor	Projected Loss of Ability to Operate	License Expiration Date
Baltimore Gas and Electric Company	Calvert Cliffs-1	1994*	2014
Northern States Power Company	Prairie Island-1	1995*	2013
Northern States Power Company	Prairie Island-2	1995*	2014
Arkansas Power and Light Company	Arkansas-1	1996*	2014
Arkansas Power and Light Company	Arkansas-2	1997*	2018
Wisconsin Electric Power Company	Point Beach-2	1998*	2013
PECO Energy Company	Limerick-2	1998	2029
IES Utilities, Inc.	Duane Arnold	1998	2014
Wisconsin Electric Power Company	Point Beach-1	1998*	2010
PECO Energy Company	Peach Bottom-2	1998	2008
Washington Public Power Supply System	Wash. Nuclear-2	1999	2023
PECO Energy Company	Peach Bottom-3	1999	2008
Maine Yankee Atomic Power Company	Maine Yankee	1999	2008
Consumers Power Company	Big Rock Point-1	1999	2000
Commonwealth Edison Company	Dresden-2	2000	2006
Virginia Power	North Anna-1	2000	2018
PECO Energy Company	Limerick-1	2000	2024
GPU Nuclear Corporation	Oyster Creek-1	2000*	2009
Louisiana Power and Light Company	Waterford-3	2000	2024
Virginia Power	North Anna-2	2001	2020
Carolina Power and Light Company	Brunswick-1	2001	2016
Florida Power and Light Company	St. Lucie-2	2001	2023
Pennsylvania Power and Light Company	Susquehanna-2	2001*	2024
Pennsylvania Power and Light Company	Susquehanna-1	2001*	2022
Commonwealth Edison Company	Dresden-3	2001	2011
Northeast Utilities Service Company	Haddam Neck	2002	2007
Wisconsin Public Service Corporation	Kewaunee	2002	2013
Baltimore Gas and Electric Company	Calvert Cliffs-2	2002*	2016
Nebraska Public Power District	Cooper Station	2002	2014
Public Service Electric and Gas Company	Salem-1	2002	2016
Consolidated Edison Company of New York	Indian Point-2	2003	2013
Georgia Power Company	Hatch-2	2003	2018
Georgia Power Company	Hatch-1	2003	2014
Boston Edison Company	Pilgrim-1	2003	2012
Northeast Utilities Service Company	Millstone-3	2003	2025
Gulf States Utilities Company	River Bend-1	2003	2025
Tennessee Valley Authority	Sequoyah-2	2003	2021
Rochester Gas and Electric Corporation	Ginna	2003	2009

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Table 10, continued

Utility	Reactor	Projected Loss of Ability to Operate	License Expiration Date
New York Power Authority	Fitzpatrick	2003	2014
Carolina Power and Light Company	Brunswick-2	2003	2014
Northeast Utilities Service Company	Millstone-1	2004	2010
Northeast Utilities Service Company	Millstone-2	2004	2015
Northern States Power Company	Monticello	2004	2010
Tennessee Valley Authority	Sequoyah-1	2004	2020
Carolina Power and Light Company	Robinson-2	2004*	2010
Vermont Yankee Nuclear Power Corporation	Vermont Yankee	2004	2012
Pacific Gas and Electric Company	Diablo Canyon-1	2004	2008
Southern California Edison Company	San Onofre-3	2005	2013
Southern California Edison Company	San Onofre-2	2005	2013
Arizona Public Service Company	Palo Verde-2	2005	2025
Niagara Mohawk Power Corporation	Nine Mile Point-1	2005	2009
Arizona Public Service Company	Palo Verde-1	2005	2024
System Energy Resources, Inc.	Grand Gulf-1	2005	2022
Commonwealth Edison Company	Zion-1	2006	2013
Kansas Gas and Electric Company	Wolf Creek-1	2006	2025
Tennessee Valley Authority	Browns Ferry-3	2006	2016
Arizona Public Service Company	Palo Verde-3	2006	2027
Commonwealth Edison Company	Quad Cities-2	2006	2012
Detroit Edison Company	Fermi-2	2006	2025
Commonwealth Edison Company	Zion-2	2006	2013
New York Power Authority	Indian Point-3	2006	2015
Pacific Gas and Electric Company	Diablo Canyon-2	2006	2010
Duke Power Company	McGuire-2	2006	2023
Union Electric Company	Callaway-1	2007	2024
Consumers Power Company	Palisades	2007*	2007
Duke Power Company	McGuire-1	2007	2021
Florida Power and Light Company	St. Lucie-1	2007	2016
Commonwealth Edison Company	Quad Cities-1	2007	2012
Omaha Public Power District	Fort Calhoun-1	2007	2013
Illinois Power Company	Clinton-1	2008	2026
South Carolina Electric and Gas Company	Summer-1	2008	2022
Public Service Electric and Gas Company	Salem-2	2008	2020
Cleveland Electric Illuminating Company	Perry-1	2009	2026
Georgia Power Company	Vogtle-1	2010	2027
Duke Power Company	Oconee-1	2010*	2013
Duke Power Company	Oconee-2	2010*	2013
Florida Power Corporation	Crystal River-3	2010	2016

