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May 27, 1983

Docket No. 50-286

Mr. J. P. Bayne
Executive Vice President - Nuclear Generation
Power Authority of the State of New York
123 Main Street
White Plains, New York 10601

Dear Mr. Bayne:

SUBJECT: STEAM GENERATOR TUBE AND GIRTH WELD REPAIRS AT THE INDIAN POINT NUCLEAR GENERATING PLANT, UNIT NO. 3 (IP-3)

The Commission has issued the enclosed Amendment No. 47 to Facility Operating License No. DPR-64 for the Indian Point Nuclear Generating Plant, Unit No. 3. The amendment modifies your license to reflect repairs and modifications to steam generator tubes and girth welds as requested by your submittals dated October 18, 1982, as supplemented by letters dated January 19, 1983, May 2, 1983 and May 3, 1983.

By letter dated October 18, 1982, you requested that the Indian Point Technical Specifications be revised in the areas of tube inspection, tube plugging limit, corrective measures and sleeve plugging limit. The amendment approves your steam generator tube sleeving/plugging program as well as your plans with respect to mitigation of worker radiation doses. Please note that the approval is for one fuel cycle (Cycle 4) and that our findings are subject to the conditions: (1) that a mid-cycle inservice inspection of the steam generator tubes be conducted in consonance with the revised plant Technical Specifications, and (2) that the status and schedules for completion of plant modifications be forwarded by January 1, 1984, in consonance with your letter of May 2, 1983. As mutually agreed to by members of your staff these conditions have been incorporated into your Technical Specifications and license, respectively.

By letter dated January 19, 1983, you provided information regarding your steam generator girth weld repair program as requested during the site visit of December 20, 1982. The enclosed amendment also approves this program. As such it provides Technical Specifications related to long term augmented inservice inspection girth weld surveillance. Please note that the approval is subject to the condition that a mid-cycle inspection of welds be conducted and preliminary results and corrective measures, if applicable, be forwarded at least five days prior to plant startup. Final results are to be forwarded within 30 days of plant startup. As mutually agreed to by members of your staff this condition has been incorporated into the IP-3 Technical Specifications.

OFFICE ▶							
SURNAME ▶							
DATE ▶							

Copies of the Safety Evaluation, Environmental Impact Appraisal and Notice of Issuance/Negative Declaration are enclosed.

Sincerely,
Original signed by:
S. A. Varga

Steven A. Varga, Chief
Operating Reactors Branch No. 1
Division of Licensing

Enclosures:
1. Amendment No. 47 to DPR-64
2. Safety Evaluation
3. Environmental Impact Appraisal
4. Notice/Negative Declaration

cc w/enclosures: See next page

NRO MEIAL SEP 5/27/83

MTEB DSmith 5/11/83

CHEB VBenaroya 5/18/83

MTEB BLiaw 5/18/83

CHEB VBENAROYA 5/27/83

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MTEB LFrank 5/11/83

~~MTEB PStoddart 5/ /83~~

RAB FCongel 5/5/83

RSB Lantz 5/5/83

CHEB BTurovlin 5/18/83

~~EEB RBallard 5/ /83~~

QAB WHaass 5/5/83

5/27/83

with Notice only

WITH NOTE ATTACHED

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

POWER AUTHORITY OF THE STATE OF NEW YORK

DOCKET NO. 50-286

INDIAN POINT NUCLEAR GENERATING UNIT NO. 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 47
License No. DPR-64

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Power Authority of the State of New York (the licensee) dated October 18, 1982, as supplemented by letters dated January 19, May 2, and May 3, 1983, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by the revisions of Paragraph 2.I and the addition of Paragraph 2.0 to Facility Operating License No. DPR-64 to read as follows:

   2.I The licensee shall implement a secondary water chemistry monitoring program to inhibit steam generator tube degradation. This program shall include:

      1. Identification of a sampling schedule for the critical parameters and control points for these parameters;

      2. Identification of the procedures used to quantify parameters that are critical to control points;

      3. Identification of process sampling points, including monitoring the condenser hot wells for evidence of condenser in-leakage;

      4. Procedure for the recording and management of data;

      5. Procedures defining corrective actions for off control point chemistry conditions; and

      6. A procedure identifying the authority responsible for the interpretation of the data, and the sequence and timing of administrative events required to initiate corrective action.

   2.0 Evaluation, status and schedule for completion of balance of plant modifications as outlined in letter dated February 12, 1983, shall be forwarded to the NRC by January 1, 1984.

3. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C(2) of Facility Operating License No. DPR-64 is hereby amended to read as follows:

   (2) <u>Technical Specifications</u>

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 47, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

4. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Steven A. Varga

Steven A. Varga, Chief
Operating Reactors Branch No. 1
Division of Licensing

Attachment:
Changes to the Technical Specifications

Date of Issuance: May 27, 1983

ATTACHMENT TO LICENSE AMENDMENT NO. 47

FACILITY OPERATING LICENSE NO. DPR-64

DOCKET NO. 50-286

Revise Appendix A as follows:

Remove Pages	Insert Pages
4.9-1	4.9-1 effective for Fuel Cycle 4
4.9-1a	4.9-1a " " " " 4
4.9-4	4.9-4 " " " " 4
4.9-5	4.9-5 " " " " 4
4.9-6	4.9-6 " " " " 4
4.2-3	4.2-3 4.2-3a
Table 54.2-1 (Sheet 8 of 12)	Table 4.2-1 (Sheet 8 of 12) Table 4.2-1 (Sheet 8a of 12)
6-19	6-19

4.9 STEAM GENERATOR TUBE INSERVICE SURVEILLANCE

Applicability

Applies to inservice surveillance of the steam generator tubes.

Objective

To assure the continued integrity of the steam generator tubes that are a part of the primary coolant pressure boundary.

