REPORT TO THE NEW YORK STATE CONSUMER PROTECTION BOARD ON THE COSTS OF THE 1991 REFUELING OUTAGE OF INDIAN POINT UNIT 2

DECEMBER 1991

SCHLISSEL ENGINEERING ASSOCIATES
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I. BACKGROUND

Schlissel Engineering Associates was retained by the New York State Consumer Protection Board to conduct an initial review of the reasonableness of the costs expended by Consolidated Edison Company of New York (Con Ed) during the 1991 Refueling Outage of the Indian Point Unit 2 Nuclear Plant. This Report presents the results of this initial review.

As part of this review, I have examined internal Con Ed reports and analyses concerning Indian Point 2's 1991 Refueling Outage and the steam generator problems experienced by the Unit since 1987. I have also reviewed the relevant correspondence between Con Ed and the U.S. Nuclear Regulatory Commission (NRC) located in the NRC's public docket files. In addition, I have examined materials concerning steam generator problems and replacements at other operating nuclear power plants. Finally, I have discussed the steam generator related repairs performed during the 1991 Refueling Outage with the staff of the New York State Public Service Commission and with Con Ed personnel, including the Indian Point 2 outage manager and the Company's Vice President of Engineering.

II. SUMMARY OF FINDINGS

The main findings of this initial review are as follows:

1. Approximately $34 million of the total $81 million cost of Indian Point 2's 1991 Refueling Outage was spent on steam generator-related work. The remaining outage costs appear to have been spent on typical refueling and routine maintenance activities.

2. The steam generator related work performed during the 1991 Refueling Outage involved measures to prevent the development of cracks in the outer walls (or shells) of each of Indian Point 2's four steam generators.

3. Indian Point 2 had found and repaired steam generator shell cracks during outages in 1987, 1989, and 1990. These repairs had cost approximately $45 million.

4. Although similar cracks have been found in the outer shells of steam generators at other operating nuclear power plants, the cracking at Indian Point 2 was the most severe.

5. Con Ed has concluded that the cracking was preceded by the formation of pits in the steam

generator shells and was propagated by stress corrosion cracking. Company tests have further shown that high levels of dissolved oxygen and copper in the secondary system water contributed to the formation of the pits.

6. Starting in the mid-1970's, Indian Point 2 experienced progressive steam generator tube and plate corrosion. Similar tube corrosion has led other utilities to replace the steam generators in operating nuclear power plants.

7. Con Ed purchased four replacement steam generators in 1985 at a cost of $36 million. The cost of these new steam generators has been included in rate base as an element of the electric M&S accounts.

8. Consequently, Con Ed spent $30 million during the 1990 Mid-Cycle Outage and $34 million during the 1991 Refueling Outage repairing the original Indian Point 2 steam generators at the same time that it was earning a return and recovering other fixed costs associated with the replacement steam generators.

9. Con Ed has stated that the repairs completed

during the 1991 Refueling Outage may allow the Company to delay the installation of the replacement steam generators for a significant number of years.

10. The Company has stated that this delay in the installation of the replacement steam generators would be beneficial in that it would lessen the likelihood that the replacement steam generators would themselves have to be replaced before the end of Indian Point 2's useful operating life.

11. The Company adopted a strategy of making expensive repairs to the original steam generators in 1990 and 1991 in the hope of delaying the installation of the replacement units. This strategy carries the risk that the replacement units will nevertheless have to be installed in the near future as a result of continued steam generator tube and plate corrosion or by the discovery of continued cracking in the outer shells of the steam generators. If this happens, most, if not all of the $64 million spent in 1990 and 1991 on repairing the original steam generators would have been wasted.

12. The Company has provided two economic cost/benefit

studies which it claims show that it is cost beneficial to delay the replacement of the Indian Point 2 steam generators as long as possible.

13. However, the first study, completed in the Spring of 1990, i.e., prior to the 1991 Refueling Outage, showed that the option of making necessary repairs in 1991 and installing the replacement steam generators in 1993 was less expensive than the option of making large repairs in 1991 and installing the replacement steam generators in 1995. The option of making necessary repairs in 1991 and installing the replacement steam generators in 1993 was also only slightly more expensive than the option of making large repairs in 1991 and installing the replacement steam generators in 1997. These results were significant given that the Company's Spring 1990 Analysis was flawed by a number of assumptions which biased the results in favor of delaying the replacement of the steam generators.

14. The results of the Company's Spring 1990 cost/benefit analysis should have led the Company to conduct a more detailed analysis prior to the 1991 Refueling Outage to determine the most economic steam generator repair/replacement

strategy. However, the Company did not perform another economic analysis until after the conclusion of the 1991 Refueling Outage by which time it had invested another $34 million in the repair of the original steam generators.

15. The Company's 1991 Analysis showed that delaying the replacement of the steam generators until 1997 or 2003 was economically beneficial. However, that analysis was also flawed by a number of assumptions which biased the results in favor of delaying the installation of the replacement steam generators.

III. THE HISTORY AND CAUSES OF THE STEAM GENERATOR SHELL CRACKING

The information provided to the CPB by Con Ed on the cost of the 1991 Indian Point 2 Refueling Outage revealed that approximately $34 million, or 40 percent of the total $81 million cost of the outage, was spent on steam generator related repairs. None of the other specific work activities performed during the outage were as expensive as the steam generator work. Moreover, the other work activities performed during the outage appear to be the types of activities typically performed during nuclear power plant refueling/maintenance outages. For these reasons, this initial review by Schlissel Engineering Associates focused on the $34 million of steam generator repairs.

Attachment A to this Report is a drawing of one of Indian Point 2's steam generators. The work performed during the 1991 Refueling Outage was undertaken to eliminate the potential for continued cracking in the area identified in this drawing as the girth weld.

A brief history of the girth area cracking problem is included as Attachment B to this Report. The pages in this attachment were taken from a January 1991 presentation by Con Ed to the NRC staff on the girth weld cracking problem.

As shown in Attachment B, a large number of girth weld cracks were first found during Indian Point 2's 1987 Refueling Outage. The Company repaired all of the cracks found during that outage and restarted the Unit.

Reinspections of the girth weld areas in each of the four steam generators during Indian Point 2's next refueling outage in 1989 revealed 75 new cracks. These cracks were also repaired and the Unit was restarted, following the completion of engineering root cause analyses. Because it considered the cracks to be significant, the NRC mandated that Con Ed conduct a mid-cycle outage prior to Indian Point 2's next scheduled refueling outage. The NRC wanted Con Ed to inspect the steam generators to determine the effectiveness of the repairs performed during the 1989 Refueling Outage.

Inspection of the girth weld areas performed during the subsequent 1990 Mid-Cycle Outage again found cracks in all four steam generators. Although no cracks were found in the areas which had been repaired during the earlier outages, new cracks were found on the unrepaired surfaces of the girth weld. In addition, cracks were found both above and below the girth weld regions. Again the cracks were repaired and the Unit was restarted,

The steam generator inspections and repairs performed during these outages cost the Company $2.9 million in 1987, $11.4 million in 1989, and $29.6 million in 1990, for a total of $44.9 million for the three outages.

Con Ed formed an Independent Review Team, comprised of outside experts, prior to Indian Point 2's 1991 Refueling Outage to develop an action plan for resolving the girth weld problem. The findings and recommendations of this Review Team were implemented during the 1991 Refueling Outage at a cost of approximately $34 million.

Consequently, at least $79 million was been spent by Con Ed on repairing and resolving the steam generator girth weld area cracking problem between 1987 and 1991.

According to NRC Information Notice No. 90-04, a copy of which is included as Attachment C to this Report, cracks in the girth weld area have been found in 18 steam generators at nuclear power plants in the U.S. and at one foreign plant. However, the cracking at Indian Point 2 was the most severe.

The Company has concluded that the girth weld cracking was preceded by the formation of pits on the inside surface of the steam generator shell and was propagated primarily by stress corrosion cracking. The Company has also concluded that the presence of high levels of dissolved oxygen in the secondary system water was the primary contributor to the formation of the pits. The Company further reported to the NRC in June of 1990 that the presence of copper in the secondary system water also contributed to the formation of the pits which preceded the girth weld cracking.

Consequently, the repair program implemented by Con Ed in 1990 and 1991 involved several measures to reduce the levels of dissolved oxygen and copper in the secondary system water. In addition, during the 1991 Refueling Outage Con Ed milled out a six-inch wide by 3/4-inch deep groove at the girth weld around the complete circumference of each steam generator. This groove was filled to the original surface with a low sulfur weld material. The contour of this weld was designed to minimize the residual stresses left in the steam generator shell by the welding process. A post-weld heat treatment was also applied to reduce the residual stresses. Finally, the rewelded area was given a smooth surface finish to mitigate pitting.

Con Ed believes that these actions plus certain measures due to be implemented during Indian Point 2's 1993 Refueling Outage will resolve the girth weld cracking issue. However, past history suggests that further unpleasant surprises may be experienced in the future. For example, the presence of a large number of girth weld cracks was not anticipated prior to 1987. Similarly, having removed the known cracks in 1987, the Company did not expect to find such significant new cracking during the inspections in 1989 or 1990.

Some of the materials reviewed by Schlissel Engineering Associates suggest that the girth weld cracking may have been avoided if the Company had properly managed the secondary system water chemistry. For example, an NRC Inspection Report for Indian Point 2 from mid-1989, included as Attachment D to this Report, noted that the cracking was related to "previous poor secondary water chemistry (oxygen and copper)." Similarly, the viewgraphs presented by Con Ed at a May 11, 1989 meeting with the NRC to discuss the girth weld cracking issue, included as Attachment E to this Report, showed Indian Point 2's water chemistry was significantly worse that the median water chemistries of other nuclear power plants and the guidelines issued by INPO and EPRI.

Further investigation on this issue, beyond that performed by Schlissel Engineering Associates in this initial review, should examine whether Con Ed has been reasonably timely in implementing measures to reduce the levels of dissolved oxygen and copper in the secondary system water. It should also examine whether the earlier adoption of these measures would have prevented the development of cracks in the steam generator outer shells.

IV. REPAIR VERSUS REPLACEMENT OF THE INDIAN POINT 2 STEAM GENERATORS

Starting in the mid-1970's Indian Point 2 began to experience progressive steam generator tube and plate corrosion. Similar corrosion has led other utilities to replace the steam generators at such operating nuclear power plants as Point Beach, Surry, Turkey Point, and Indian Point Unit 3.

Concern that corrosion would force the Company to install new steam generators at Indian Point 2 led Con Ed to purchase four modified steam generators from Westinghouse in 1985.[1] These replacement steam generators cost $36 million and are stored onsite at Indian Point. According to Con Ed, the cost of these replacement steam generators has been included in rate base as an element of the electric M&S accounts.

Con Ed has also retained Westinghouse to prepare engineering cost and schedule estimates for the replacement of the Indian Point 2 steam generators. In its June 1990 Construction Cost Estimate, Revision 1, included as Attachment F to this Report, Westinghouse estimated that the removal of the original steam generators and the installation of the replacement units would cost approximately $120 million and would require an outage of slightly longer than four months. Based on these estimates, the Company has informed shareholders in its 1990 Annual Report that the replacement of the steam generators could cost an additional $127 million and require an outage of up to six months. Copies of the relevant pages from the Company's 1990 Annual

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1 These replacement steam generators have modified designs and use materials which are more resistant to the forms of tube and plate corrosion experienced at operating nuclear power plants.

Report, and supporting worksheets, are included as Attachment G to this Report.

The Company's purchase of replacement steam generators, and their availability during the 1990 Mid-Cycle Outage and the 1991 Refueling Outage, raises the question of whether Con Ed was reasonable in spending $30 million in 1990 and $34 million in 1991 to repair the original Indian Point 2 steam generators instead of installing the replacement units. Officials at Con Ed have given both engineering and economic reasons for their decision to make expensive repairs to the original steam generators in 1990 and 1991 in order to delay the installation of the replacement units as long as possible.

Indian Point 2 began commercial operations in 1974. The Company's engineering argument concerning repair versus replacement of the steam generators is based on the hope/expectation that the Unit will continue to operate for 60 years until 2034. As Mr. Durkin, Con Ed's Vice President for Engineering, explained, if the installation of the replacement steam generators can be delayed until after the turn of the century there is a good possibility that the replacement steam generators would last until the Unit ends its useful operating life. If the steam generators were replaced in the near future, e.g., 1993, there would be a greater chance that the replacement steam generators might themselves have to be replaced before the end of the Unit's service life.

Mr. Durkin also explained that the steam generator tube and plate corrosion was continuing but at a reduced rate. This gives the Company additional confidence that the installation of the replacement steam generators can be delayed beyond the year 2000 if the cracking problem has been resolved.