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Table 10, continued

Utility	Reactor	Projected Loss of Ability to Operate	License Expiration Date
Tennessee Valley Authority	Browns Ferry-2	2010	2014
Alabama Power Company	Farley-1	2010	2017
Georgia Power Company	Vogtle-2	2010	2029
Public Service Electric and Gas Company	Hope Creek-1	2010	2026
Tennessee Valley Authority	Browns Ferry-1	2010	2013
Duquesne Light Company	Beaver Valley-2	2011	2027
Commonwealth Edison Company	Byron-1	2011	2024
Indiana Michigan Power Company	Cook-2	2011	2017
Duke Power Company	Catawba-2	2011	2026
Duke Power Company	Oconee-3	2011*	2014
Indiana Michigan Power Company	Cook-1	2011	2014
Duke Power Company	Catawba-1	2011	2024
Commonwealth Edison Company	Byron-2	2011	2026
Duquesne Light Company	Beaver Valley-1	2012	2016
Commonwealth Edison Company	Braidwood-1	2012	2026
Florida Power and Light Company	Turkey Point-3	2012	2007
Virginia Power	Surry-1	2012*	2012
North Atlantic Energy Service Corporation	Seabrook-1	2012	2026
Commonwealth Edison Company	Braidwood-2	2013	2027
Virginia Power	Surry-2	2013*	2013
Alabama Power Company	Farley-2	2013	2021
Commonwealth Edison Company	LaSalle-1	2013	2022
Florida Power and Light Company	Turkey Point-4	2013	2007
GPU Nuclear Corporation	Three Mile Island-1	2014	2014
Commonwealth Edison Company	LaSalle-2	2015	2023
Toledo Edison Company	Davis-Besse-1	2017*	2017
Niagara Mohawk Power Corporation	Nine Mile Point-2	2017	2026
Carolina Power and Light Company	Shearon Harris-1	2018	2026
TU Electric	Comanche Peak-1	2019	2030
TU Electric	Comanche Peak-2	2021	2033
Houston Lighting and Power Company	South Texas-1	2027	2027
Houston Lighting and Power Company	South Texas-2	2028	2028

* indicates that on-site cask storage is either available or planned

Source: EIA Service Report, Spent Nuclear Fuel Discharges from U.S. Reactors 1993 (Energy Information Administration, February 1995) raised concerns about lack of public participation and failure to extensively test the casks to ensure maximum containment of radiation. At the Palisades reactor in Michigan, for example, the Nuclear Regulatory Commission refused to hold public hearings on a proposed dry cask, having approved the container under a generic licensing procedure. Michigan citizens have found several instances of site-specific issues that can not be addressed generically, along with several examples of the NRC's violating its own rules.[109] The citizens' concerns have already been borne out—flaws have been found in one of the loaded casks at Palisades, and the utility will have to unload and repair the container.[110] The NRC has also cited Northern States Power for contractor violations in the making of casks for the Prairie Island reactors' irradiated fuel.[111]

The daunting challenges presented by high-level radioactive waste will influence utility decisions to renew a reactor's license. Faced with the politically unsavory task of attempting to site additional dry cask storage at the reactor site, utilities with reactors nearing the end their licenses may opt to avoid the political and economic costs and decide instead to retire the nuclear reactor.

Conclusions and Recommendations

No nuclear reactor has yet operated for the 40-year term of its operating license. It appears increasingly unlikely that older reactors will remain competitive with alternative sources of electricity. Given the myriad safety problems facing aging reactors, many nuclear power plants will have difficulty even lasting to the end of their current licenses. When one also considers the utilities' ever growing high-level radioactive waste problem and dismal economics, operation of nuclear reactors beyond 40 years seems like a pipe dream. In fact, the nuclear industry acknowledges that Wall Street will not take license renewal seriously until a licensee actually receives a renewed license.