Specification

Steam generator tubes shall be determined operable by the following inspection program and corrective measures:

A. Inspection Requirements

1. Definitions

a. Imperfection is an exception to the dimension, finish, or contour required by drawing or specification.

b. Degradation means a service-induced cracking, wastage, wear or corrosion.

c. Degraded Tube is a tube that contains imperfections caused by degradation large enough to be reliably detected by eddy current inspection. This is considered to be 20% degradation.

d. % Degradation is an estimate % of the tube wall thickness affected or removed by degradation.

e. Defect is an imperfection of such severity that it exceeds the plugging limit. A tube containing a defect is defective.

f. Tube Plugging Limit is the tube imperfection depth at or beyond which the tube must either be removed from service or repaired. This is considered to be an imperfection depth of 40%. However, for the purposes of identifying defective tubes due to pitting between the tube sheet and first support plate of the cold leg side of all four steam generators, the tube plugging limit shall be an imperfection depth of 50% or greater.

g. Sleeve Plugging Limit - is the sleeve imperfection depth at or beyond which the sleeved tube must be removed from service or repaired. This is considered to be an imperfection depth of 40% for tube sleeves.

h. <u>Tube I pection</u> is an inspection of tubes from the point of entry (hot leg side) completely around the U-bend to the top support of the cold leg. However, for purposes of the inspection performed as a result of the March 24, 1982 tube leak on the cold leg side of SG-33, the inspection required by Table 4.9-1 may be performed on the cold leg side of the steam generators up to the second support plate on that side, except that in at least one steam generator, the inspection shall extend to the sixth tube support plate on the cold leg side.

4. Interval of Inspection

a. The first inservice inspection of steam generators should be performed after six effective full power months but not later than completion of the first refueling outage.

b. Subsequent inservice inspections should be not less than 12 or more than 24 calendar months after the previous inspection.

c. If the results of two consecutive inspections, not including the preservice inspection, all fall in the C-1 category, the frequency of inspection may be extended to 40-month intervals. Also, if it can be demonstrated through two consecutive inspections that previously observed degradation has not continued and no additional degradation has occurred, a 40-month inspection interval may be initiated.

d. A special mid-cycle inspection of both steam generator tubes and girth welds shall be conducted during fuel cycle 4. These tests shall be conducted no sooner than after 6 months of power operation and no later than after 9 months of power operation. The girth weld inspection shall be as indicated in the safety evaluation for Amendment No. 47 to License No. DPR-64. The tube inspection shall be as indicated in the Power Authority's letter of October 18, 1982. The test programs themselves for each test shall be forwarded to NRC 30 days prior to implementation.

B. Corrective Measures

All leaking tubes and defective tubes should be: (1) plugged, or (2) repaired.

C. Reports

1. Following each inservice inspection of steam generator tubes, the number of tubes plugged and repaired in each steam generator shall be reported to the Commission within 15 days. (Following the Cycle 4 inservice inspection of steam generator girth welds preliminary results and corrective measures shall be forwarded within five days of plant startup and final results are to be forwarded within 30 days of plant startup.)

2. The complete results of the steam generator tube inservice inspection shall be reported in writing on an annual basis for the period in which the inspection was completed per Specification 6.9.2.f. (The final Cycle 4 mid-cycle test report shall be forwarded within 30 days of plant startup.) This report shall include:

a. Number and extent of tubes inspected.

b. Location and percent of wall-thickness penetration for each indication of an imperfection.

c. Identification of the tubes plugged and the tubes repaired.

deterioration due to design, manufacturing errors, or chemical imbalance. Inservice inspection of steam generator tubing also provides a means of characterizing the nature and cause of any tube degradation so that corrective measures can be taken.

An essentially 100% tube inspection was performed on each tube in every steam generator by eddy current techniques prior to service in order to establish a baseline condition for the tubing. This inspection was conducted under conditions and with equipment and techniques equivalent to those expected to be employed in the subsequent inservice inspections.

The plant is expected to be operated in a manner such that the secondary coolant will be maintained within those limits found to result in negligible corrosion of the steam generator tubes. If stress corrosion cracking occurs, the extent of cracking during plant operation would be limited by the limitation of steam generator leakage between the primary coolant system and the secondary coolant system. Cracks having a primary-to- secondary leakage less than 500 gallons per day during operation will have an adequate margin of safety against failure due to loads imposed by design basis accidents. Operating plants have demonstrated that primary-to-secondary leakage as low as 0.1 gpm will be detected. Leakage in excess of 432 gallons per day per steam generator or 1 gpm total through all four steam generators will require plant shutdown and an unscheduled eddy current inspection, during which the leaking and defective tubes will be located and either: (1) plugged, or (2) repaired. The 500 gallon per day limit is also consistent with the assumptions used to develop the Technical Specification limit for secondary coolant activity.

Wastage-type defects are unlikely with all volatile treatment (AVT) of secondary coolant. However, even if this type of defect occurs, the steam generator tube surveillance specification will identify steam generator tubes with imperfections having a depth greater than 40% of the 0.050 inch tube wall thickness as being unacceptable for continued service. The results of steam generator tube burst and collapse tests have demonstrated that tubes having wall thickness not less than 0.025 inch have adequate margins of safety against failure due to loads imposed by normal plant operation and design basis accidents.

A 10% allowance for tube degradation that may occur between inservice tube examinations added to the 40% tube plugging limit provides an adequate margin to assure that SG tubes acceptable for operation will not have a minimum tube wall thickness less than the acceptable 50% of normal tube wall thickness (i.e., 0.025 in) during the service lifetime of the tubes.

Steam generator tube inspections of operating plants have demonstrated the capability to reliably detect wastage type defects that have penetrated 20% of the original 0.050 inch wall thickness.

The definition of tube plugging limit also provides that a tube imperfection depth of 50% or greater shall be applied to tubes which have experienced pitting on the cold leg side of a steam generator between the tube sheet and first support plate.

This 10% increase in allowable tube degradation is acceptable since burst tests, corrected to 600°F, of representative tubing with various flaw types, lengths and wall thicknesses, have demonstrated that 25% remaining wall thickness for all flaw lengths is adequate to withstand the max ΔP (2650 psi) calculated to occur during faulted conditions. A 50% plugging limit also incorporates 25% margin. A 10% margin for measurement inaccuracies is considered sufficient, leaving a 15% safety margin for corrosion allowance.