However, the Company's decision to make large repairs in 1990 and 1991 and delay the installation of the replacement steam generators as long as possible also carries the risk that either (a) the repairs made in 1990 and 1991 will not resolve the shell cracking problem or (b) that currently unanticipated steam generator tube and plate corrosion will force the Company to replace the original steam generators in the near future. Under these circumstances, most if not all of the $64 million spent on repairs in 1990 and 1991 to repair the original steam generators would have been wasted.[2]

In fact, unpleasant surprises have almost been the norm during the operating history of steam generators in the nuclear industry. Unexpected forms of steam generator tube and plate corrosion have been experienced at operating nuclear power plants. This corrosion has often progressed much faster than expected. For example, the use of a new

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Actually, Con Ed would almost certainly attempt to recover the uncollected balance of this $64 million from ratepayers if the original steam generators are retired before the end of Indian Point 2's currently projected forty year service life.

type of probe for inspecting steam generator tubes recently led to the unexpected discovery of a large number of corroded tubes at the Trojan Nuclear Plant in Oregon. A similar discovery would reduce Indian Point 2's capacity factor and could force Con Ed to install the replacement steam generators significantly earlier than it currently plans.

The Company's has provided two economic cost/benefit analyses which it claims show that it is cost beneficial to delay the replacement of the steam generators as long as possible. The first analysis, a copy of which is included as Attachment H to this Report, was prepared in the Spring of 1990. The second analysis, a copy of which is included as Attachment I to this Report, was prepared after the 1991 Refueling Outage.

As shown on page 1 of Attachment H, the Company's claim that the Spring 1990 Analysis showed that there was a cost benefit to delaying the installation of the replacement steam generators is simply wrong. In fact, the Company's Spring 1990 Analysis showed that the option of making necessary repairs in 1991 and installing the new steam generators in 1993 was less expensive than making large repairs in 1991 and installing the new steam generators in 1995. The present worth cost of the option of making necessary repairs in 1991 and installing the new steam generators in 1993 was only very slightly, i.e., $2 million, more expensive than the cost of making large repairs in 1991 and delaying the installation of the new steam generators until 1997.

Moreover, the Company's Spring 1990 Analysis was flawed by a number of assumptions which biased the results in favor of delay. First, the direct cost assumed in the Spring 1990 Analysis for the removal of the original steam generators and the installation of the replacement units was significantly higher than both the cost estimate prepared by Westinghouse in 1990 and the figure published by the Company in its 1990 Annual Report. For example, the 1990 Annual Report noted that the installation of the replacement steam generators was expected to cost $127 million in 1990 dollars.[3] However, the Spring 1990 Analysis assumed that the replacement would cost $192 million in 1992 and 1993 year dollars. Even allowing for a reasonable escalation rate, and the inclusion of an additional $10 million of decontamination costs, the replacement cost used in the Spring 1990 Analysis was substantially higher than the figure published in the 1990 Annual Report.

In addition, the Spring 1990 Analysis allowed the same $75 million for replacement power costs regardless of the year in which the replacement was performed. This is unrealistic. There would certainly be escalation in replacement power costs after 1993 due to fuel price

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3 The Company's 1989 Annual Report, issued in the Spring of 1990, noted that the replacement of the steam generators was expected to cost $122 million, in 1989 dollars, and would request an outage of approximately six months. These were essentially the same cost and schedule estimates as were published in the Company's 1990 Annual Report.

increases and/or changes in the fuel mix due to load growth.

The Spring 1990 Analysis also used a very low 2.2 percent annual escalation rate for the direct cost of removing the original steam generators and installing the replacement units. By way of contrast, the Westinghouse June 1990 Cost Estimate for the steam generator replacement assumed that prices would escalate, on average, at 5.2 percent per year. The use of the unreasonably low 2.2 percent escalation rate biased the Spring 1990 Analysis in favor of delay by understating what the cost of the replacement would be in current year dollars in 1995 and 1997.

Finally, the Spring 1990 Analysis did not even consider the option of performing the steam generator replacement during the 1991 Refueling Outage. The Westinghouse 1990 Construction Estimate for the Indian Point 2 steam generator replacement, Attachment F to this Report, concluded that a one year lead time was needed to make detailed plans for the replacement and to retain expert contractors and personnel. Thus, the Company may have been able to make the steam generator replacement during the 1991 Refueling Outage if it had expedited pre-outage planning starting in 1990. Had the Company done so, it could have avoided spending the $34 million it ultimately spent during the 1991 Refueling Outage to repair the original steam generators.

Together, these flaws biased the results of the Spring 1990 Analysis in favor of delaying the steam generator replacement. However, as discussed above, the results of the Spring 1990 Analysis nevertheless showed that the option of making limited repairs in 1991 and installing the replacement steam generators during Indian Point 2's 1993 Refueling Outage was less expensive than delaying the replacement until 1995 and was only very slightly more expensive than delaying the replacement until 1997. Thus, this option deserved more analysis prior to the 1991 Refueling Outage. But the Company did not make another cost/benefit analysis until <u>after</u> the 1991 Refueling Outage by which time it had invested another $34 million to repair the original steam generators. This was not reasonable.

The economic analysis performed by the Company after the 1991 Refueling Outage did show that it was cost beneficial to delay the replacement of the steam generators as long as possible. However, that analysis was flawed by a number of assumptions which biased the results in favor of delay. First, the 1991 Analysis did not reflect any escalation for replacement power costs and the costs resulting from additional worker exposure to radiation. In addition, the 1991 Analysis used a very low annual escalation rate for the direct cost of the removal of the original steam generators and the installation of the replacement units for the years after 1995. As discussed above, these are unreasonable assumptions.

Finally, both the Spring 1990 Analysis and the 1991 Analysis assumed that Indian Point 2's annual capacity factors would be the same whether the steam generator replacement was made in 1993, 1995, 1997 or 2003. This is not necessarily a reasonable assumption. In fact, it would probably be more reasonable to expect that the Unit's capacity factor would be higher after the steam generator replacement than before.

V. RECOMMENDATIONS

This initial review has identified several critical questions which must be answered by the Company before it is permitted to recover the remaining costs of the 1991 Refueling Outage from ratepayers:

a. Did the Company's inadequate management of the secondary system water chemistry lead to or contribute to the formation of cracks in the outer shells of all four steam generators at Indian Point 2?

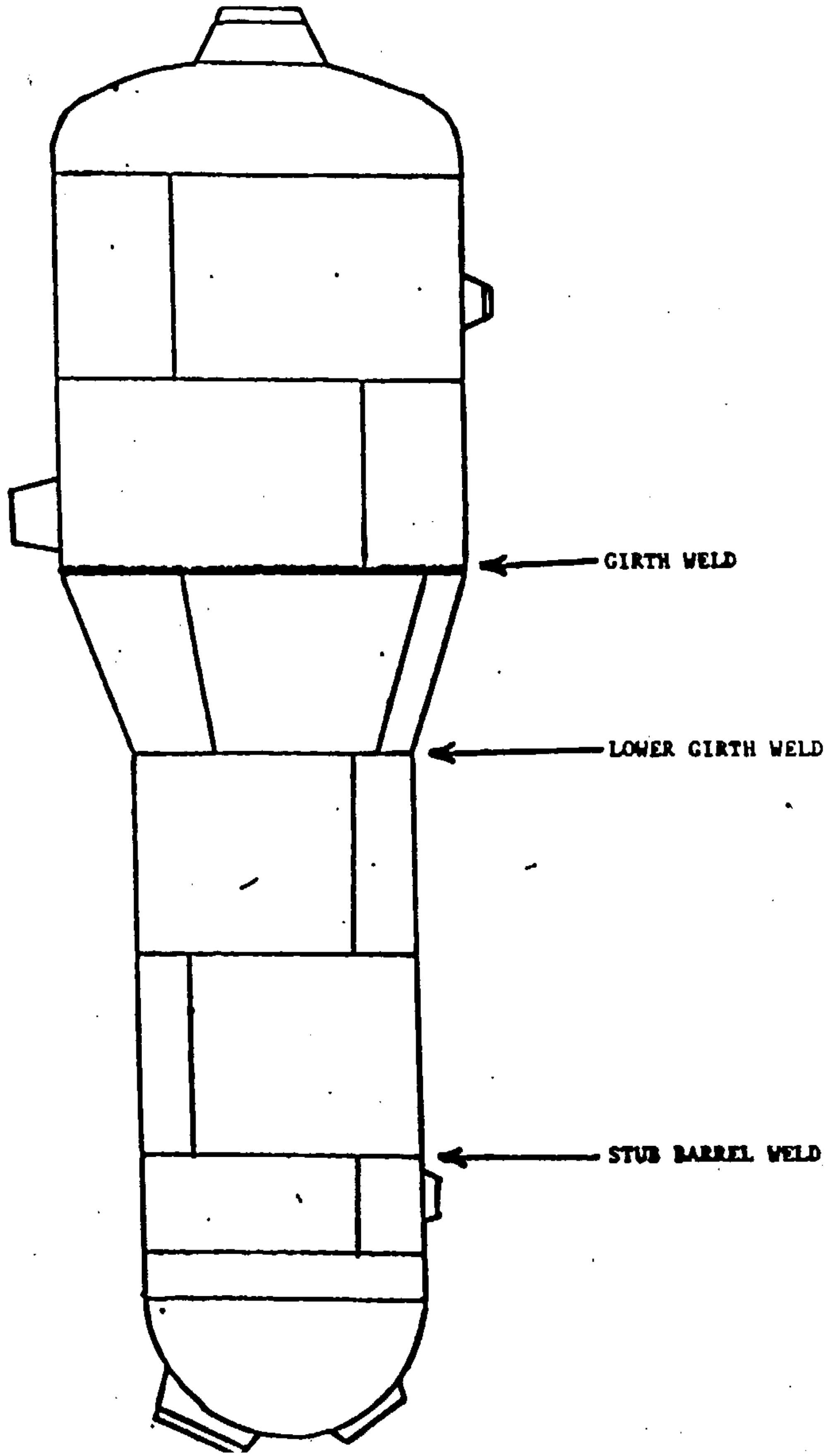
b. Given the repeated discovery of cracks in 1987, 1989, and 1990, why wasn't the Company ready to proceed with the installation of the replacement steam generators during the 1991 Refueling Outage?

c. Was the completion of expensive repairs during the 1991 Refueling Outage more economic that a strategy of performing limited repairs during that outage and scheduling the installation of the replacement steam generators for the Unit's 1993 Refueling Outage?

Moreover, given the Company's $79 million investment in repairing the original Indian Point 2 steam generators since 1987, and its current position that the replacement steam generators may not be installed until substantially after than the year 2000, the undepreciated cost of the replacement steam generators should be removed from plant in service. Instead, the Company should be allowed to accrue AFUDC on these costs until the date when they are installed in Indian Point 2. This regulatory treatment would properly allocate the costs of the replacement steam generators to those future ratepayers who will presumably benefit from their use.