So why is the NRC, the agency charged with protecting the public health and safety, pursuing license renewal? The answer lies in the woeful economics of the nuclear industry. Absent the ability to amortize large capital additions over an additional 20 year period, nuclear utilities will be forced by economics and safety to retire nuclear reactors prior to license expiration. Early shutdown of nuclear reactors may result in stranding large utility investments and under-funding utility commitments such as decommissioning and waste disposal. Utilities do not want to make their investors swallow these costs. Through license renewal, they can shift the risk of the bad investment in nuclear power from the investor to the ratepayer. While some states have regulatory requirements that plants be "used and useful," many industry analysts believe that utilities will nonetheless be allowed to recoup some or all of their uneconomic investment in nuclear power.

Public Citizen believes that the state utility regulators have a responsibility to ensure that ratepayers are not gouged by the nuclear utilities. The ratepayers already bear both the high costs and the radiological risks of nuclear power. They should not bear the cost of nuclear reactors that are not producing electricity. If utilities continue to operate uneconomical nuclear reactors, they, not the ratepayer, should bear the financial consequences. However, if nuclear reactors are too expensive to operate but utilities refuse to close them down due to stranded investment, we have an economic recipe for a nuclear disaster.

The NRC's attempt to extend the operating licenses of nuclear reactors is little more than a regulatory "slight of hand" which would allow utilities to shift the financial risk of nuclear power from the investor to the ratepayer.

The new license renewal rule has no foundation in safety. Merely relying upon the current regulatory process to protect the public while failing to require that reactors document compliance with the current licensing basis is an abdication of the Commission's responsibility. Absent any enforceable standard for renewal, the NRC's new license renewal rule appears to be little more than a rubber stamp. If the Commission were truly concerned with safety, it would ensure that aging, unsafe and uneconomical reactors are shut down. Rather than extending the operation of nuclear reactors, the NRC should develop objective criteria on which to base a decision to retire a nuclear reactor. Unfortunately, the NRC has never done so.

Table 1 Changes in $\overline{RT}_{PTS}$ at EOL for all PWRs if revision 2 were used

Plant name	Applicable screening criterion, deg. F	RT_{PTS} at end of licensed life, deg. F PTS Rule	Rev. 2	End of licensed life (CP + 40) unless noted)	Estimated year screening criterion will be reached PTS Rule	Rev. 2
Arkansas Nuclear One, Unit 1	300	264	268	2008	2049	2049
Arkansas Nuclear One, Unit 2	270	180	172	2012	>2050	>2050
Beaver Valley Power Station, Unit 1	270	258	257	2010	2032	2048
Byron Station Unit 1	270	123	113	2024 (OL + 40)	>2050	>2050
Callaway Plant	270	161	152	2024 (OL + 40)	>2050	>2050
Calvert Cliffs Nuclear Power Plant Unit No. 1	270	238	301	2014 (OL + 40)	2039	1997
Calvert Cliffs Nuclear Power Plant Unit No. 2	270	199	197	2016 (OL + 40)	>2050	>2050
Catawba Nuclear Station, Unit 1	270	104	87	2024 (OL + 40)	>2050	>2050
Catawba Nuclear Station, Unit 2	270	127	120	2024 (OL + 40)	>2050	>2050
Donald C. Cook Nuclear Plant Unit No. 1	300	251	260	2009	2033	2037
Donald C. Cook Nuclear Plant Unit No. 2	270	205	210	2009	>2050	>2050
Crystal River Unit 3	300	267	257	2008	2029	2039
Davis Besse Unit No. 1	300	217	249	2011	>2050	>2050

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Table 1 (Continued)

Plant name	Applicable screening criterion, deg. F	RT_{PTS} at end of licensed life, deg. F PTS Rule	Rev. 2	End of licensed life (CP + 40) unless noted)	Estimated year screening criterion will be reached PTS Rule	Rev. 2
Diablo Canyon Unit No. 1	270	202	270	2008	>2050	2008
Diablo Canyon Unit No. 2	270	209	211	2010	>2050	>2050
Joseph M. Farley Nuclear Plant Unit No. 1	270	191	186	2012	>2050	>2050
Joseph M. Farley Nuclear Plant Unit No. 2	270	233	233	2012	>2050	>2050
Fort Calhoun	270	235	302	2008	2030	1993
R. E. Ginna Nuclear Power Plant	300	266	283	2006	2032	2026
Haddam Neck Plant	270	165	159	2004	>2050	>2050
Indian Point Unit 2	270	214	226	2006	>2050	>2040
Indian Point Unit 3	270	269	269	2009	2010	2010
Kewaunee Nuclear Power Plant	300	329	313	2013 (OL + 40)	2004	2006
Maine Yankee Atomic Power Plant	300	243	252	2008	>2050	>2050
Millstone 2	300	197	187	2010	>2050	>2050
McGuire Unit 1	270	247	256	2013	2038	2038