The definition of sleeve plugging limit provides that a sleeve imperfection depth of 40% (.0156 inch) or greater shall be applied to tube sleeves.

The definition of tube inspection also provides that the steam generator inspection conducted as a result of the March 24, 1982 tube leak may be performed on the cold leg sides up to the second support plate on that side except that in at least one steam generator the inspection shall extend up to the sixth tube support plate on the cold leg side. This is acceptable since the leakage which initiated this inspection occurred on the cold leg side and since a 100% inspection of the cold leg side of one steam generator up to the sixth tube support plate on that side revealed negligible defects. In addition, a 100% inspection of the hot leg sides of two steam generators up to the sixth tube support plate revealed negligible defects.

the expiration of one-third of the inspection interval (with credit for no more than 33-1/3 percent if additional examinations are completed) and at least 50 percent shall have been completed by the expiration of two-thirds of the inspection interval (with credit for no more than 66-2/3 percent). The remaining required examinations shall be completed by the end of the inspection interval. Successive inspections shall meet the requirements of Paragraph IS-243 of the ASME Rules for In-Service Inspection of Nuclear Reactor Coolant Systems.

4.2.8 BASES

The inspection program, where practical, is in compliance with Section XI of the ASME Code for In-Service Inspection of Nuclear Reactor Coolant Systems dated January 1970. Though examinations in certain areas are desirable, it should be recognized that equipment and techniques to perform the inspection are still in development. In all areas scheduled for volumetric examination, a detailed pre-service mapping will be conducted using techniques expected to be used for post-operation examinations. The areas indicated for inspection represent those of representative stress levels, and therefore will serve to indicate potential problems before significant flaws develop there or at other areas. As more experience is gained in operation of pressurized-water reactors, the time schedule and location of inspection may be altered or, should new techniques be developed, consideration may be given to incorporate these new techniques into this inspection program.

The techniques for inspection include visual inspections, ultrasonic, radiographic, magnetic particle and liquid penetrant testing of selected parts during refueling periods or other appropriate plant outages.

The augmented inspection of steam generator weld number 6 required by table 4.2.-1., Item No. 3.8 (NOTE), may be deleted with specific approval of the NRC if experience over an interval of approximately three refueling outages or changes of plant components indicate that this augmented inspection is no longer necessary. For this augmented inspection the 45° shear wave method was chosen based on the review of the original ultrasonic data. This search was the most sensitive of the three used (0°, 45° and 60°). It has also been determined that it will be adequate to perform the inspection by UT in the vertical plane only. This method of search will detect cracks parallel to the weld which were typical of those originally found. There were a limited number of cracks reported on the original ultrasonic inspection as transverse, however, in reviewing subsequent radiographs, magnetic particle and liquid penetrant examination results, it is evident that these cracks emanated from defects parallel to the weld.

The inspection requirements of this section shall apply to all pressure-containing components that are part of the system boundary defined herein. Due to the design of Indian Point Unit #3, there may be areas where weld access is impossible due to high radiation and/or physical access problems. Exception is taken to performing inspections in these areas.

TABLE 4.2-1 (Sheet 8a of 12)

Item No.	Examination Category	Components and Parts to be Examined	Method	Extent of Examination (Percent in 10 Year Interval)	Remarks
		PIPING PRESSURE BOUNDARY			
4.1	F	Vessel, pump and valve safe-ends to primary pipe welds and safe-ends in branch piping welds.	UT, PT & V	100%	This examination covers only the pressurizer safe-ends.
4.2	J-1	Circumferential and longitudinal pipe welds and branch pipe connections welds larger than 4 inches in diameter	V & UT	25%	Exception is taken to inaccessible welds and welds where examination techniques limit inspections.
4.3	G-1	Pressure-retaining bolting		Not applicable	
4.4	G-2	Pressure-retaining bolting	V	100%	

Item No.	Examination Category	Components and Parts to be Examined	Method	Extent of Examination (Percent in 10 Year Interval)	Remarks
3.8	–	Secondary side shell welds	UT	See Remarks	The total examination completed over the service life time will be equivalent of having performed 100% of the required examination.

3.8 (NOTE): AUGMENTED STEAM GENERATOR GIRTH WELD INSPECTION

To provide surveillance of the steam generator welds number 6, after the repairs made during the 1982/1983 outage, the Authority will perform ultrasonic inspection using the 45° shear wave method of one hundred and seventy five (175) linear inches of weld. Thirty five (35) inches will be examined on Steam Generators 31, 32 and 33. Seventy (70) inches will be examined on Steam Generator 34.

The following areas have been selected for this augmented examination:

Steam Generator	Location on Circumference	Segment Location
31	204" clockwise to 239" from 0 Reference	17–20
32	316" clockwise to 334" from 0 Reference	26–28
	348" clockwise to 365" from 0 Reference	29–31
33	360" clockwise to 395" from 0 Reference	30–33
34	0 Reference clockwise to 18"	0–2
	505" clockwise to 522" from 0 Reference	42–0
	168" clockwise to 203" from 0 Reference	14–17

a. Records of any drawing changes reflecting facility design modifications made to systems and equipment described in the Final Safety Analysis Report.

b. Records of new and irradiated fuel inventory, fuel transfers and assembly burnup histories.

c. Records of facility radiation and contamination surveys.

d. Records of radiation exposure for all individuals entering radiation control areas.

e. Records of gaseous and liquid radioactive material released to the environs.

f. Records of transient or operational cycles for those facility components designed for a limited number of transient cycles.

g. Records of training and qualifications for current members of the plant staff.

h. Records of in-service inspections performed pursuant to these Technical Specifications.

i. Records of Quality Assurance activities required by the QA manual.

j. Records of reviews performed for changes made to procedures or equipment or reviews of tests and experiments pursuant to 10 CFR 50.59.

k. Records of meetings of the PORC and the SRC.

l. Records for Environmental Qualification which are covered under the provisions of paragraph 6.13.

m. Records of secondary water sampling and water quality.