ATTACHMENT A

STEAM GENERATOR SHELL WELDS



ATTACHMENT B

BRIEF HISTORY

1987 INSPECTION

- GIRTH WELD: CRACKS FIRST DETECTED IN 1987

  CIRCUMFERENTIAL CRACKS; UPPER SHELL-TO-TRANSITION CONE ID SURFACE; 4 SGS

  BOAT SAMPLES EXAMINED BY BROOKHAVEN, BECHTEL, LUCIUS PITKIN INC., AND CON EDISON METALLURGICAL LAB

  MECHANISM PREDOMINANTLY CORROSION FATIGUE

  REPAIRED BY GRINDING AND PROPER CONTOURING

  FINDINGS AND REPAIRS SIMILAR TO OTHER PLANTS

BRIEF HISTORY

1989 INSPECTION

- GIRTH WELD: CRACKS IN ALL 4 SGs

  MECHANISM PREDOMINANTLY CORROSION FATIGUE

  ROOT CAUSE: THERMAL SHOCK/THERMAL CYCLING

  HIGHER DISSOLVED OXYGEN AND CHEMISTRY EXCURSIONS CONTRIBUTING FACTORS

  MODIFICATIONS: DOWNCOMER RESISTANCE PLATES REMOVED; TIME DELAY (90 SECS) ON MAIN FEED SHUT OFF

  REPAIRED BY GRINDING; LOCAL WELD BUILDUP IN SG 22 WITH PWHT

BRIEF HISTORY

1989 INSPECTION

- o FW NOZZLES: CRACKS IN NOZZLE BORE INNER RADIUS, NOZZLE BRACKETS FILLET WELDS OF SG 22 AND 23

  MECHANISM CORROSION FATIGUE

  REPAIRED BY GRINDING AND WELDING (HALF BEAD)

  INSPECTION PORTS: LEAKS DURING SERVICE (11/89)

  CORE SAMPLES EXAMINED

  MECHANISM COMBINATION STRESS CORROSION CRACKING AND CORROSION FATIGUE

  REPAIRED BY TWO INCH OPENING, SHELL WELDING (HALF BEAD) AND GASKETED MECHANICAL JOINT

BRIEF HISTORY

1990 MID-CYCLE INSPECTION

- o GIRTH WELD: CRACKS IN ALL 4 SGS, MOSTLY ON UNREPAIRED SURFACES AND GRINDOUT SLOPES; SOME CRACKS WITHIN 1/4 INCH OF PREVIOUS CRACKS

  NO CRACKS ON WELD REPAIRED SURFACE

  MECHANISM PREDOMINANTLY STRESS CORROSION CRACKING

  REDUCED ROLE OF CORROSION FATIGUE

  CRACKS ABOVE AND BELOW THE GIRTH WELD -- ASSOCIATED WITH FABRICATION ATTACHMENT WELDS

  SEVERAL OTHER AREAS OF SGS EXAMINED

  REPAIRED BY GRINDING AND WELDING (HALF BEAD) TO ORIGINAL CONFIGURATION

BRIEF HISTORY

1990 MID-CYCLE INSPECTION

- o FW NOZZLES: CRACKS IN NOZZLE FACE, LOWER SECTION OF NOZZLE BORE, NOZZLE TO PIPE WELD, 4 SGs

  NOZZLE BOAT SAMPLE -- MECHANISM CORROSION FATIGUE

  STRIPING/STRATIFICATION LOADS; LONG TERM EFFECT

- o FEEDRING SUPPORT BRACKETS: CRACKS IN FILLET WELD

  MECHANISM CORROSION FATIGUE

  NO MATERIAL DEGRADATION

  REPAIRED BY GRINDING AND WELDING (HALF BEAD)

GIRTH WELD INDICATION SUMMARY

OUTAGE PERIOD	SG 21	SG 22	SG 23	SG 24	OPERATING PERIOD
1987 Refueling					UP TO 1987
No. of Indications	38	101	102	50	
Maximum Depth	0.56	1.07	1.01	0.57	
Average Depth	0.29	0.49	0.31	0.29	
1989 Refueling					BETWEEN 1987 AND 1989
No. of Indications	5	49	15	7	
Maximum Depth	0.3	0.95	0.33	0.34	
Average Depth	0.21	0.26	0.16	0.14	
1990 Mid-cycle					BETWEEN 1989 AND 1990
No. of Indications	42	23	14	10	
Maximum Depth	0.48	0.51	0.27	0.33	
Average Depth	0.11	0.09	0.08	0.17	

ATTACHMENT C

90-18

UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF NUCLEAR REACTOR REGULATION
WASHINGTON, D.C. 20555

January 26, 1990

NRC INFORMATION NOTICE NO. 90-04: CRACKING OF THE UPPER ~~SHELL-TO-TRANSITION~~ CONE GIRTH WELDS IN STEAM GENERATORS

Addressees:

All holders of operating licenses or construction permits for Westinghouse-designed and Combustion Engineering-designed nuclear power reactors.

Purpose:

This information notice is intended to alert addressees to continuing problems related to cracking of the upper shell-to-transition cone girth welds in the steam generators (SGs) originally described in Information Notices 82-37, "Cracking in the Upper Shell to Transition Cone Girth Weld of a Steam Generator at an Operating Pressurized Water Reactor" and 85-65, "Crack Growth in Steam Generator Girth Welds." It is expected that recipients will review the information for applicability to their facilities and consider actions, as appropriate, to avoid similar problems. However, suggestions contained in this information notice do not constitute NRC requirements; therefore, no specific action or written response is required.

Description of Circumstances:

During the 1989 refueling outage at Zion Unit 1, a scheduled inservice inspection (ISI) was performed on the SG "D" upper shell-to-transition cone girth weld. The ultrasonic testing (UT) detected flaw indications that exceeded the allowable standard of Section XI of the ASME Code, Article IWC-3000 (Table IWB-3511-1). Based upon these results, the extent of UT was initially expanded to include the girth weld in SG "C" and further expanded to include SGs "A" and "B." All surface indications were removed by grinding, contoured to established profiles, and accepted by magnetic particle testing (MT) methods. The deepest repair excavation was approximately 0.50 inch in depth by 6.45 inches in length. Boat samples were removed for metallography. The results of the metallography are still under investigation by the licensee.

During the 1987 refueling outage at Indian Point Unit 2, flaw indications were detected during a scheduled ISI of the same upper shell-to-transition cone girth weld. Visual examination of the inside circumference revealed essentially horizontal intermittent linear indications around the entire weld length of SG #22. Subsequently, UT and MT were extended to essentially 100 percent of this girth weld in all SGs. A total of 291 surface indications were reported in the four SGs, with the most severe cracking occurring in SG #22. The linear indications were predominantly in the vicinity of the weld heat-affected zones.

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A repair program was completed that included progressive grinding to established profiles and nondestructive examination. All observed cracks detected by MT were removed; however, the corrosion pits outside the repair areas were not removed before the plant started up after the refueling outage. The repair resulted in a series of grooves that extended around essentially the entire circumference of SG #22 with the maximum depth of excavation approximately 1.07 inch, whereas the wall thickness is typically 3.5 inches. Eight boat samples were removed for metallurgical analysis. On the basis of this analysis, the licensee concluded that the cracking was most likely caused by corrosion fatigue.

During the 1989 refueling outage at Indian Point Unit 2, an MT was initially conducted on one third of the inside circumference of the SG #22 girth weld. Linear indications were detected during this examination. Subsequently, 100 percent of the inside circumferences of the girth welds in all SGs were inspected. Linear indications were also detected in these additional examinations. All observed cracks were ground out again; the maximum depth of grinding to remove the new flaw indications was 0.95 inch. A weld repair of localized areas and a post-weld heat treatment (PWHT) were accomplished on SG #22. An MT performed after the PWHT detected additional surface indications, which were later removed. The licensee concluded that the probable cause of the cracking was corrosion fatigue resulting from the combined action of thermal cycling, oxygen in the auxiliary feedwater, and copper alloys from the feedwater system. The licensee removed the downcomer flow resistance plate to minimize the thermal cycling mechanism. The licensee also committed to shutdown for an MT inspection during a mid-cycle outage to evaluate the effectiveness of corrective actions.

Discussion:

Cracks and linear indications on the inner circumference have been detected in the upper shell-to-transition cone girth weld in 18 SGs in the United States. In addition, linear indications have been found at one foreign plant. The degree of cracking ranges from severe in the case of Indian Point Unit 2 to isolated and dispersed at Zion Unit 1. At the domestic plants flaws have been observed only in Westinghouse Model 44 and Model 51 vertical recirculating U-tube SGs with the feedwater ring design.

The manufacturer, the affected licensees, and the NRC staff are still evaluating the available information to establish the root cause of the cracking problem and its generic implication. A common factor was the general corrosion pitting on the inside surface of the SGs. Metallography found that the surface pits served as crack initiation sites. The current information indicates that the degradation probably results from corrosion-assisted thermal fatigue. Thermal cycling results from relatively cold water that impinges upon the weld region during reactor trips from full power and certain transient operations. At Indian Point Unit 2, copper alloys from the feedwater system and the downcomer flow resistance plate probably were contributing factors.

The flaw indications can be detected with enhanced UT procedures that are performed by experienced nondestructive examination personnel. The upper shell-to-transition cone weld is located at a gross structural discontinuity. The weld is relatively wide and typically has an irregular crown. These inherent geometric features commonly result in innocuous reflectors. In addition, subsurface flaw indications are known to exist near the inside diameter surface of SGs at several plant sites. In order to distinguish innocuous reflectors from cracks, the following processes may be necessary: scanning at a high gain, the use of multiple transducers with optimum angles, careful plotting of reflector locations, and examination by experienced personnel.

The rules of Section XI of the ASME Code require a volumetric examination of one upper shell-to-transition cone weld during each 10-year inspection interval. The required examinations may be limited to one SG or may be distributed among all the SGs. However, if general corrosion pitting of the SG shell is known to exist, the requirements of Section XI of the ASME Code may not be sufficient to differentiate isolated cracks from inherent geometric conditions. In lieu of volumetric examinations, visual and MT examinations of the interior circumference of the girth weld were used by the licensee of Indian Point Unit 2 to detect the surface-connected flaws.

This information notice requires no specific action or written response. If you have any questions about the information in this notice, please contact one of the technical contacts listed below or the appropriate NRR project manager.

Charles E. Rossi

Charles E. Rossi, Director
Division of Operational Events Assessment
Office of Nuclear Reactor Regulation

Technical Contacts: Martin R. Hum, NRR
(301) 492-0932

Robert A. Hermann, NRR
(301) 492-0911

Attachment: List of Recently Issued NRC Information Notices

ATTACHMENT D

DCS

JUL 1 0 1989

Docket No. 50-247

Consolidated Edison Company of
New York, Inc.
ATTN: Mr. Stephen Bram
Vice President, Nuclear Power
Indian Point Station
Broadway and Bleakley Avenues
Buchanan, New York 10511

Gentlemen:

Subject: Inspection Report No. 50-247/89-14

This letter refers to the routine safety inspection conducted by Mr. H. J. Kaplan of this office on May 15-19, 1989 and June 5-9, 1989 at the Indian Point Station Unit No. 2 in Buchanan, New York. Mr. Kaplan discussed the results of the inspection with you and your staff at the conclusion of the inspection.

This inspection focused on the girth weld cracking and related repairs in steam generator No. 22 as described in the NRC Region I Inspection Report which is enclosed with this letter. Within these areas, the inspection consisted of selective examinations of procedures and representative records, interviews with personnel, and observations by the inspector. Within the scope of this inspection, no violations were observed.

Your cooperation with us in this matter is appreciated.

Sincerely,

Original Signed By:
P. K. Eapen

Jacque P. Durr
Chief, Engineering Branch
Division of Reactor Safety

Enclosure: NRC Region I Inspection Report Number 50-247/89-14

OFFICIAL RECORD COPY IR IP2 89-14 - 0001.0.0
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cc w/encl (w/o SGI):
Jude G. Del Percio, Manager, Regulatory Affairs
Brent L. Brandenburg, Assistant General Counsel
P. Kokolakis, Director, Nuclear Licensing
M. Peckham, Assistant to Resident Manager
A. Budnick, General Manager, Administrative Services
Department of Public Service, State of New York
State of New York, Department of Law
Walter Stein, Secretary - NFSC
Licensing Project Manager, NRR
Public Document Room (PDR)
Local Public Document Room (LPDR)
Nuclear Safety Information Center (NSIC)
NRC Resident Inspector (w/SGI)
State of New York

bcc w/encl (w/o SGI):
Region I Docket Room (with concurrences)
Management Assistant, DRMA (w/o encl)
Section Chief, DRP
B. Clayton, EDO
C. Y. Cheng, NRR

RI:DRS	RI:DRS	RI:DRS
Kaplan/kl/rw	Strosnider	Durr
7/ 7 /89	7/ 7 /89	7/ 10/89

U.S. NUCLEAR REGULATORY COMMISSION
REGION I

Report No. 50-247/89-14

Docket No. 50-247

License No. DPR-26

Licensee: Consolidated Edison Company of New York, Inc.
4 Irving Place
New York, New York 10003

Facility Name: Indian Point Unit 2

Inspection At: Buchanan, New York

Inspection Conducted: May 15-19 and June 5-9, 1989

Inspector: HJ. Kaplan 7-7-89
H. J Kaplan, Senior Reactor Engineer date

Approved by: Jack Strosnider 7/7/89
J. Strosnider, Chief, Materials & Processes date
Section, Engineering Branch, DRS

Inspection Summary: Routine unannounced inspection on May 15-19 and June 5-9, 1989 (Inspection Report No. 50-247/89-14)

Areas Inspected: The inspector focused this inspection on the girth weld crack in steam generator 22. The activities inspected included weld repair, stress relief, post stress relief cracking, grinding repairs and NDE.

Results: It was concluded that the licensee and subcontractors were conducting activities involving the repair in a systematic and controlled manner. Cracking that occurred after stress relief was repaired by grinding, and extensive magnetic particle inspection provided evidence that steam generator 22 was free of ID surface cracks. The girth welds in steam generators 21, 23, and 24, although not weld repaired, were ground to remove indications and magnetic particle inspected to assure the absence of defects.