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Table 1 (Continued)

Plant name	Applicable screening criterion, deg. F	RT_{PTS} at end of licensed life, deg. F: PTS Rule	Rev. 2	End of licensed life (CP + 40) unless noted)	Estimated year screening criterion will be reached: PTS Rule	Rev. 2
McGuire Unit No. 2	270	194	188	2013	>2050	>2050
North Anna Power Station Unit 1	270	225	221	2011	>2050	>2050
North Anna Power Station Unit 2	270	228	217	2011	>2050	>2050
Oconee Nuclear Station Unit 1	270	239	249	2007	2046	2034
Oconee Nuclear Station Unit 2	300	299	292	2013 (OL + 40)	2014	2019
Oconee Nuclear Station Unit 3	300	233	261	2007	>2050	>2050
Palisades Plant	270	270	322*	2007	2007	1992*
Palo Verde Unit 1	270	142	128	2024 (OL + 40)	>2050	>2050
Point Beach Nuclear Plant Unit 1	270	249	269	2010 (OL + 40)	2029	2011
Point Beach Nuclear Plant Unit 2	300	293	305	2013 (OL + 40)	2018	2008
Prairie Island Unit 1	300	178	181	2008	>2050	>2050

*Depending on the efficacy of the proposed flux reduction program, RT_{PTS} at end of life will be reduced and the Screening Criterion will not be reached before 1992.

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Table 1 (Continued)

Plant name	Applicable screening criterion, deg. F	RT_{PTS} at end of licensed life, deg. F: PTS Rule	RT_{PTS} at end of licensed life, deg. F: Rev. 2	End of licensed life (CP + 40) unless noted)	Estimated year screening criterion will be reached: PTS Rule	Estimated year screening criterion will be reached: Rev. 2
Prairie Island Unit 2	300	222	208	2008	>2050	>2050
Rancho Seco	270	264	255	2008	2012	2018
H. B. Robinson Steam Electric Plant, Unit No. 2	300	269	262	2007	2032	>2050
Salem Generating Station Unit 1	270	255	256	2008	2021	2020
Salem Generating Station Unit 2	270	160	202	2008	>2050	>2050
San Onofre Nuclear Generating Station, Unit 1	270	265	228	2004	2010	>2050
San Onofre Nuclear Generating Station, Unit 2	270	145	137	2013	>2050	>2050
San Onofre Nuclear Generating Station, Unit 3	270	131	124	2013	>2050	>2050
Sequoyah Nuclear Plant Unit 1	270	240	225	2010	>2050	>2050
Sequoyah Nuclear Plant Unit 2	270	172	166	2010	>2050	>2050
St. Lucie Unit 1	270	231	239	2010	2050	>2050
St. Lucie Unit 2	270	179	172	2023 (OL + 40)	>2050	>2050

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Table 1 (Continued)

Plant name	Applicable screening criterion, deg. F	RT_{PTS} at end of licensed life, deg. F PTS Rule	Rev. 2	End of licensed life (CP + 40) unless noted)	Estimated year screening criterion will be reached PTS Rule	Rev. 2
Virgil Summer	270	162	155	2013	>2050	>2050
Surry Power Station Unit 1	270	269	260	2012	2013	2019
Surry Power Station Unit 2	270	225	233	2013	>2050	>2050
Three Mile Island Nuclear Station Unit 1	270	270	262	2008	2008	2011
Trojan Nuclear Plant	270	191	196	2011	>2050	>2050
Turkey Point Unit 3	300	263	283	2007	2035	2020
Turkey Point Unit 4	300	263	283	2007	2035	2020
Waterford 3	270	84	83	2024 (OL + 40)	>2050	>2050
Wolf Creek Generating Station Unit No. 1	270	140	131	2025 (OL+40)	>2050	>2050
Yankee Rowe	270	239	249	1997	2029	2025
Zion Station Unit 1	300	311	299	2008	2005	2011
Zion Station Unit 2	270	259	249	2008	2017	2023

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Table 2 Select list of PWR's, arranged in order of the increase in RT_{PTS} if the formula in revision 2, R.G. 1.99, were put in the PTS rule