6.11 RADIATION AND RESPIRATORY PROTECTION PROGRAM

6.11.1 Procedures for personnel radiation protection shall be prepared consistent with the requirements of 10 CFR Part 20 and shall be approved, maintained and adhered to for all operations involving personnel radiation exposure as to maintain exposures as far below the limits specified in 10 CFR Part 20 as reasonable achievable. Pursuant to 10 CFR 20.103 allowance shall be made for the use of respiratory protective equipment in conjunction with activities authorized by the operating license for this plant in determining whether individuals in restricted areas are exposed to concentrations in excess of the limits specified in Appendix B, Table I, Column 1 of 10 CFR 20.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION
RELATED TO AMENDMENT NO. TO FACILITY OPERATING LICENSE NO. DPR-64
POWER AUTHORITY OF THE STATE OF NEW YORK
INDIAN POINT NUCLEAR GENERATING UNIT NO. 3
DOCKET NO. 50-286

1.0 INTRODUCTION

During 1982 two major steam generator repair programs were undertaken at the Indian Point Nuclear Generating Plant, Unit No. 3 (IP-3). These two programs consisted of sleeving and, if necessary, plugging steam generator tubes and the repair of steam generator secondary side upper girth welds.

By letter dated October 18, 1982, Power Authority of the State of New York (licensee) submitted an application for license amendments to the Technical Specifications for Indian Point Nuclear Generating Plant, Unit No. 3. These proposed Technical Specification changes would allow operation of IP-3 with steam generator tubes having degradation exceeding the proposed plugging limit of 50% nominal wall thickness for pitted tubes, provided these tubes have been repaired by insertion of sleeves into the tubes to bridge the degraded or defective portion of the tubes.

To provide a technical basis for proposed sleeve repair program, the licensee has submitted Westinghouse Report WCAP-10145 (Proprietary), Revision 1, dated October 1982, and entitled "Indian Point 3 Steam Generator Sleeving Report Prepared for Power Authority of the State of New York."

The sleeving concept and design are based on observation to date that the tube degradation due to pitting attack has occurred on the cold leg of the tube bundle, confined to a height of approximately two feet above the tubesheet.

By letter dated January 19, 1983, the licensee also provided information regarding steam generator girth weld repair as requested during the site visit of December 20, 1982. Subsequently, by letter dated May 3, 1983, the licensee provided a description of the proposed girth weld surveillance program to be implemented over the life of the plant. In essence, the IP-3 girth weld repair program consists of removing (grinding out) and rewelding of approximately 1200 linear inches of defective welds in all four steam generators.

The evaluation of the licensee's repair programs is provided in the following sections. The major areas of review are: (1) materials engineering and chemical engineering considerations, (2) the effects of modification on reactor physics, and (3) the licensee's worker dose mitigation program as related to "As-Low-As-Reasonably-Achievable" (ALARA) requirements. These review areas are discussed in section 2.1, 2.2 and 2.3, respectively. By letter dated January 11, 1983, the licensee provided the Quality Assurance commitments required to ensure adequate monitoring and review of all steam generator repairs.

2.1 MATERIALS ENGINEERING EVALUATION

2.1.1 Sleeve Congiguration and Sleeving Process

The sleeving process consists of installing, inside the steam generator tube, a smaller diameter tube (sleeve) to span the degraded portion of the parent tube. The sleeves are designed and analyzed in accordance with the 1980 Edition of Section III of the ASME Boiler and Pressure Vessel Code and applicable regulatory guides to restore integrity of degraded steam generators tubes as a primary pressure boundary. The sleeve material and processes also meet the requirements of the Code. The sleeve material composition meets Section II of the Code, while mechanical properties meet Code Case N-20/1484-3, "Ni-Cr-Fe Tubing at Yield Strength of 40,000 psi," 6/22/79.

The sleeves are fabricated from thermally treated Inconel 600 tubing to provide maximum resistance to stress corrosion cracking and pitting. The sleeves are inserted inside the existing tube (mill annealed Inconel 600) and joined to the tube ID at the upper and lower sleeve ends. The sleeves are 0.740 inch OD with a 0.039 inch wall and are in lengths of 36, 40, or 44 inches.

At the upper end, the sleeve configuration consists of a 4.0 inch section which is hydraulically expanded into the original tube, and a section 1.125 inch long which is roll expanded within the 4.0 inch hydraulically expanded section to form an acceptable joint. The sleeve is rolled to a torque sufficient to produce adequate leak tightness and load carrying capability, but within a maximum OD bulge so as not to develop a high residual stress and, therefore, maintain adequate resistance to stress corrosion cracking. The residual stress in the transition region is estimated to be in the range of 20 KSI after the expansion process.

At the lower end, the sleeve configuration consists of a section 4.0 inch long which is hydraulically expanded into the original tube and a section 2.125 inch long which is roll expanded into the original roll expanded portion of the tube to form an acceptable leak tight joint. The sleeve is rolled to a torque sufficient to produce 4 to 6 percent wall reduction in the sleeve wall. This range of wall thinning has been established through laboratory testing as the range which is effective in terms of both leak-tightness and mechanical strength. In this sleeve configuration, the roll expanded region extends from the end of the sleeve to a point just below the roll transition in the original tube. The lower end of the sleeve has a preformed section to facilitate the seal formation.

To minimize stress concentrations and to permit inspectability in the area of the upper expanded region, the transition from the expanded to unexpanded protion of the sleeve is made as gradual as possible. Four representative sleeve expansion transition joints were subjected to corrosion testing in a primary water loop to provide additional assurance of the integrity of the transition joint. The test results are discussed in Section 2.1.16.2.