DETAILS

1.0 Persons Contacted

Consolidated Edison Company

J. Quinn, General Manager, Nuclear Power Gen.
S. Bram, Vice President, Nuclear Power
M. Miele, General Manger, Tech. Serv.
V. Ellwanger, Principal Eng.
Serge Brodske, Sr. Q.A. Eng.
John Curry, Chief, Plant Eng.
George Wasilenko, QA

U.S. Nuclear Regulatory Commission

* P. Kelley, Resident Inspector

* Denotes those present at the exit meeting interview on May 19 and June 9, 1989

The inspector also contacted other licensee employees during the course of the inspection.

2.0 Inspection Scope

This inspection was conducted to review the licensee's activities regarding the repair of cracks in steam generator 22.

2.1 Background

During the present 1989 outage, inservice inspection (ISI) disclosed circumferential cracks in the upper shell to transition girth welds in steam generators (SGs) 21, 22, 23 and 24. Cracking of the girth welds was previously encountered during the 1987 outage. Repair at that time consisted of removal of the defects by grinding, without any weld repair, followed by magnetic particle inspection. With regard to the present condition, the licensee in a meeting on May 11, 1989 at NRR stated that their intention was to weld repair and stress relieve (post weld heat treat) some of the ground out cavities in SG 22, followed by magnetic particle inspection and radiography. SG 22 was by far the worst vessel from the standpoint of depth and frequency of cracking. The cavities in the other SGs were evaluated as acceptable on the basis of stress and fatigue analyses. The cause of the cracking was believed to be due to corrosion-fatigue as indicated by the results of metallographic studies of boat samples removed from SGs 22 and 24. The corrosion aspects were related to previous poor secondary water chemistry (oxygen and copper); the fatigue aspects were related to thermal transients resulting from injection of cold feedwater during startup, plant operation in the hot standby mode, and certain plant transients. The repair was under the technical direction of Westinghouse (W) utilizing, Welding Services, Inc. for welding and Cooper Heat, Inc. for stress relieving.

2.2 Findings

During the inspection of April 15-19, 1989 the inspector reviewed the proposed repair procedure MP-13.84 "Girth Weld Repair of SG 22" and other documents relating to the problem. With minor exceptions the procedure was found to be detailed and technically acceptable. The applicable ASME codes for the repair were identified as Section XI and Section III 1980 Edition and Winter 81 Addenda. The major procedural details for repair welding the SA 302 Mn-Mo alloy steel vessel are noted as follows: (a) Ten (10) cavities ranging in length between 4" and 11", and exceeding 3/4" in depth, were to be weld repaired by Welding Services, Inc. using an automatic Tungsten Inert Gas (TIG) process; (b) preheat was specified as 300°F using electric heaters; (c) an 1125°F - 4 hr. stress relief after welding was specified; and (d) magnetic particle and radiographic inspection was specified before and after stress relief.

The inspector observed the licensee's welder training activities utilizing the automatic welding equipment intended for the production repair. The mockup simulated accessibility and cavity configuration and orientation The inspector noted that the welding head rode on a rail that followed the contour of the ID of the simulated vessel. In production the rail is to be attached to the ID of the vessel by tack welding using the manual TIG process in accordance with Welding Service Procedure MO 313 with 250° F preheat. Removal of the rail is to be accomplished in accordance with a detailed procedure (Appendix B of Procedure WSI-IP 690416) by mechanical means followed by etching to assure removal of heat affected zones and by magnetic particle inspection.

The inspector verified that the WSI automatic welding procedure A03172 for welding the cavities was qualified in accordance with ASME Section IX requirements utilizing the production heat of wire (Ht. 083195). In addition to performing the required bend and tensile tests, drop weight and charpy impact tests also were performed. The Nil Ductility Temperature for WPS A0317 was determined to be -50°F. A review of filler material certification for Ht 083195 indicated conformance to the chemistry requirements of ER-80S-D2 (Mn-Mo) as specified in SFA5.28.

Prior to stress relief, the entire circumference of the girth weld was magnetic particle inspected in accordance with MQS procedure 21.A.202. No indications were detected. The weld repair areas were inspected by radiography in accordance with MQS procedure 20.A.100 with no indications found. On June 6, 1989, the inspector was informed that the ten cavities had been weld repaired followed by radiography and magnetic particle inspection. No indications were disclosed. After stress relief, magnetic particle inspection disclosed sixty (60) crack like, circumferentially oriented indications, two of which were located in the edges of the weld repair areas and one in the original weld metal. The remaining fifty-seven (57) were located in previously ground out cavities which were not weld repaired.

The cause of the new cracks in the ground out areas has not been confirmed with complete certainty. A boat sample with a ¼" long magnetic particle indication was removed for examination by W. Their examination failed to uncover any cracks in the boat sample. W concluded that the indication in the boat sample may have originated from oxide cracking or surface distress marks on the ID surface of the girth weld and postulated that any cracks present in the girth weld existed prior to stress relief as small, tight cracks that opened up because of compressive stresses that occurred during heat treatment. Although W discounted the possibility that cracks were initiated by the heat treatment, the stress concentration effects associated with the ground out cavities, coupled with the lower strength of the weld metal at the higher temperature may have been responsible for the cracking. The two indications in the toes of the weld repair areas are not unexpected because of stress concentration effects that often occur at the edges of welds. It is noted that radiography after stress relief did not disclose any indications.

In order to determine if there were other factors which may have contributed to the girth weld cracking such as improper or faulty stress relief, the inspector thoroughly reviewed the heat treatment records and temperature charts. The review indicated that the 1125°F stress relief was conducted in accordance with Cooperheat procedure 19424-CHP-006 Rev 4 and the requirements of ASME Section III 1980 Ed, Winter Addenda. The specific requirements were: (a) 1125°F ± 25°F for 4 hrs, in a circumferential band around the vessel extending 6" below center line of girth weld to 24" above the center line of the feedwater nozzle; (b) maximum variation of 250° F within any 15 ft. interval of weld length; and (c) maximum cooling rate of 100° F/hr. Heat treatment was accomplished using 138 heaters and 46 thermocouples. The heaters were positioned on the outside of the vessel in an annulus formed by two layers of ceramic fiber insulation. Insulation also was provided contiguous with the inside surface of the vessel. The inspector reviewed W report "Evaluation of the Girth Weld Area Stress Relief Operation for IP-2 Steam Generator" - (Calculation Note No. SM-89-82) which provided the parameters required to preclude excessive thermal gradients. Except for minor deviations as detailed in Cooperheat Deviation Report 001 and reviewed by Westinghouse, the inspector concluded that the stress relief was performed in accordance with the prescribed procedure. As required by procedure 19424-OHP-006, the inspector verified that the production records included the required Westinghouse and Cooperheat QC signatures for various checkpoints in the operation. The new cracks were removed by grinding and the entire circumference reinspected by magnetic particle inspection. It is noted that a mid-cycle magnetic particle inspection of the girth welds in SGs 22 and 23 is planned for the next fuel cycle.

The inspector also reviewed the visual inspection results of the ID inner radius of the feedwater nozzles. Visual examination in lieu of the required Section XI volumetric inspection was granted in a letter from NRR dated March 8, 1988. The inspection revealed linear indications in the inner radius of the lower half of the nozzles in SGs 22 and 23 which were subsequently verified by liquid penetrant inspection. SGs 21 and 24 were found to be free of indications. Corrective action consisted of removing these indications, which required grinding to a depth of .1" - .2", followed by visual inspection and liquid penetrant testing. The licensee provided a stress analysis report indicating that removal of these indications in these areas would not affect the integrity of the subject nozzles.

3.0 Closed Violation 87-20-01: Use of Unauthorized Codes Repairs on Service Water System.

On July 20-24, 1987, NRC determined that the licensee employed two unauthorized repair methods for six leaking joints in the service water system. These methods consisted of depositing a weld overlay over four joints without removing the defects and using clamps on the other two joints. Both of these methods, although considered temporary by the licensee, are not sanctioned by the Code and requires NRR approval. Since that time the licensee has replaced these joints as detailed in MOD packages MFI-87-00826 and CFN-87-50787. In addition, the licensee continued to monitor the service water system by radiography. The inspection plan which was initiated in 1981 includes welds inside and outside the containment. The inspector reviewed radiographic records representing the inspections performed in 1987 and 1989 outages. In 1987, 114 welds were radiographed. Of these, seven were rejectable and subsequently repaired. In the present 1989 outage, 45 welds were radiographed; all were acceptable. By the next outage all welds in the SWS will have been radiographed.

4.0 Management Meeting

Licensee management was informed of the scope and purpose of the inspections at the entrance meeting on May 15, 1989. The findings were discussed with the licensee representatives at the May 19 and June 9, 1989 exit interviews.

At no time during the inspection was written material provided to the licensee by the inspector.

ATTACHMENT E

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

May 17, 1989

P

Docket No. 50-247

LICENSEE: Consolidated Edison Company of New York, Inc.

FACILITY: Indian Point Nuclear Generating Unit No. 2

SUBJECT: MEETING MINUTES REGARDING MAY 11, 1989 MEETING TO DISCUSS INDIAN POINT 2 STEAM GENERATION GIRTH WELD CRACKS, FEEDWATER NOZZLE CRACKS AND REPAIR PROGRAM

A meeting was held in the NRC One White Flint offices in Rockville, Maryland, with Consolidated Edison and NRC staff representatives to discuss cracks which have been detected in the steam generator girth welds and feedwater nozzles and the licensee's proposed repair program. The meeting was requested by the staff. Enclosure 1 is a list of the meeting attendees.

Enclosure 2 is a copy of the meeting agenda. Enclosure 3 shows the weld grind out depths and the crack growth observed during the last operational cycle. Enclosure 4 is a copy of the licensee's briefing charts used to discuss secondary water chemistry during the last operational cycle. In addition to Enclosures 2, 3, and 4, the licensee used a series of briefing charts that were identified as "Proprietary." Those charts were submitted in a letter dated May 11, 1989 to Donald S. Brinkman from W. J. Johnson of Westinghouse. That letter has been docketed.

The staff requested that the licensee make a docketed submittal describing the inspections and repair work that have been performed and a description of the licensee's proposed course of future actions. The staff requested this submittal be submitted for staff review at least one week prior to restart from the current outage and that this submittal include responses to the following questions:

1. The results of inspections of girth welds at other Westinghouse plants. This should include the types of inspection techniques utilized and whether the downcomer flow resistance plates are installed or when removed.

2. Was there a surface inspection of the Indian Point 2 steam generator girth welds following original fabrication stress relief of these welds? If so, what methods were used and what were the findings?

3. Were any weld repairs made after original fabrication stress relief?

4. The remaining wall thickness required for the following criteria:

   a. Crack growth for each steam generator equal to the maximum growth in that steam generator during last fuel cycle,

DFOl
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PDR ADOCK 05000247
P PDC

b. $K_1 = \frac{200}{2}$,

c. Residual weld stress is to be considered,

d. Pressure and thermal stresses from the limiting transient are to be included, and

e. a/l equivalent to size of flaws ground out in last fuel cycle.

The staff also informed the licensee that due to various uncertainties in the evaluation of the girth weld cracks, the staff would recommend to its management that a mid-cycle inspection of the girth welds be required as a condition for restart approval.

Original signed by

Donald S. Brinkman, Senior Project Manager
Project Directorate I-1
Division of Reactor Projects I/II

Enclosures:
As stated

cc: See next page

DISTRIBUTION

Docket file	NRC PDR
Local PDR	J. Sniezek
PDI-1 Rdg	RCapra
DBrinkman	OGC
EJordan	BGrimes
ACRS (10)	H. B. Clayton
CVogan	PSwetland
BElliot	JStrosnider
RHermann	C.Y. Cheng
ALee	ATaboada
MHum	PRandall
HKaplan	DSellers

[IP-2 5/11 MTG SUMMARY]

OFC	:PDI-1	:PDI-1	:PDI-1	:	:	:	:
NAME	:CVogan	:DBrinkman:vr	:RCapra	:	:	:	:
DATE	:5/17/89	:5/17/89	:5/17/89	:	:	:	:

OFFICIAL RECORD COPY

Mr. Stephen B. Bram
Consolidated Edison Company
of New York, Inc.