	Applicable screening criterion, deg. F	RT_{PTS} at end of licensed life, deg. F		Difference: Rev. 2 minus PTS Rule
		PTS Rule	Rev. 2	
Diablo Canyon Unit No. 1	270	202	270	68
Fort Calhoun	270	235	302	67
Calvert Cliffs Nuclear Power Plant Unit No. 1	270	238	301	63
Palisades Plant	270	270	322	52
Salem Generating Station Unit 2	270	160	202	42
Davis Besse Unit No. 1	300	217	249	32
Oconee Nuclear Station Unit 3	300	233	261	28
Point Beach Nuclear Plant Unit 1	270	249	269	20
Turkey Point Unit 3	300	263	283	20
Turkey Point Unit 4	300	263	283	20
R. E. Ginna Nuclear Power Plant	300	266	283	17
Indian Point Unit 2	270	214	226	12
Point Beach Nuclear Plant Unit 2	300	293	305	12
Oconee Nuclear Station Unit 1	270	239	249	10
Yankee Rowe	270	239	249	10

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Table 3 PWR's that equal, or exceed, or come within one degree of the screening criterion at EOL if revision 2 were used

Plant name	Applicable screening criterion, deg. F	RT_{PTS} at end of licensed life, deg. F, per Rev. 2	Difference	End of licensed life (CP + 40) unless noted)	Estimated year screening criterion will be reached
Palisades Plant	270	322	52	2007	1992*
Fort Calhoun	270	302	32	2008	1998
Calvert Cliffs Nuclear Power Plant Unit No. 1	270	301	31	2014 (OL + 40)	1997
Kewaunee Nuclear Power Plant	300	313	13	2013 (OL + 40)	2006
Point Beach Nuclear Plant Unit 2	300	305	5	2013 (OL + 40)	2008
Diablo Canyon Unit No. 1	270	270	0	2008	2008
Indian Point Unit 3	270	269	-1	2009	2010
Point Beach Nuclear Plant Unit 1	270	269	-1	2010 (OL + 40)	2011
Zion Station Unit 1	300	299	-1	2008	2011

*Depending on the efficacy of the proposed flux reduction program, RT_{PTS} at end of life will be reduced and the Screening Criterion will not be reached before 1992.

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Table 4 PWR's that equal, or exceed, or come within one degree of the screening criterion at EOL if the PTS rule is not changed

Plant name	Applicable screening criterion, deg. F	RT_{PTS} at end of licensed life deg. F, per the PTS rule	Difference	End of licensed life (CP + 40) unless noted)	Estimated year screening criterion will be reached
Kewaunee Nuclear Power Plant	300	329	29	2013 (OL+40)	2004
Zion Station Unit 1	300	311	11	2008	2005
Palisades Plant	270	270	0	2007	2007
Three Mile Island Nuclear Station Unit 1	270	270	0	2008	2008
Indian Point Unit 3	270	269	-1	2009	2010
Oconee Nuclear Station Unit 2	300	299	-1	2013 (OL+40)	2014
Surry Power Station Unit 1	270	269	-1	2012	2013

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Appendix B: Reactor Operating and Replacement Power Costs

Utility	Reactor	O+M Costs (Mills/ KWH)	Replace-ment Cost*	Margin (MIlls/ KWH)
Arkansas Power and Light Company	Arkansas-1	20.46	15.7	4.76
Arkansas Power and Light Company	Arkansas-2	20.46	15.7	4.76
Duquesne Light Company	Beaver Valley-1	26.95	12.4	14.55
Duquesne Light Company	Beaver Valley-2	26.95	12.4	14.55
Consumers Power Company	Big Rock Point-1	65.99	22.9	43.09
Commonwealth Edison Company	Braidwood-1	15.24	28.2	-12.96
Commonwealth Edison Company	Braidwood-2	15.24	28.2	-12.96
Tennessee Valley Authority	Browns Ferry-1	48.64	NA	
Tennessee Valley Authority	Browns Ferry-2	48.64	8.5	40.14
Tennessee Valley Authority	Browns Ferry-3	48.64	8.5	40.14
Carolina Power and Light Company	Brunswick-1	52.81	19.6	33.21
Carolina Power and Light Company	Brunswick-2	52.81	19.6	33.21
Commonwealth Edison Company	Byron-1	14.2	28.4	-14.2
Commonwealth Edison Company	Byron-2	14.2	28.4	-14.2
Union Electric Company	Callaway-1	16.31	13	3.31
Baltimore Gas and Electric Company	Calvert Cliffs-1	21.42	23.7	-2.28
Baltimore Gas and Electric Company	Calvert Cliffs-2	21.42	23.7	-2.28
Duke Power Company	Catawba-1	16.66	19.7	-3.04
Duke Power Company	Catawba-2	16.66	19.7	-3.04
Illinois Power Company	Clinton-1	27.59	9.1	18.49
TU Electric	Comanche Peak-1	22.91	19.8	3.11
TU Electric	Comanche Peak-2	22.91	19.8	3.11
Indiana Michigan Power Company	Cook-1	23.33	11.8	11.53
Indiana Michigan Power Company	Cook-2	23.33	11.8	11.53
Nebraska Public Power District	Cooper Station	20.08	12.8	7.28
Florida Power Corporation	Crystal River-3	24.74	25.3	-0.56
Toledo Edison Company	Davis-Besse-1	24.99	11.7	13.29
Pacific Gas and Electric Company	Diablo Canyon-1	18.69	30.7	-12.01
Pacific Gas and Electric Company	Diablo Canyon-2	18.69	30.7	-12.01
Commonwealth Edison Company	Dresden-2	29.48	28	1.48
Commonwealth Edison Company	Dresden-3	29.48	28.5	0.98
Iowa Electric Light and Power Company	Duane Arnold	22.67	12.1	10.57
Alabama Power Company	Farley-1	18.8	23	-4.2
Alabama Power Company	Farley-2	18.8	23	-4.2
Detroit Edison Company	Fermi-2	26.91	18	8.91
New York Power Authority	Fitzpatrick	34	31.7	2.3
Omaha Public Power District	Fort Calhoun-1	33.38	10.9	22.48
Rochester Gas and Electric Corporation	Ginna	23.32	33	-9.68