A considerable amount of actual field experience in installing the mechanical sleeves in tubes has been obtained at San Onofre Unit 1. In addition, experience gained at Point Beach, which utilizes the same model steam generator as Indian Point 3, is directly applicable. In fact, the sleeving processes and parameters successfully employed at Point Beach in the hands-on mode has practically been duplicated for the Indian Point Unit 3 sleeving program. This experience, along with conditions specific to Indian Point Unit 3 steam generators, forms the basis for the process parameters selected for the installation of sleeves in these steam generators. While the overall processes are very similar to the San Onofre program, there are differences in sleeve design and tube dimensions. Nevertheless, the experiences obtained at San Onofre and Point Beach did contribute to the successful mechanical sleeving operation at Indian Point 3.

2.1.2 Post Process Inspection Plan for Sleeved Tubes

Utilizing eddy current equipment and processes specifically developed for verification of the presence of sleeving expansions and determination of sleeve inside diameters of various expanded regions, data were collected on 100 percent of the installed sleeves. After all hydraulic expansions and hard rolls are performed, eddy current testing of each sleeved tube was conducted.

The eddy current data were analyzed for all sleeve installations to verify that all sleeves received the required hydraulic and roll expansions. Additional analyses of the same eddy current data were performed on 10 percent of the sleeve installations from each lot to obtain average diameters of roll regions for additional engineering assurance that equipment/tooling was performed satisfactorily.

The average diameter measurements were evaluated versus the expected tolerances established through the design requirements, laboratory testing results, and previous experience. If process data were determined to be outside of the expected ranges, further dimensional analysis was performed and tubes not satisfying the basic process check criteria were dispositioned on a tube-by-tube basis. Tubes which could be made to meet dimensional specifications were re-expanded. Those outside the dimensional recovery range were plugged.

2.1.3 Inservice Inspection Plan For Sleeved Tubed

In order to assure maintenance of the integrity of this new primary pressure boundary, the regulation requires that periodic inspections of the sleeved tubes be performed. This new pressure boundary consists of the sleeve, the joint at the primary face of the tubesheet, and the joint at the top end of the sleeve.

The Inservice Inspection program consists of the following: The sleeves were eddy current inspected to obtain a base line signature. Periodic inspections will be performed per the Indian Point 3 Technical Specifications to monitor sleeve wall conditions. This inspection will be performed with standard multi-frequency eddy current equipment as used in the base line inspection. The plugging criterion is being established on the same basis as the original tubes; i.e., 40% through-wall flaw produced by degradation mechanisms other than pitting.

In the event degradation occurs in one of the joint regions, other more sensitive techniques can be used to gain additional information about the condition of the sleeve assembly. These techniques involve the use of probes consisting of "cross-wound" coils. It has been demonstrated in the laboratory that such a probe can detect tube wall penetration at the transition region since such a probe is relatively insensitive to the discontinuation with 360 degree symmetry and thus has improved detectibility of tube degradation near the end of the sleeve.

As part of the periodic inspection of the sleeved tubes, there will be a series of pressure tests. These tests will verify the integrity of the mechanical joint sleeve system at a pressure which provides a margin against leakage under normal operating loads, which is 2235 psig primary pressure and 755 psig secondary pressure for a Δp of 1480 psig. The tests will consist of both primary and secondary pressure loadings on the entire tube bundle.

In these tests, the primary to secondary pressure boundary will be exposed to a pressure of 1900 psid. The objective of this test is to establish margins for normal operation conditions and for conditions closely representing a steam line break. Any sleeve joint that may have degraded to the extent that it has an unacceptable load carrying capability should leak.

For the other test of the joints, a secondary side leak test of 800 psi will be imposed on the entire bundle. Surveillance of the primary side will permit the detection of unacceptable sleeved tubes which do not maintain acceptable leak rates. Those sleeved tubes will be repaired or removed from service by plugging.

2.1.4 Leak Rate Tests

An extensive leak testing program was conducted by Westinghouse to establish whether projected leakage during normal operation from the maximum number of sleeves (based on the average leakage per tube in the testing program multiplied by the total number of sleeved tubes) is less than the primary-to-secondary leakage limit specified in the Technical Specifications during normal operation, which is 0.3 gallons per minute primary to secondary leakage per steam generator. Using the above criteria the allowable leak rate per sleeve for normal operation, assuming 5976 sleeves, is 15.2 drops per minute.

Leakage tests were performed for the lower sleeve joint to simulate five years of normal operation with fatigue loading and temperature cycling. The average leak rate was approximately 0.11 drops/minute per lower sleeve joint, which is much less than the 15.2 drops per minute - using the present Technical Specification limits.

Upper sleeve joints were subjected to a series of comprehensive leak tests simulating five years of normal operation based on fatigue loadings and thermal cycles. Leakage did not deviate from an initial leak rate of 0.4 drops per minute per joint after the simulated five years of normal operation with 5000 fatigue cycles and 20 to 32 temperature cycles from $< 150^{\circ}F$ to $600^{\circ}F$ and back to $< 150^{\circ}F$.

Tests that were designed to simulate feedline break accident conditions indicated that leakage during this transient did not exceed the Technical Specification limit for normal operation.

2.1.5 Plugging Limit for Pitted Tubes

The licensee proposed that for the purposes of identifying defective tubes which must be removed from service (plugged) or repaired (sleeved) due to pitting between the tube sheet and first support plate of the cold leg, the tube plugging limit shall be an imperfection depth of 50%.

Technical justification for a 50% plugging limit for pitted tubes is based on tube burst tests on pitted tubes removed from service which indicated that tubes having wall thickness not less than 0.0125 inches have adequate margins of safety against failure loads imposed by normal plant operation and design basis accidents.

Based on a 25% minimum wall thickness, plus a 15% allowance for corrosion plus a 10% allowance for ECT measurement, the licensee proposed a 50% plugging limit until the next inservice inspection at mid-cycle (9 months of power operation). The bases for establishing this 50% plugging limit meet the criteria specified in Regulatory Guide 1.121. "Bases for Plugging Degraded PWR Steam Generator Tubes" and will provide the same margin of safety as the 40% plugging limit established for other types of tube degradation.