Indian Point Nuclear Generating
Station 1/2

cc:

Mayor, Village of Buchanan
236 Tate Avenue
Buchanan, New York 10511

Ms. Donna Ross
New York State Energy Office
2 Empire State Plaza
16th Floor
Albany, New York 12223

Mr. Jude Del Percio
Manager of Regulatory Affairs
Consolidated Edison Company
of New York, Inc.
Broadway and Bleakley Avenue
Buchanan, New York 10511

Senior Resident Inspector
U.S. Nuclear Regulatory Commission
Post Office Box 38
Buchanan, New York 10511

Mr. Brent L. Brandenburg
Assistant General Counsel
Consolidated Edison Company
of New York, Inc.
4 Irving Place - 1822
New York, New York 10003

Director, Technical Development
Programs
State of New York Energy Office
Agency Building 2
Empire State Plaza
Albany, New York 12223

Mr. Peter Kokolakis, Director
Nuclear Licensing
Power Authority of the State
of New York
123 Main Street
White Plains, New York 10601

Mr. Walter Stein
Secretary - NFSC
Consolidated Edison Company
of New York, Inc.
4 Irving Place - 1822
New York, New York 10003

Regional Administrator, Region I
U.S. Nuclear Regulatory Commission
475 Allendale Road
King of Prussia, PA 19406

Charlie Donaldson, Esquire
Assistant Attorney General
New York Department of Law
120 Broadway
New York, New York 10271

ENCLOSURE 1

ATTENDEE LIST

May 11, 1989 Meeting to Discuss IP-2 Steam Generator Girth Weld and Feedwater Nozzle Cracking

NAME	TITLE	ORGANIZATION
D. S. Brinkman	Senior Project Manager	NRC/NRR
R. A. Capra	Director PDI-1	NRC/NRR
Michael L. Miele	Gen'l Mgr. Technical Services	Con Ed
Richard Lambert	Chief Mechanical Eng	Con Ed
Charles J. Durkin, Jr.	VP-Engineering	Con Ed
Robert Spring	Principal Nuclear Engineer	Con Ed
George Wasileniko	Principal Consultant QA	Con Ed
Barry J. Elliot	Sr. Matl. Engineer	NRC/NRR
Al R. Vaia	Manager, Nuclear Services Engr.	Westinghouse
Jack Strosnider	Chief, Engineering Branch	NRC/Region I
Robert Hermann	Chief, Metallurg Sect., EMTB	NRC/NRR
C. Y. Cheng	Chief, Mat. Eng. Branch	NRC/DESR/EMTB
Gutti V. Rao	Sr. Engr., Materials Tech	Westinghouse
Carl Czajkowski	Research Engr./BNL	Brookhaven Nat'l Lab
Arnold Lee	Mechanical Engineer	NRC/NRR
Al Taboada	Sr Mat Eng	NRC/RES
Martin Hum	Materials Engineer	NRC/NRR
Pryor N. Randall	Matl Eng Branch	NRC/NRR
Warren H. Bamford	Fellow Engineer	W Matls Tech
Rick D. Rishel	Senior Engineer	W Matls Tech
Charles V. Hayes	Sr Operations Analysis Eng	Con Ed
David A. DelSignore	Sr. Materials & Welding Eng	W Matls Tech
Mimi Weaver	W RCS Components Lic	Westinghouse-NS
Steve Swigart	W Project Manager	Westinghouse
Alex Ball, Jr.	W Operating Plant Lic	Westinghouse
Jim Gasperini	W Projects	Westinghouse
W. Alan Homyk	Mgr, Chemistry & Radwaste	Con Ed

Attendees Cont'd

NAME	TITLE	ORGANIZATION
Serge Brozski	Sr. QA Engineer	Con Ed
Peteris Skulte	Sr. Engineer	Con Ed
Sam Sinha	Stress Analysis	Con Ed
Manvel Marina	Mechanical Engineer	Con Ed
Chong Chiu	Consultant	
Herb Kaplan	Reactor Insp	NRC
Dave Sellers	Eng	NRC
John Houtman	Mgr. Struct. Mech's	Westinghouse

ENCLOSURE 2

CONSOLIDATED EDISON/NRC MEETING STEAM GENERATOR GIRTH WELD AND FEEDWATER NOZZLE REPAIRS

MAY 11, 1989 — NRC OFFICES

AGENDA

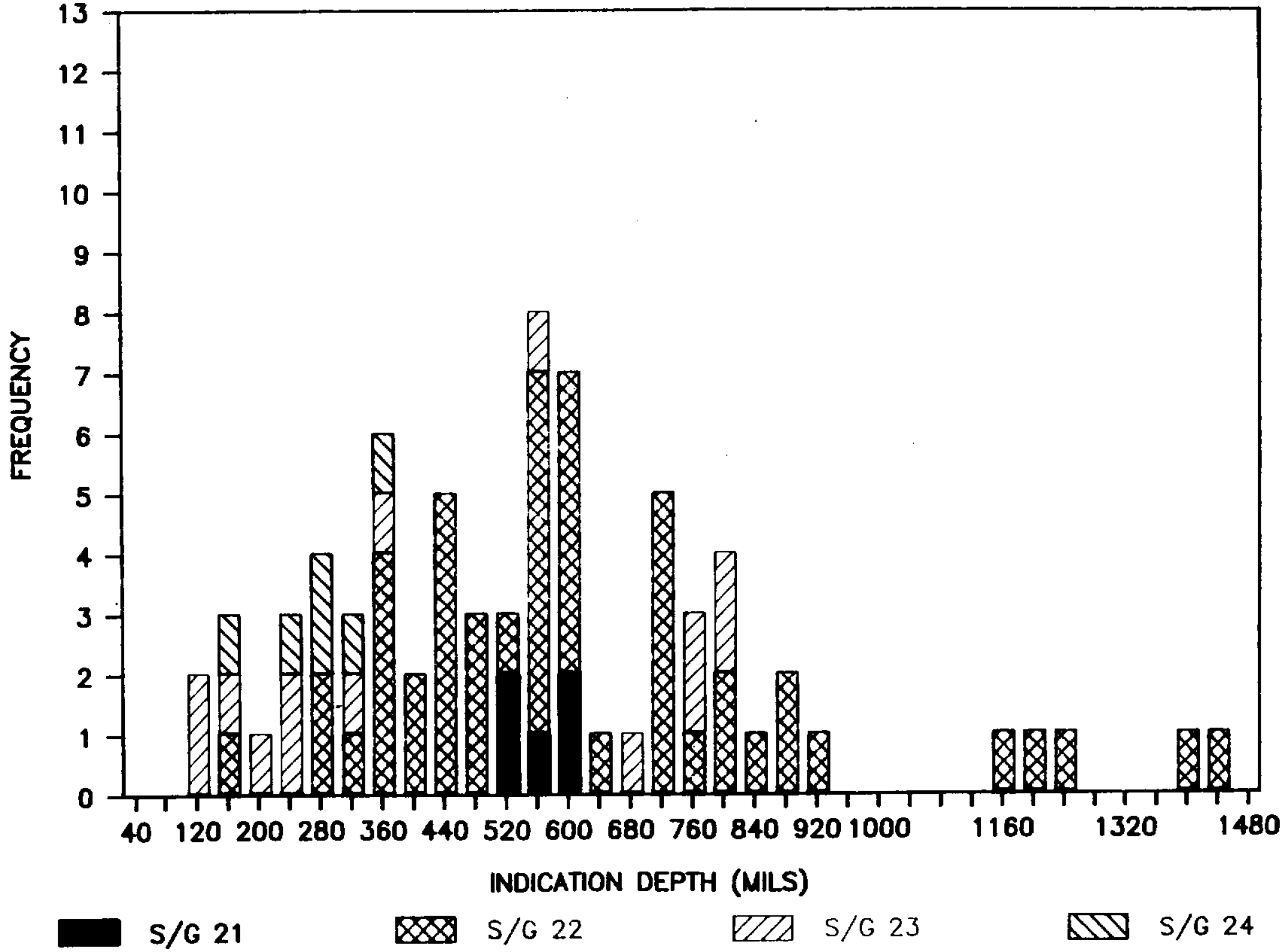
I.	INTRODUCTION	C. DURKIN
II.	ACTION PLAN OVERVIEW	R. LAMBERT
III.	SUMMARY OF INSPECTION RESULTS	G. WASILENKO
IV.	STEAM GENERATOR GIRTH WELD DETAILED DISCUSSIONS	(PROPRIETARY)
	A. Industry Experience	A. VAIA
	B. Metallurgical Examinations	C. RAO
	C. Indian Point 2 Experience	A. VAIA
	D. Feedwater Thermal Hydraulics	J. HOUTMAN
	E. Fatigue Usage Evaluations	J. HOUTMAN
	F. Downcomer Plate History	J. HOUTMAN
	G. Fracture Mechanics Evaluations	W. BAMFORD
	H. In-Service Inspection	G. WASILENKO
	I Stress Relief Process	A. VAIA
	J. Independent Consultant Review	C. CHIU
V.	SUMMARY OF GIRTH WELD ACTION PLAN	R. LAMBERT

VI. FEEDWATER NOZZLE DETAILED DISCUSSIONS (PROPRIETARY)

A. Inspection Results	R. RISHEL
B. Industry Experience	J. HOUTMAN
C. Root Cause Discussion	J. HOUTMAN
D. Stress and Fatigue Evaluations	J. HOUTMAN
E. Fracture Mechanics Evaluations	W. BAMFORD
F. In-Service Inspection	G. WASILENKO

VII. SUMMARY OF FEEDWATER NOZZLE ACTION PLAN — R. LAMBERT

INDIAN POINT STATION UNIT 2
STEAM GENERATOR GIRTH WELD
1989 FINAL DEPTH



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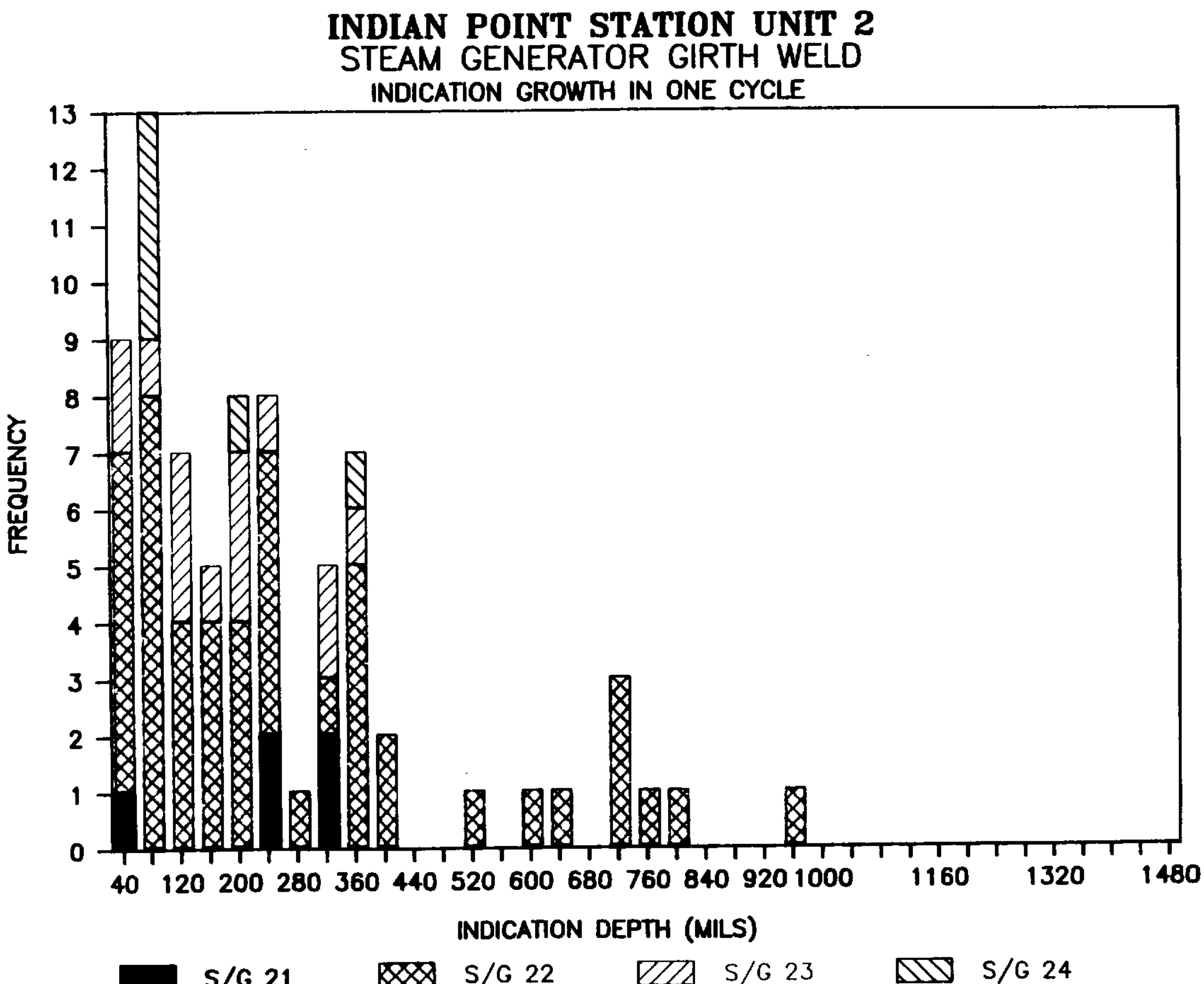