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Utility	Reactor	O+M Costs (Mills/ KWH)	Replace-ment Cost*	Margin (MIlls/ KWH)
System Energy Resources, Inc.	Grand Gulf-1	20.58	12.7	7.88
Northeast Utilities Service Company	Haddam Neck	30.88	21.8	9.08
Georgia Power Company	Hatch-1	21.79	21.1	0.69
Georgia Power Company	Hatch-2	21.79	21	0.79
Public Service Electric and Gas Company	Hope Creek-1	19.07	23.9	-4.83
Consolidated Edison Company of New York	Indian Point-2	28.96	31.9	-2.94
New York Power Authority	Indian Point-3	48.03	33.2	14.83
Wisconsin Public Service Corporation	Kewaunee	20.52	22.3	-1.78
Commonwealth Edison Company	LaSalle-1	16.63	27.9	-11.27
Commonwealth Edison Company	LaSalle-2	16.63	27.9	-11.27
Philadelphia Electric Company	Limerick-1	17.93	23	-5.07
Philadelphia Electric Company	Limerick-2	17.93	23	-5.07
Maine Yankee Atomic Power Company	Maine Yankee	15.74	26.7	-10.96
Duke Power Company	McGuire-1	17.25	19.2	-1.95
Duke Power Company	McGuire-2	17.25	19.2	-1.95
Northeast Utilities Service Company	Millstone-1	36.02	22.9	13.12
Northeast Utilities Service Company	Millstone-2	36.02	23.3	12.72
Northeast Utilities Service Company	Millstone-3	28.99	22.8	6.19
Northern States Power Company	Monticello	20.06	13.8	6.26
Niagara Mohawk Power Corporation	Nine Mile Point-1	27.77	29.7	-1.93
Niagara Mohawk Power Corporation	Nine Mile Point-2	27.01	30	-2.99
Virginia Power	North Anna-1	13.77	20.7	-6.93
Virginia Power	North Anna-2	13.77	20.7	-6.93
Duke Power Company	Oconee-1	14.22	19.8	-5.58
Duke Power Company	Oconee-2	14.22	19.8	-5.58
Duke Power Company	Oconee-3	14.22	19.8	-5.58
GPU Nuclear Corporation	Oyster Creek-1	36.59	22.2	14.39
Consumers Power Company	Palisades	23.42	22.9	0.52
Arizona Public Service Company	Palo Verde-1	22.14	20.8	1.34
Arizona Public Service Company	Palo Verde-2	22.14	20.8	1.34
Arizona Public Service Company	Palo Verde-3	22.14	20.8	1.34
Philadelphia Electric Company	Peach Bottom-2	27.47	21.4	6.07
Philadelphia Electric Company	Peach Bottom-3	27.47	21.3	6.17
Cleveland Electric Illuminating Company	Perry-1	36.08	10.2	25.88
Boston Edison Company	Pilgrim-1	28.93	25.4	3.53
Wisconsin Electric Power Company	Point Beach-1	14.28	22.9	-8.62
Wisconsin Electric Power Company	Point Beach-2	14.28	22.9	-8.62
Northern States Power Company	Prairie Island-1	14.92	13.9	1.02
Northern States Power Company	Prairie Island-2	14.92	13.8	1.12