2.1.6 Burst Test Data

A pitted tube (R22C45) removed from Indian Point 3 steam generator No. 31 during the fall 1981 inspection, having a measured pit depth of approximately 65% and a pit diameter of approximately 0.1 inch, was pressurized to 10,000 psi with slight bulging but no rupture and no leakage. This strength is comparable to a virgin (non-pitted) tube.

Based on burst test data obtained on the tubes with artificially induced pits and submitted to the NRC on November 6, 1981, (IPN-81-85) 0.3 inch diameter pits with remaining wall thicknesses of 25% (0.0125 inch) will withstand pressures in excess of 6000 psig. This is well above three times normal operating pressure differential between primary to secondary side (1480 psi X 3 = 4440 psi).

In addition, the result of recent tube burst tests of similar tubing with artificially induced pits submitted to NRC by Millstone Unit 2 (Docket No. 50-336) on March 1, 1982 for a myriad of test conditions indicated that the worst case (four axially aligned pits: .125" dia., separated by .01 inch ligaments) burst pressures for deepest pits (75% and 88% of tube wall degraded) were approximately 5200 psi and 4300 psi, respectively. These pressures were greater than and slightly less than three times the normal operating pressure differential (1480 psi X 3 = 4440 psi). Based on these tests, a minimum wall thickness of 0.0125 inch for pitted tubes is acceptable.

2.1.7 Corrosion Allowance

To determine the growth rate of pitting over the 3.5 month period between the fall 1981 steam generator inspection outage and the current cycle refueling outage, a sample of 116 data points was utilized. These data points represent defects which were quantified before and after the 3.5 month operating period using the same eddy current testing technique. The average change in defect size (including defects of which the ECT indications appeared to have shrunk due to interpretation uncertainties) is an increase of 5.9% over the 3.5 month period or 1.7% per month. Assuming a 9 month operating period prior to the next inspection, a defect growth of 15.3% is calculated which is essentially the same as the corrosion allowance used in justifying the 50% plugging limit.

2.1.8 Eddy Current Allowance

A comparison of laboratory examination results with field ECT data (CE probe) of four tubes pulled in 1982 showed ECT field measurement inaccuracies of 0%, 1%, 8%, and 13% for 100%, 100%, 70%, and 60% through-wall penetration respectively. In all cases the measurement errors were toward the conservative side (field data identified larger defects than the measured lab results).

Based on the above, a 10% ECT inaccuracy allowance is considered sufficient.

2.1.9 Corrective Actions

Immediate corrective actions planned by the licensee to improve the steam generator integrity and minimize the pitting progression in the steam generators include:

- sludge lancing
- modified layup procedure
- vacuum deaeration of make-up water

The effectiveness of these corrective actions in minimizing the pitting progression will be evaluated in Section 2.1.16.

2.1.10 Sleeve Configuration and Sleeving Process Conclusion

Based upon our evaluation, we find that the proposed 50% plugging limit for pitted tubes is acceptable and that the sleeving repair method for degraded steam generator tubes to be an acceptable repair alternative to plugging. We find that the sleeving repairs produce a sleeved tube of acceptable strength and metallurgical properties, corrosion resistance, leak tightness, and inservice inspectability and that the preservice integrity of the sleeved tubes is assured by having implemented the post sleeve process examinations. These findings are subject to the condition that a mid-cycle (9 months of power operation) steam generator inspection be conducted.

We also conclude that the sleeving does not result in a decrease in safety margins, an increase in the probability of an accident, or an accident not previously analyzed. This is based upon our evaluation that the repair did not impair the structural integrity or modify the original design basis of the steam generators. Since the sleeves are fabricated from thermally treated Inconel 600 with improved resistance to stress corrosion cracking, the repair effort will restore safety margins of the degraded tubes. Therefore, this repair does not involve a significant hazards consideration.

2.1.11 Girth Weld Repair Program

Indian Point 3 is a 925 MWe pressurized water reactor. The primary coolant system has four loops, each equipped with a Westinghouse Model 44 Series steam generator (vertical u-tube design). During the refueling outage of March 1982, a leak was observed at the upper shell to transition zone girth weld of steam generator 32. Subsequent examinations of the welds of all four steam generators revealed that these particular girth welds on each of the four steam generators (31, 32, 33, and 34) were extensively cracked on the inside surface. Over 600 magnetic particle indications were found in these welds which were confirmed to be cracks by ultrasonic inspection. Other shell welds were inspected and no cracks were found. The unit has had approximately three years of effective full power operation since starting commercial operation in 1976.

The steam generator shell is constructed of SA302 Grade B material of approximately 3-1/2 inches in thickness. The closure weld had a nominal 45° included angle weld preparation and was welded from the outside surface of the vessel by the submerged arc process with a backing spacer strip. The spacer strip was then removed by back-gouging and the weld completed by welding from the inside surface with the shielded metal arc welding (SMAW) process using E8018-C3 electrode. The weld was then continuously stress relieved at 1000°F minimum for three hours/inch of thickness (12 hours total soak time).

2.1.12 Failure Description

The upper shell to transition zone weld is located just below the feedwater ring in the normal operating water level zone where it is subject to thermal cycling. The crack locations had no obvious relationship to the feedwater rings. The cracks were in the circumferential direction parallel to the direction of welding. There is no relationship between original shop weld repair areas and the crack locations except the leak which occurred in an original weld repair area.

The cracks appeared to be predominantly in the weld, although the cracks were in a wide band around the assumed centerline of the weld, indicating the possibility that some cracks were located in the base metal or its heat affected zone (HAZ). Metallurgical boat samples were removed from steam generators 31 and 32 and a six inch diameter plug containing the leak path in steam generator 32 was removed for failure analysis and metallurgical examination.