INDIAN POINT STATION UNIT 2
STEAM GENERATOR GIRTH WELD
INDICATION GROWTH IN ONE CYCLE
FREQUENCY
0
1
2
3
4
5
6
7
8
9
10
11
12
13
40
120
200
280
360
440
520
600
680
760
840
920
1000
1160
1320
1480
INDICATION DEPTH (MILS)
S/G 21
S/G 22
S/G 23
S/G 24

5/12/89

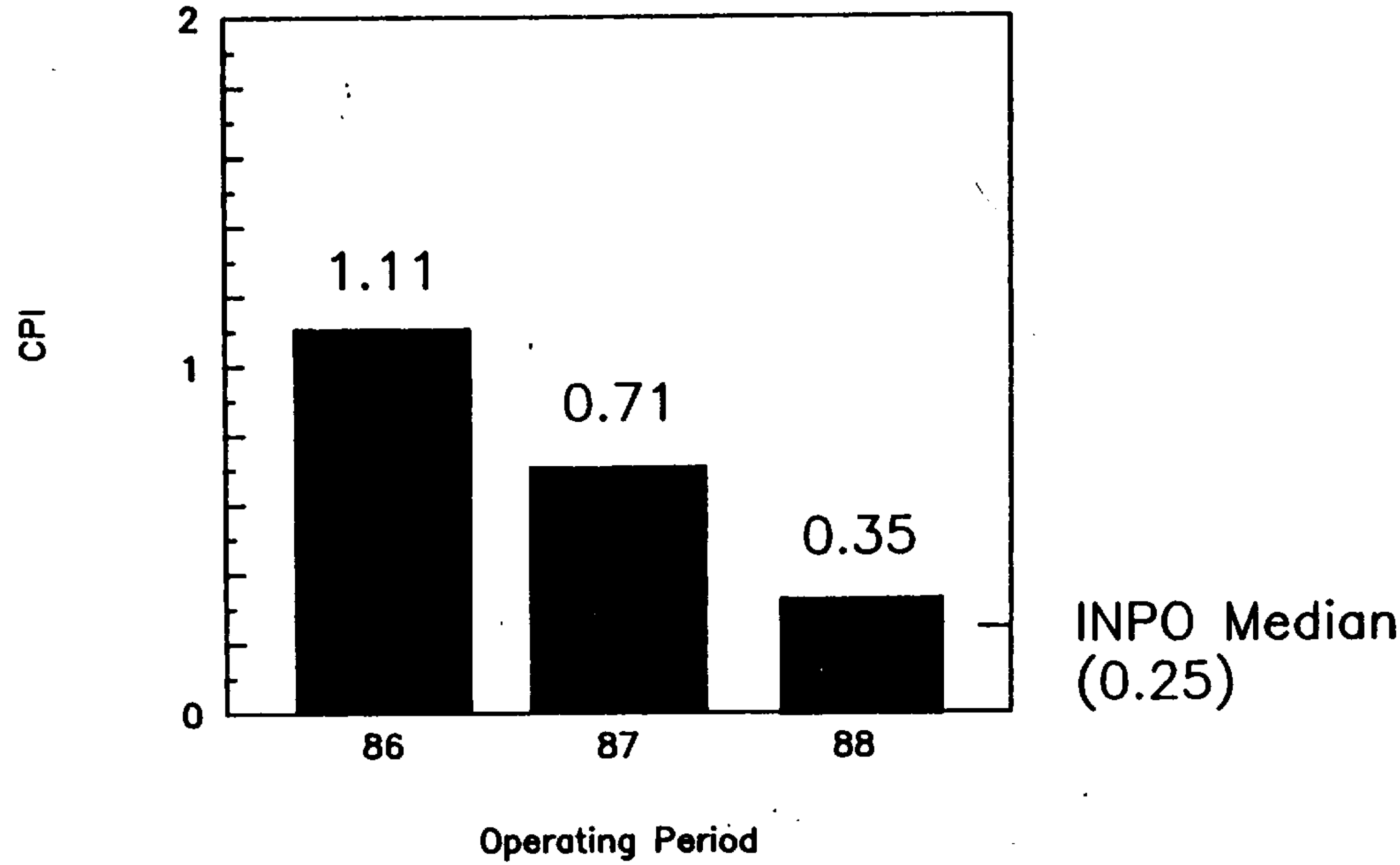
JMF

Chemistry Performance Index (CPI)

* Consists of five secondary–side water quality parameters
  * Condensate System – Dissolved Oxygen (10 ppb)
  * Steam Generator Blowdown
    - Chloride (20 ppb)
    - Sulfate (20 ppb)
    - Sodium (20 ppb)
    - Cation Conductivity (0.8 uS/cm)
* Used as a report card to track station performance over time and relative to the industry
* Implemented by INPO effective January 1988

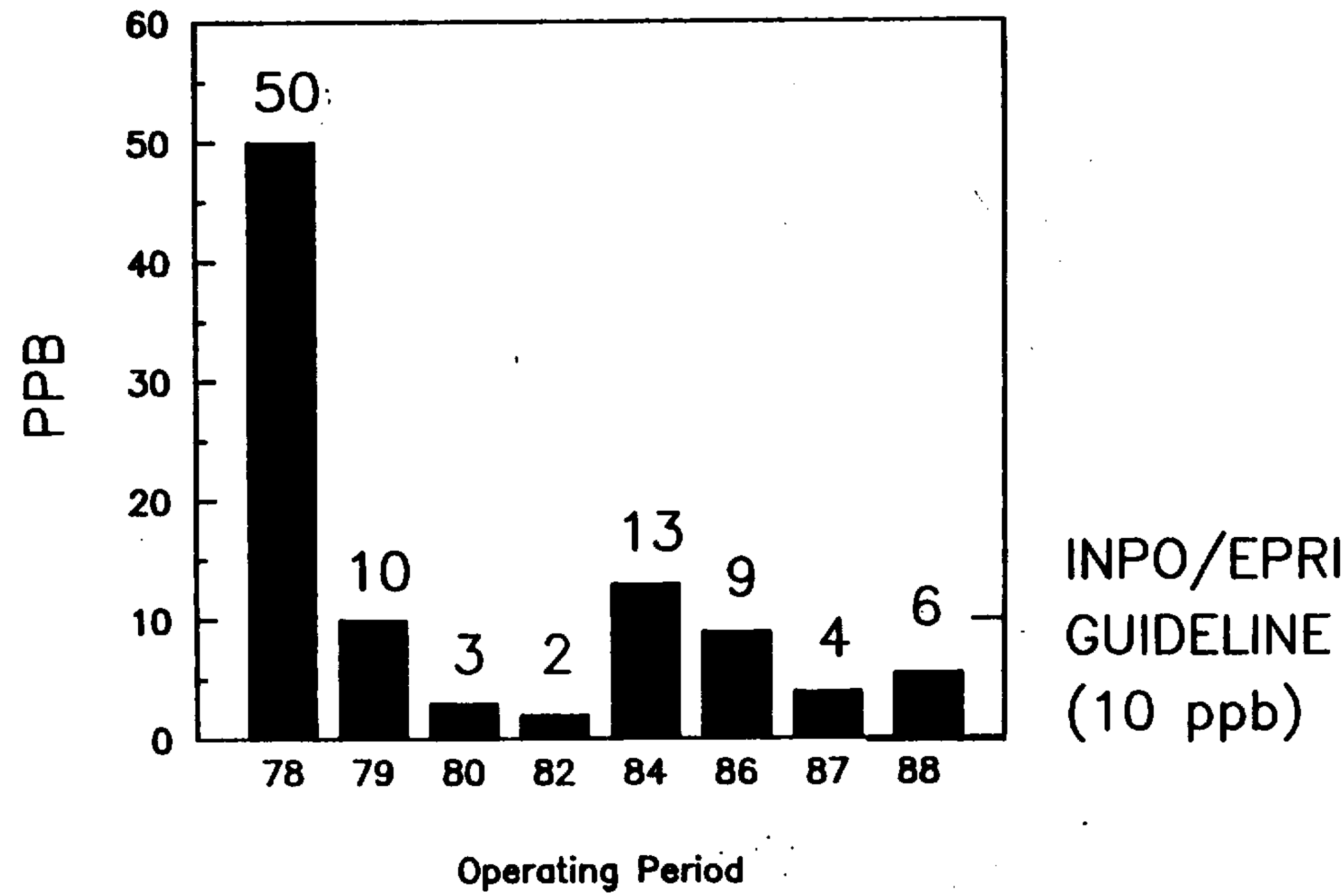
Indian Point #2

INPO Chemistry Performance Index (CPI)