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Appendix B, continued

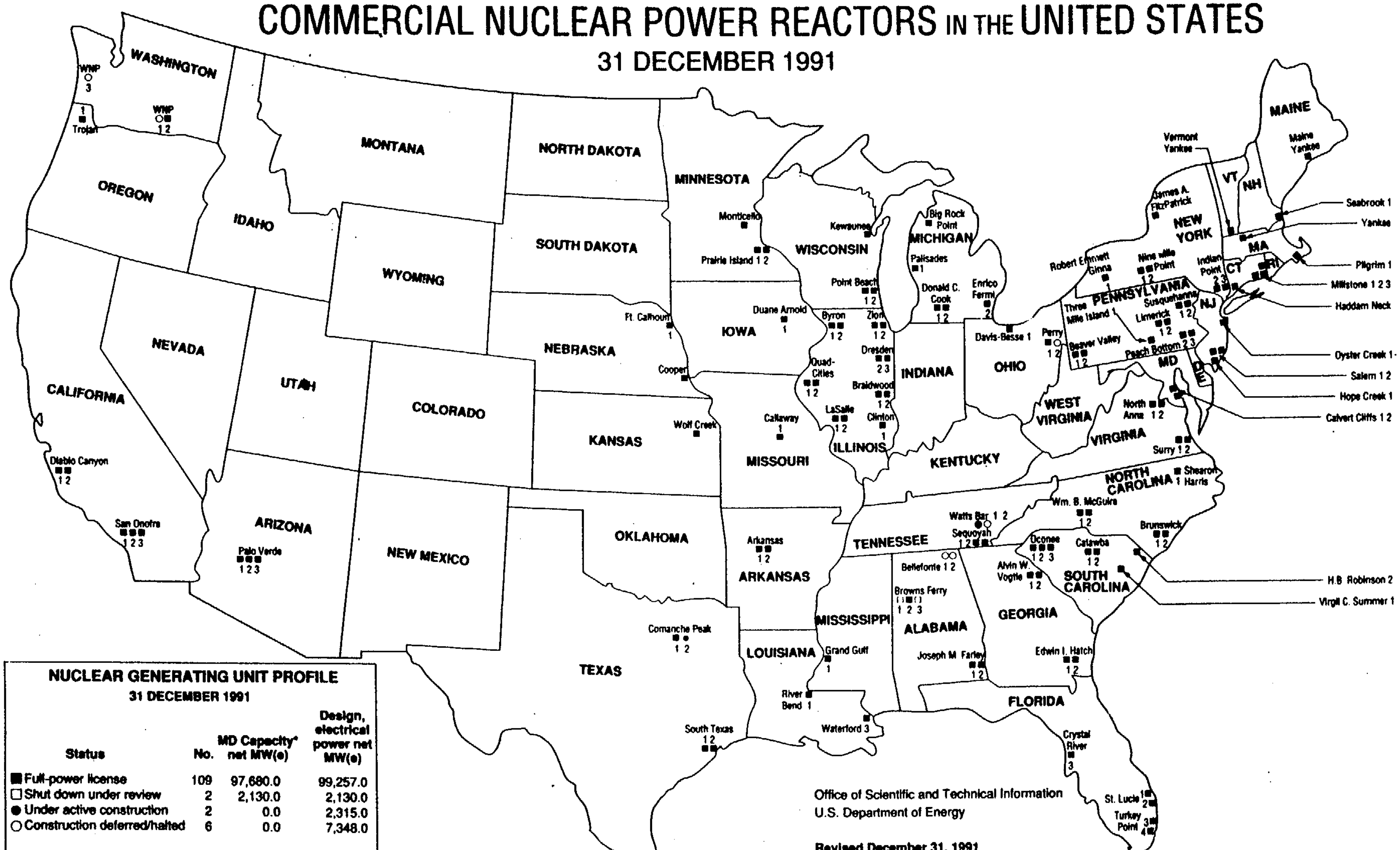
Utility	Reactor	O+M Costs (Mills/ KWH)	Replace-ment Cost*	Margin (MIlls/ KWH)
Commonwealth Edison Company	Quad Cities-1	25.64	24.4	1.24
Commonwealth Edison Company	Quad Cities-2	25.64	25	0.64
Gulf States Utilities Company	River Bend-1	48.38	12	36.38
Carolina Power and Light Company	Robinson-2	23.35	19.7	3.65
Public Service Electric and Gas Company	Salem-1	25.47	23.7	1.77
Public Service Electric and Gas Company	Salem-2	25.47	23.7	1.77
Southern California Edison Company	San Onofre-2	23.35	26.4	-3.05
Southern California Edison Company	San Onofre-3	23.35	26.4	-3.05
North Atlantic Energy Service Corporation	Seabrook-1	21.9	24.2	-2.3
Tennessee Valley Authority	Sequoyah-1	21.87	8.7	13.17
Tennessee Valley Authority	Sequoyah-2	21.87	8.7	13.17
Carolina Power and Light Company	Shearon Harris-1	16.24	20	-3.76
Houston Lighting and Power Company	South Texas-1	79.76	20.7	59.06
Houston Lighting and Power Company	South Texas-2	79.76	20.7	59.06
Florida Power and Light Company	St. Lucie-1	19.65	25.6	-5.95
Florida Power and Light Company	St. Lucie-2	19.65	25.6	-5.95
South Carolina Electric and Gas Company	Summer-1	18.61	18.8	-0.19
Virginia Power	Surry-1	15.76	19.3	-3.54
Virginia Power	Surry-2	15.76	19.3	-3.54
Pennsylvania Power and Light Company	Susquehanna-1	19.64	22.6	-2.96
Pennsylvania Power and Light Company	Susquehanna-2	19.64	22.6	-2.96
GPU Nuclear Corporation	Three Mile Island-1	19.9	23.4	-3.5
Florida Power and Light Company	Turkey Point-3	52.63	25.4	27.23
Florida Power and Light Company	Turkey Point-4	52.63	25.4	27.23
Vermont Yankee Nuclear Power Corporation	Vermont Yankee	24.18	25.8	-1.62
Georgia Power Company	Vogtle-1	14.87	19.7	-4.83
Georgia Power Company	Vogtle-2	14.87	19.7	-4.83
Washington Public Power Supply System	Wash. Nuclear-2	25.01	18.8	6.21
Louisiana Power and Light Company	Waterford-3	18.27	15.8	2.47
Kansas Gas and Electric Company	Wolf Creek-1	15.67	18.2	-2.53
Commonwealth Edison Company	Zion-1	21.43	28	-6.57
Commonwealth Edison Company	Zion-2	21.43	28.1	-6.67