2.1.13 Failure Analysis

The licensee performed metallographical evaluations and failure analyses. In the plug containing the leak path, a flaw was found which was characterized as hot cracking of a massive weld repair made during original fabrication. Image enhancement of the original production radiographs was able to detect this flaw. The leak path intersected this flaw. However, the flaw did not seem to change the direction of the inside surface crack which eventually penetrated the shell wall. In summary, a flaw on the inside surface of the vessel grew in size until it intersected a flaw which existed since original steam generator fabrication. The flaw continued to grow in size until it became a through wall crack. Steam erosion had occurred on the leak path walls and, therefore, it was not possible for metallographical examinations to characterize the surfaces for identification of failure mechanisms. Massive elemental copper deposits were present on the surfaces of the leak path. The initiating inside surface crack intersected at an angle of approximately 45° with the weld centerline and rotated to a vertical plane (perpendicular to the weld centerline) at mid thickness and propagated to the outside surface in this orientation. All other cracks were on the inside surface of the girth weld area, parallel to the weld centerline. Metallographic examinations of the boat and plug samples showed the cracking to be transgranular, with slight branching.

It is the staff's position that the pits served as the stress concentrators from which the cracks were initiated. Cracks did not form without pits being present. Pitting occurred extensively on the inside surfaces of the steam generators in the girth weld areas. Pits were found with and without cracks. However, no cracks were found independent of pits.

Cracks determined to be the farthest from the centerline of the weld were polished and etched to determine if they were actually present in the base metal and/or their HAZ's. Such a determination could not be made due to the wandering nature of the girth weld (a sinusoidal pattern reflecting fabrication fitup tolerances) and due to weld filling and blending to a smooth transition on the inside surfaces of the steam generators. However, the vast majority of cracks were initiated in weld metal. The characteristics of a crack, being transgranular, slightly branched and filled with oxides did not change as a crack went from weld metal, across a HAZ and into base metal.

Extensive efforts were made to map all flaws in the steam generator girth welds and to correlate magnetic particle indications with ultrasonic indications. This was difficult because: (1) the girth weld has a slight included angle, (2) the differences in thickness between the steam generator shell transition zone and upper cylinder results in varying girth weld angle relationships, and (3) the varying amount of weld filler metal used to blend the inside transition radius. In the majority of situations correlations were attained.

Selected sections of the original production radiographs were enhanced; e.g., the leak area and other areas of original fabrication with large weld repairs. In the plug area, the enhanced radiographs revealed flaws, one of which intersected the leak path.

The adequacy of the original post weld heat treatment was also investigated. Hardness traverses of the heat affected zone (HAZ) were reviewed. These traverses of the steam generator weldments had high peak hardnesses, up to Rockwell C42. The hardness values observed in the HAZ were higher than anticipated for welds in SA302 grade B (manganese molybdenum low alloy steel) which had been tempered by a post weld heat treatment (PWHT). There is, however, little baseline HAZ microhardness weld traverse data for comparison. Tests were conducted to determine if these high hardnesses could be taken as an indication that PWHT had not been performed or inadequately performed during original fabrication. In one set of data, there was no reduction in hardness at a PWHT temperature of 1000°F. Another set of data showed a slight reduction in the peak hardness after PWHT temperatures of 1000°F. The records at Westinghouse Tampa Division, where Indian Point 3 steam generator girth welds were made, showed a postweld heat treatment had occurred which met the requirements of the code. The laboratory test data can be characterized as showing that the peak HAZ hardnesses measured were not unusual and that a 1000°F heat treatment temperature is the minimum temperature effective in softening a heat affected zone in SA-302 Grade B. Based on this it is the staff's position that the steam generators were originally built to code and that the original "as-built" design and fabrication was not the cause of the extensive cracking.

Metallographic examinations of crack surfaces, other than the leak path, could not positively determine the failure mechanism because low alloy steels do not always produce consistent crack surface markings which can be associated with a given crack mechanism. Therefore, fatigue and stress corrosion were investigated in an attempt to define the failure mechanism. There was some evidence of fatigue, such as beachmarks, rays, and semi-circular shadings of the crack surfaces centered around a pit. However, these indications can also be caused by stress corrosion cracking where the environment is a major factor. The environment also caused pitting of the Inconel 600 tubes, indicating unusual conditions.

The licensee (Lucius Pitkin Laboratory's Technical Report No. 7164) characterized the cause of cracking as a combination of corrosion and fatigue. The staff also had an independent laboratory evaluation performed by Brookhaven National Laboratory and its conclusions were basically the same (NUREG/CR-3281, BNL-NUREG-51670). In sum, it is the staff's conclusion that the combination of the stress concentration effect of the angular joint configuration, the location in the vicinity of the normal operating water level, and the residual stress level remaining after PWHT combined with corrodant species, causes the failure. It should be noted that other longitudinal and circumferential butt welds showed no evidence of cracking.

2.1.14 Repair

All cracked weld areas were removed by grinding and rewelded using weld procedures qualified in accordance with Section IX of the ASME boiler and Pressure Vessel Code (ASME Code).

The girth weld repairs were conducted in close, confined areas. Water shielding of the steam generator tube bundle was necessary to reduce radiation. This water shielding combined with high welding preheat temperature causes temperature-humidity conditions which are unacceptable for welding with E8018-C3 electrodes which are normally used for this material. Therefore, the E7018 electrode was then selected to improve welding conditions to make repair possible by reducing preheat temperatures. This electrode also has less tendency for hydrogen cracking than the E8018-C3 electrode. To meet strength requirements, selected lots of E7018 were qualified for higher than normal strengths and the weld procedure qualification was tested over a range of PWHT temperatures to determine the maximum temperature that could be used. The PWHT temperature of 1150°F was specified based upon that being the maximum temperature at which the E7018 weld metal still retained a minimum of 80,000 psi tensile strength. After welding, the welds were inspected by radiography and liquid penetrant techniques prior to post weld heat treatment and all indications removed. The PWHT temperature was maximized because the licensee believes that an adequate stress relief was not accomplished by the PWHT of original fabrication and was a major factor in causing the steam generator girth welds to crack.