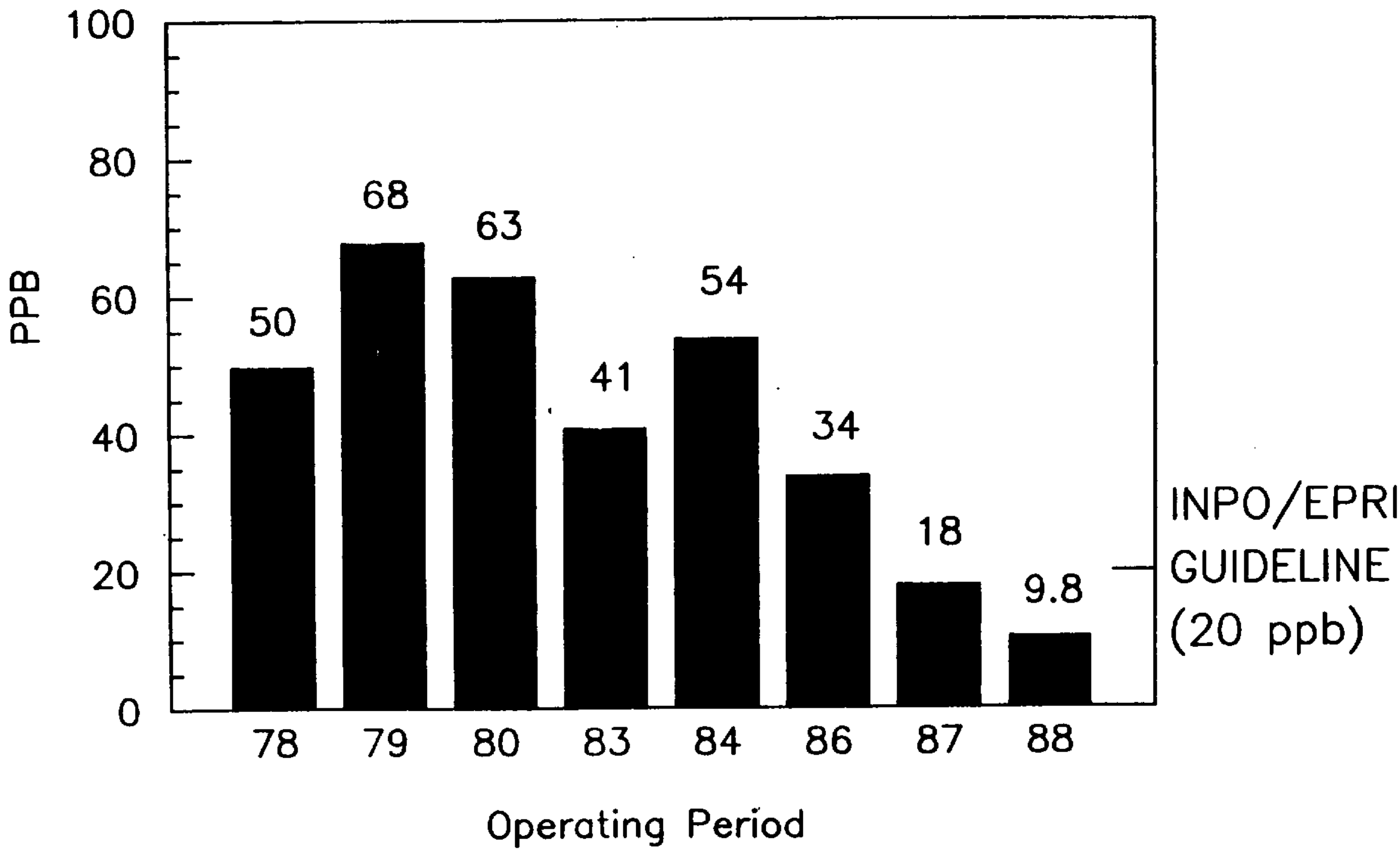
Indian Point #2

Condensate Dissolved Oxygen



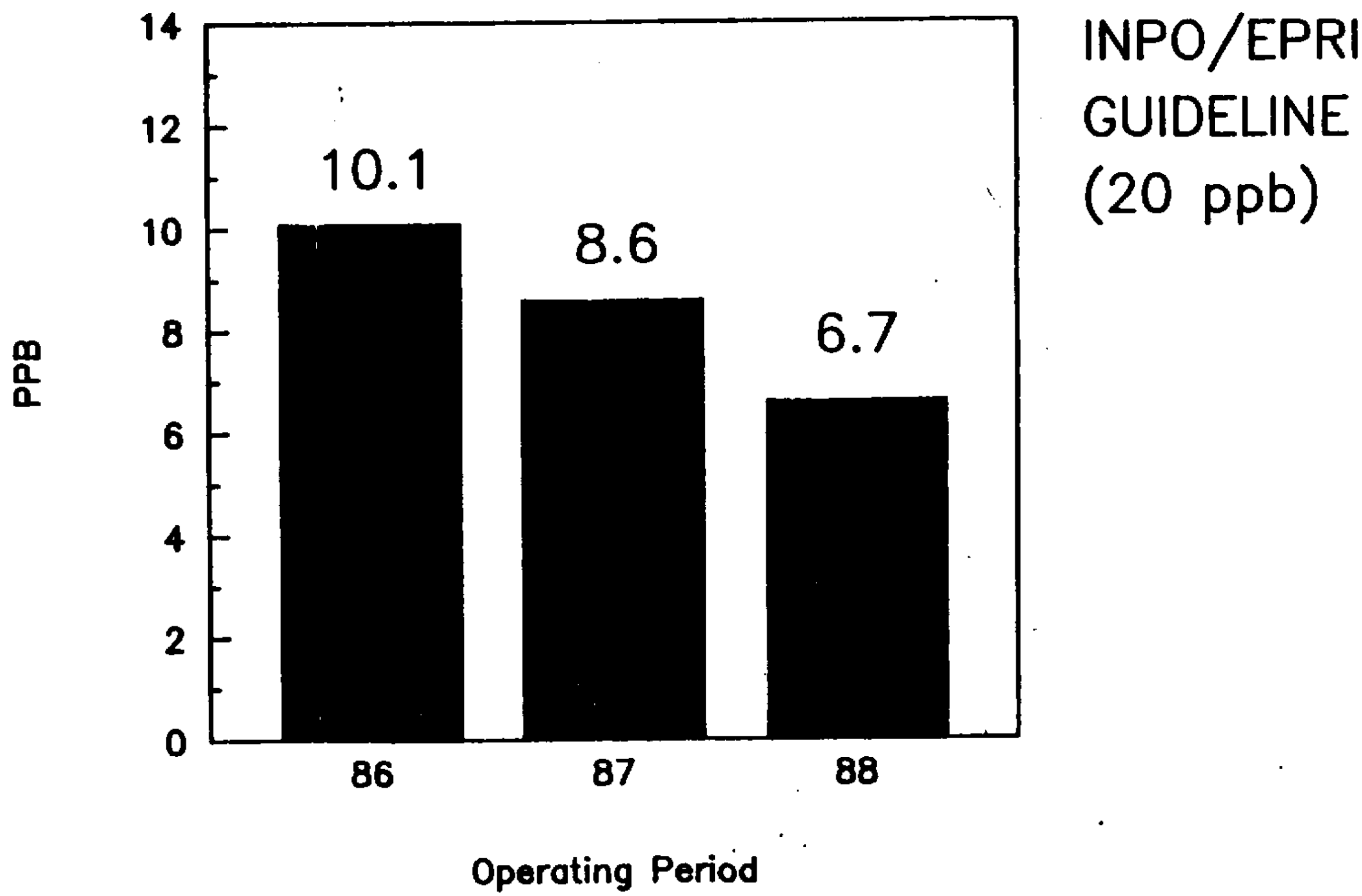
Indian Point #2

Steam Generator Blowdown Chloride



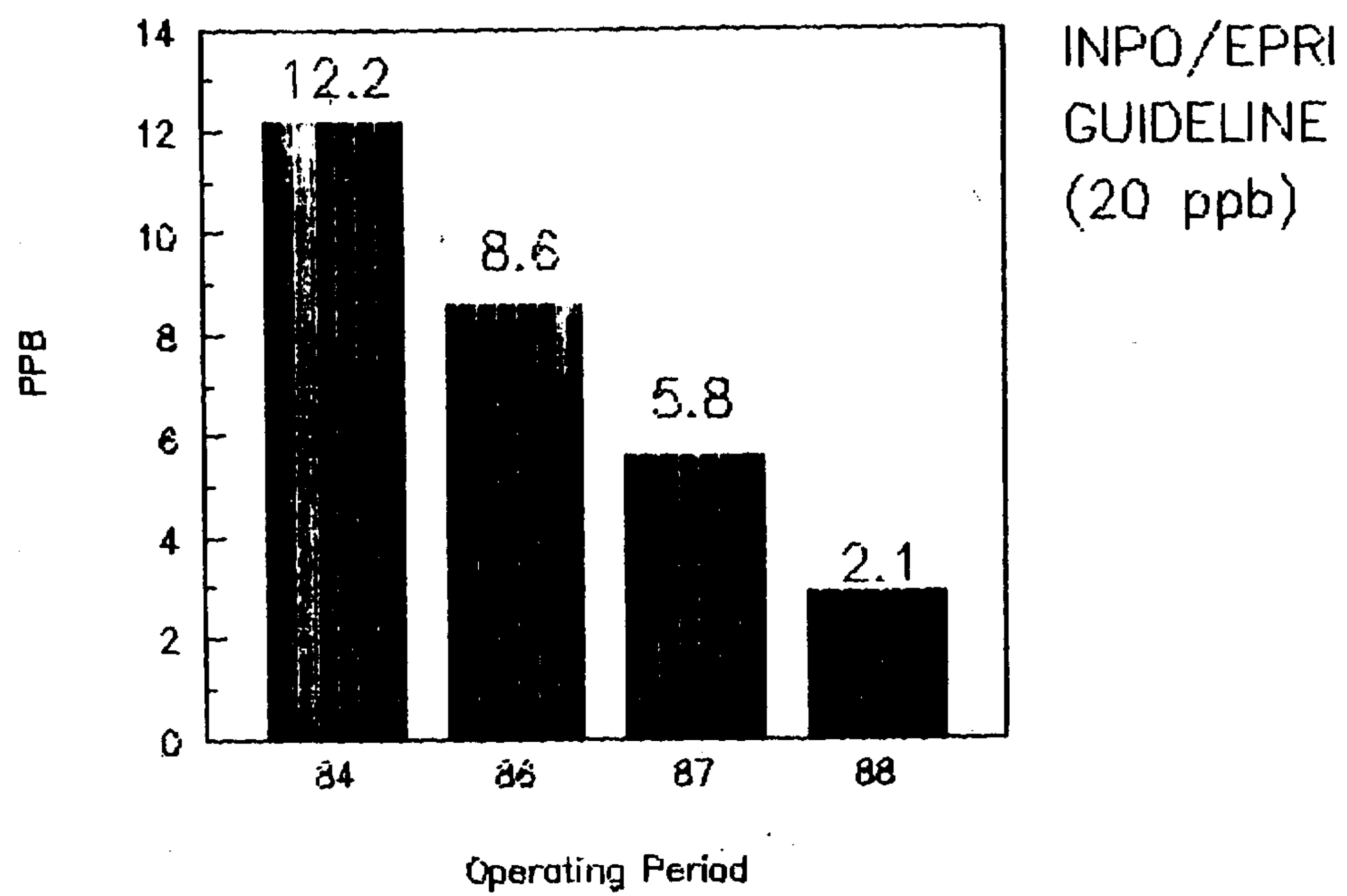
Indian Point #2

Steam Generator Blowdown Sulfate



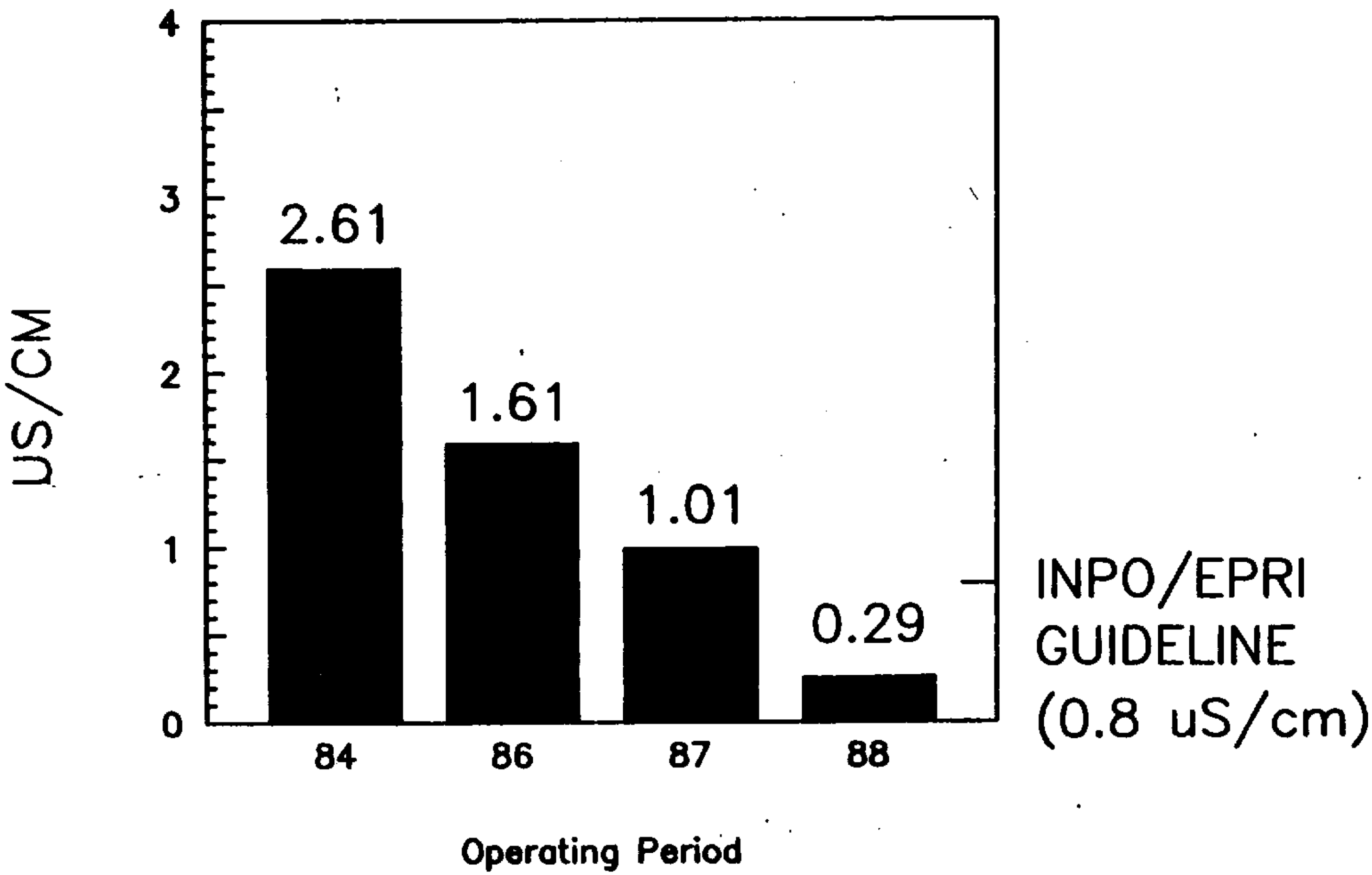
Indian Point #2

Steam Generator Blowdown Sodium



Indian Point #2

Steam Generator Blowdown Cation Conductivity



Actions Implemented to Improve Secondary Side Chemistry

* Prefiltration
* Carbon Filters
* Feedwater Heater Replacement
* Hydrazine Addition To Turbine Hood Sprays
* High Temperature Magnetic Filtration
* Copper Soaks
* Sludge Lance (CECIL)
* Boric Acid Addition
* Switch from Phosphate to AVT
* On–Line Instrumentation
  (73 on–line instruments and 38 grab sample points)

Table 10

Sludge Removal

SG	1989	1987
21	408 lbs*	306 lbs
22	168 lbs	201 lbs
23	131 lbs	316 lbs
24	265 lbs	226 lbs
Total	972 lbs	1049 lbs

*CECIL used only in SG 21.

ATTACHMENT F
[NOT INCLUDED]

ATTACHMENT G

1990 Annual Report

participated in the optional contributory program prior to retirement.

In December 1990, the Financial Accounting Standards Board issued SFAS No.106, "Accounting for Postretirement Benefits Other than Pensions." The standard, which will become effective for calendar year 1993, requires accrual of benefit costs over a shorter period than present methods and at higher assumed rates of escalation which would substantially increase such expenses. This matter is under review by the PSC. The Company believes that any increase in expense which results from the adoption of SFAS No.106 in 1993 will be recovered in rates.