*Replacement costs are for the winter of 1993-4

Sources: U.S. Nuclear Regulatory Commission, Replacement Energy costs for Nuclear Electricity—Generating Units in the United States: 1992-1996, NUREG/CR-4012, October 1992, p. 79-190; Inside NRC, June 13, 1994, p. 1-4

COMMERCIAL NUCLEAR POWER REACTORS IN THE UNITED STATES

31 DECEMBER 1991



NUCLEAR GENERATING UNIT PROFILE

31 DECEMBER 1991

Status	No.	MD Capacity* net MW(e)	Design, electrical power net MW(e)
■ Full-power license	109	97,680.0	99,257.0
□ Shut down under review	2	2,130.0	2,130.0
● Under active construction	2	0.0	2,315.0
○ Construction deferred/halted	6	0.0	7,348.0
Total	119	99,810.0	111,050.0

*Maximum Dependable Capacity or Design Electrical Rating

Because of space limitations, symbols do not reflect precise locations.

DOE/OSTI-8200-R55 (Suppl.)
DE92012402

Office of Scientific and Technical Information
U.S. Department of Energy

Revised December 31, 1991

From *Nuclear Reactors Built, Being Built, or Planned*
(DOE/OSTI-8200-R55)

Notes

1. 56 Fed.Reg. 64943

2. 56 Fed.Reg. 64946

3. U.S. Nuclear Regulatory Commission, "Regulatory Options for Nuclear Power Plant License Renewal, Draft for Comment," NUREG-1317, August 1988, p. x.

4. Advisory Committee on Reactor Safeguards, Subcommittee on Regulatory Policies and Practices: License Renewal, ACRS-T-1789, March 26, 1990, pp. 153, 154.

5. Id. at 165, 166. emphasis added.

6. U.S. Nuclear Regulatory Commission, "Resolution of Deviations Identified During The Systematic Evaluation Program," SECY-92-223, June 19, 1992, p. 7.

7. U.S. Nuclear Regulatory Commission, Memorandum for: James M. Taylor, Executive Director for Operations, From: Samuel J. Chilk, Secretary, Subject: SECY-92-223—Resolution of Deviations Identified During The Systematic Evaluation Program, September 18, 1992, p. 1.

8. Id. at 1.

9. U.S. Congress, Office of Technology Assessment, Aging Nuclear Power Plants: Managing Plant Life and Decommissioning, OTA-E-575, September 1993, p. 14.

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