NOTE G CONTINGENCIES

INDIAN POINT Nuclear generating units similar in design to the Company's Indian Point 2 unit have experienced problems of varying severity in their steam generators, which in several instances have required steam generator replacement. Inspections of the Indian Point 2 steam generators since 1976 have revealed various problems, which appear to be progressive. The remaining service life of the steam generators is uncertain and may be shorter than the unit's life. The projected service life of the steam generators is reassessed periodically in the light of inspections made during scheduled outages of the unit. Based on data from the latest inspection and other sources, the Company presently estimates that steam generator replacement will not be required before 1993, and possibly not until some years later. To avoid procurement delays in the event replacement is necessary, the Company has purchased, and stored at the site, replacement steam generators. If replacement of the steam generators is required, such replacement is presently estimated (in 1990 dollars) to require additional expenditures of approximately $127 million (exclusive of replacement power costs) and an outage of approximately six months. However, securing necessary permits and approvals or other factors could require a substantially longer outage if steam generator replacement is required on short notice.

NUCLEAR INSURANCE The insurance policies covering the Company's nuclear facilities for property damage, excess property damage, and outage costs permit assessments under certain conditions to cover insurers' losses. As of December 31, 1990, the highest amount which could be assessed for losses during the current policy year under all of the policies was $22.1 million. While assessments may also be made for losses in certain prior years, the Company is not aware of any losses in such years which it believes are likely to result in an assessment.

Under certain circumstances, in the event of nuclear incidents at facilities covered by the federal government's third-party liability indemnification program, the Company could be assessed up to $66 million per incident of which not more than $10 million may be assessed in any one year. This limit is subject to adjustment for inflation in future years.

The Company participates in an insurance program covering liabilities for injuries to certain workers in the nuclear power industry. In the event of such injuries, the Company is subject to assessment, up to an estimated maximum of approximately $3.2 million.

SUPERFUND CLAIMS The federal Comprehensive Environmental Response, Compensation and Liability Act of 1980 ("Superfund") by its terms imposes strict liability, regardless of fault, upon generators of hazardous substances for resulting removal and remedial costs and environmental damages. The Company has received process or notice concerning possible claims under Superfund relating to several sites at which it is alleged that hazardous substances generated by the Company (and, in most instances, a large number of other potentially responsible parties) were deposited. The total removal, remedial and environmental damage costs at certain sites may be substantial, but based on the information and relevant circumstances known to the Company at this time, it is the opinion of the Company that its share of these costs will not have a material adverse effect on the Company's financial position.

ASBESTOS CLAIMS Suits were brought in New York State and federal courts against the Company and many other defendants, wherein several hundred plaintiffs sought large amounts of compensatory and punitive damages for deaths and injuries allegedly caused by exposure to asbestos at various premises of the Company. Many of these suits have been disposed of for immaterial amounts. The amounts specified in the remaining suits exceed $10 billion but the Company believes that these amounts are greatly exaggerated, as were the claims already disposed of. Extensive pretrial discovery will be required to evaluate the plaintiffs' claims fully. However, based on the information and relevant circumstances known to the Company at this time, it is the opinion of the Company that these suits will not have a material adverse effect on the Company's financial position.

1990 ANNUAL REPORT UPDATE

INDIAN POINT STEAM GENERATOR REPLACEMENT ESTIMATE

S U M M A R Y		MILLIONS $
- CAPITAL		
PURCHASE STEAM GENERATORS & STORAGE		36
INSTALL GENERATORS PLUS ASSOCIATED WORK		61
COMPANY LABOR & NPG SUPPORT SERVICES		5
SUBTOTAL		102
CONTINGENCY		10
ASSOCIATED OVERHEADS		14
SUBTOTAL CAPITAL		126
- REMOVAL		
* NET RETIREMENT	($ 50,300,000)	
* ORIGINAL COST	($ 13,241,000)	
REMOVAL COST	$ 37,059,000	37
- EXPENSE		0
TOTAL		163
SUBTRACT STEAM GENERATORS		-36
		127

NOTES: 1) ESTIMATE BASED ON CURRENT COST
2) ESCALATION AND AFDC ARE NOT INCLUDED

ESTIMATING SECTION UPDATED: 01/14/91

PROJECT NO...52429-00
BUDGET NO....
ESTIMATE NO..90-2125-OM
EST. DATE....02/14/91
PROJ. ENG....W. BENNETT
PROJ. EST....J. LOMBARDO

ENGINEERING ESTIMATING

ORDER OF MAGNITUDE ESTIMATE

APPROP. START COMPL
ENG/DES START COMPL
PROCUR. START COMPL
CONSTR. START 03/01/91 COMPL 12/01/91
PROJECT IN SERVICE..........: 12/01/91
OUTAGE IS REQUIRED.

LOCATION.....INDIAN POINT GENERATING STATION
DESCRIPTION..REPLACEMENT OF STEAM GENERATORS.

ITEM	COMPANY MHRS	COMPANY LABOR$	COMPANY EQ/MAT$	CONTRACT LABOR AND MATERIAL $	TOTAL DIRECT	0% ESCAL	OVERHEADS & AFDC	VARIOUS CONTING	TOTALS
PURCHASE EQUIPMENT									
STEAM GENERATORS			35000000		35000000		4495400		39495400
CONSTRUCTION CONTRACTS									
TRANSP,RIG&NEW STORAGE BLDG.				1260000	1260000		161800		1421800
SUPPORT FAC.				5442000	5442000		698900	15%	7062000
OLD S/G STORAGE				1873000	1873000		240600	15%	2430600
PLANT MODIFICATIONS				4909000	4909000		630500	15%	6370400
TEMP. SYSTEMS				803000	803000		103100	15%	1042000
INTERFERENCE REMOVAL				940000	940000		120700	15%	1219800
NEW S/G INSTALLATION				12891000	12891000		1655700	15%	16728700
TRANSPORT & RIGGING				716000	716000		91900	15%	929100
INTERFERENCE REPLACEMENT				3646000	3646000		468200	15%.	4731300
PLANT RESTORATION				1660000	1660000		213200	15%	2154200
PLANT RETURN TO SERVICE				2203000	2203000		283000	15%	2858900
TENSION DEVICE				2253000	2253000		289400	15%	2923800
RADIATION SERVICES				15893000	15893000		2041300	15%	20624400
SUPPORT SERVICES				4972000	4972000		638600	15%	6452200
COMPANY LABOR									
SUPPORT LABOR		2376000			2376000		908600	15%	3777300
P. M. &I. / ADMIN.		596000			596000		227900	15%	947500
OTHER DIRECT COSTS									
N.P SUPPORT SERVICES				2182000	2182000		280200	15%	2831500
		2972000	35000000	61643000	99615000		13549000	10836900	124000900

SAY $ 124,000,000 (2 CUTS)

SAY $ 126,000,000 (4 CUTS)

CAPITAL ESTIMATE TOTAL - $ 124,000,000   ASSOCIATED RETIREMENT - $ 50,302,000   ASSOCIATED EXPENSE - NONE

OVERHEADS: 7.00 % ENGINEERING; 3.80 % A & S; 25.40 % P'ROLL TAX & PENS; TOTAL OH'S = $ 13,549,000
(OH'S) ($ 6,972,800) ($ 4,050,200) ($ 2,526,000)

REMARKS COST BASED ON WESTINGHOUSE ESTIMATE.

ESTIMATING SEC.
APPROVED BY................../.../...   REVIEWED BY................../.../...   ORGANIZATION................../.../...

ATTACHMENT H

INDIAN POINT 2
STEAM GENERATOR

PRESENT WORTH COST (PWC) OF ALTERNATIVES
(1990 ESTIMATE)

- LARGE REPAIRS IN 91, NEW CONDENSERS IN 93, AND
  - NEW SGs IN 1995 --- PWC = $359 MILLION
  - NEW SGs IN 1997 --- PWC = $336 MILLION

- NECESSARY REPAIRS IN 91, NEW CONDENSERS IN 93, AND
  - NEW SGs IN 1993 --- PWC = $338 MILLION

BREAKDOWN OF COST STREAM FOR INDIAN POINT 2 STEAM GENERATOR REPLACEMENT ALTERNATIVES (IN MILLIONS $) (1990 APPROXIMATION)

OPTION: LARGE REPAIRS IN 91/NEW SG IN 95
PRESENT WORTH COSTS = $359 MILLION

YEAR -------	91	92	93	94	95	96	97	98	99
1. SG FIX/INSPECTION	39	5	10	5					
2. SG REPLACEMENT									
CAPITAL					152				
ENERGY REPLACE	15	7.5		15	60				
EXPENSES			10		43				
3. CONDEN/FWH REPLACEMT									
CAPITAL			25						
EXPENSES			6						

OPTION: LARGE REPAIRS IN 91/NEW SG IN 97
PRESENT WORTH COSTS = $336 MILLON

YEAR -------	91	92	93	94	95	96	97	98	99
1. SG FIX/INSPECTION	39	5	5		10	10			
2. SG REPLACEMENT									
CAPITAL							162		
ENERGY REPLACE	15	7.5				15	60		
EXPENSES					10		48		
3. CONDEN/FWH REPLACEMT									
CAPITAL			25						
EXPENSES			6						

OPTION: NECESSARY REPAIRS IN 91/NEW SG IN 93
PRESENT WORTH COSTS = $338 MILLION

YEAR -------	91	92	93	94	95	96	97	98	99
1. SG FIX/INSPECTION	15	10							
2. SG REPLACEMENT									
CAPITAL			143						
ENERGY REPLACE		15	60						
EXPENSES			49						
3. CONDEN/FWH REPLACEMT									
CAPITAL			25						
EXPENSES			6						

ATTACHMENT I

INDIAN POINT 2
STEAM GENERATORS

PRESENT WORTH COST (PWC) OF ALTERNATIVES

- LARGE REPAIRS IN 91, NEW CONDENSERS IN 93, AND
  - NEW SGs IN 1995 --- PWC = $402 MILLION
  - NEW SGs IN 1997 --- PWC = $379 MILLION
  - NEW SGs IN 2003 --- PWC = $318 MILLION
- NECESSARY REPAIRS IN 91, NEW CONDENSERS IN 93, AND
  - NEW SGs IN 1993 --- PWC = $414 MILLION

BREAKDOWN OF COST STREAM FOR INDIAN POINT 2 STEAM
GENERATOR REPLACEMENT ALTERNATIVES (IN MILLIONS $)

OPTION: LARGE REPAIRS IN 91/NEW SG IN 2003
PRESENT WORTH COSTS = $318 MILLION

YEAR -------	91	92	93	94	95	96	97	98	99	0	1	2	3
1. SG FIX/INSPECTION	34		2		3		3		3		3		
2. SG REPLACEMENT													
CAPITAL			6									155	66
ENERGY REPLACE	15												60
EXPENSES			19										
3. CONDEN/FWH REPLACEMT													
CAPITAL	10	13	10										
4. ADDITIONAL MAN-REM	8												11

OPTION: NECESSARY REPAIRS IN 91/NEW SG IN 93
PRESENT WORTH COSTS = $414 MILLION

YEAR -------	91	92	93	94	95	96	97	98	99
1. SG FIX/INSPECTION	15	10							
2. SG REPLACEMENT									
CAPITAL	6	102	44						
ENERGY REPLACE		15	60						
EXPENSES	19								
3. CONDEN/FWH REPLACEMT									
CAPITAL	10	13	10						
4. ADDITIONAL MAN-REM	4		11						

BREAKDOWN OF COST STREAM FOR INDIAN POINT 2 STEAM
<u>GENERATOR REPLACEMENT ALTERNATIVES (IN MILLIONS $)</u>

OPTION: LARGE REPAIRS IN 91/NEW SG IN 95
PRESENT WORTH COSTS = $402 MILLION

YEAR -------	91	92	93	94	95	96	97	98	99
1. SG FIX/INSPECTION	34		2						
2. SG REPLACEMENT									
CAPITAL			6	110	49				
ENERGY REPLACE	15				60				
EXPENSES			19						
3. CONDEN/FWH REPLACEMT									
CAPITAL	10	13	10						
4. ADDITIONAL MAN-REM	8				11				

OPTION: LARGE REPAIRS IN 91/NEW SG IN 97
PRESENT WORTH COSTS = $379 MILLON

YEAR -------	91	92	93	94	95	96	97	98	99
1. SG FIX/INSPECTION	34		2		3				
2. SG REPLACEMENT									
CAPITAL			6			120	53		
ENERGY REPLACE	15						60		
EXPENSES			19						
3. CONDEN/FWH REPLACEMT									
CAPITAL	10	13	10						
4. ADDITIONAL MAN-REM	8						11		