As mentioned above, the licensee was concerned about reducing residual stresses as much as possible by maintaining high post weld heat treatment temperature. The weld repairs in steam generator 33 were inspected after PWHT by the liquid penetrant technique. Many minor linear indications were found with a maximum length of 1 inch and depth of 0.375 inches. This crack depth is approximately 12.5% of the steam generator shell wall thickness and this exceeds the 3.5% allowed by ASME Code, Section XI, Table IWB 3510, for Class I components. Section XI does not provide acceptance standards and re-examination requirements for Class 2 components and recommends that the Class 1 requirements be used. The acceptance standards and re-examination requirements for Class 1 components were used in this analysis. All indications were removed by grinding and re-inspected by liquid penetrant. Steam generator 34 was liquid penetrant inspected after PWHT and a large number of small indications were found. The entire general weld area was ground approximately 1/16 inch in depth and then again dye penetrant inspected. One hundred and thirteen flaws remained. After grinding, the longest indication found was 0.625 inches. Eight others were more than 0.25 inch and the balance, 104, were 0.25 inch or less in length. In order to reduce radiation exposure of workers by 22 person-rem the licensee requested: (1) that the liquid penetrant inspection of the two remaining steam generators (31 and 32) not be performed, and (2) that the remaining linear indications (cracks) in steam generator 34 be allowed to remain as is. The licensee provided a fracture mechanics analysis to justify this request.

To justify the acceptability of leaving these flaws in place, the licensee submitted a fracture mechanics analysis, "Fracture Mechanics Analysis of the Indian Point Unit 3 Nuclear Generating Station Steam Generator Girth Weld Cracks Under Pressurized Thermal Shock Conditions," prepared by Fracture Proof Design Corporation. The analysis in this report is based on several cooldown transients while the pressure is maintained at the design pressure of 1085 psig. These water temperature transients are similar to those used by the NRC staff in its consideration of pressurized thermal shock scenarios involving reactor vessels. The most severe thermal transient, or the worst case scenario assumed, was a cooldown from the normal operating temperature to 70°F in about three minutes. The staff does not consider this to be a credible event because: (1) the cooldown rate is more rapid than would reasonably be expected, and (2) maintenance of a high steam generator pressure during a cooldown is extremely unlikely unless the steam generator were isolated and under water-solid conditions (in this case a rapid cooldown is much less likely). However, this scenario can be used in the fracture mechanics analysis because it is a conservative worst case scenario which will result in an upper bound of stresses in the steam generator wall. The report submitted by the licensee includes linear elastic and elastic-plastic fracture mechanics analyses of assumed continuous circumferential pre-existing cracks in the steam generator wall. The elastic-plastic approach is necessary only for very deep cracks, and therefore, was considered not necessary in our evaluation.

In lieu of evaluating the fracture mechanics portion of the licensee's analysis, the staff performed independent analyses for various assumed 360° circumferential cracks up to 45 percent through-wall (approximately 1.6 inches deep). For this range of crack depths linear-elastic fracture mechanics is considered adequate. In the analyses, it is conservatively assumed that the phenomenon of warm prestressing is not effective. For the worst case scenario, warm prestressing would occur at about two minutes and would prevent initiation at later times into the transient. Disregarding warm prestressing, the closest approach to crack initiation would be between three and four minutes for cracks about 0.25 inch deep. Even if a shallow crack were to initiate, that is, grow deeper into the walls, the staff's analysis predicts that it would arrest at less than 40 percent through the wall. Under severe thermal shock conditions such as this assumed scenario, shallow pre-existing cracks are more likely to initiate than deeper cracks because of the higher thermal stresses and lower material toughness near the cooled surface. Thus, a surface crack 0.25 inch deep is more likely to initiate than a 0.375 inch deep crack in the pressurized thermal shock analysis. Therefore, the catastrophic failure of the steam generator under these very severe conditions is precluded even if the vessel had pre-existing complete circumferential cracks approaching half the wall thickness in depth and that operation with small cracks is not expected to jeopardize the ultimate integrity of the steam generator shell. Therefore, assuming the same size cracks in steam generators 31 and 32, these steam generators need not be further inspected before plant restart. The staff also concludes that the repair program and test results are acceptable.

A review of fabrication records showed that the steam generators at Indian Point 3 had no significant differences from other steam generators fabricated during the same time frame. This particular weld is included in normal inservice inspection programs at all other PWR's in accordance with 10 CFR 50.55(a)(g), and there has been no record of extensive cracks at any other plant as were found at Indian Point 3. Therefore, it is the staff's position that the cause of the cracking found in March 1982 is due to a corrosive environment unique to this plant. Likewise it is the staff's position that if IP3 is operated with secondary water chemistry limits specified in the water chemistry monitoring and control program, minor flaws will not grow. This conclusion is based upon experience with other operating plants. Large, complex weldments always have some small flaws, and there has not been any experience to date of operating plants developing cracks as occurred at Indian Point 3. To provide additional assurance, the licensee has already performed a UT base-line test after PWHT and hydro-test and has proposed augmented inservice inspection of the same portions of the steam generators to monitor for possible flaw growth. The licensee proposes to ultrasonically inspect the same 35 inches of the girth weld on each steam generator during cycle 4 as well as during refueling outages. The areas chosen are generally where cracks had developed in the past and where some of the largest weld repairs were made. The area in steam generator 34 includes known flaw sites detected after PWHT. An additional 35 inches is to be monitored on steam generator 32 in the area where the plug with the leak path was removed and another plug was welded in to fill the hole. This totals 175 inches of girth weld in the four steam generators in the augmented inservice inspection. After review of the licensee's proposal, the staff determined that the frequency of the augmented inspections was adequate but that the initial inspection of the 175 inches be performed after approximately 9 months of power operation. These augmented inspections will provide flaw growth trend information and are acceptable alternatives to removing the small flaws with the attendant person-rem of exposure. With base line UT results to use for comparison, the midcycle and subsequent periodic UT inspections can be effective in identification of small cracks. This testing will also result in early identification of failure reoccurrence and facilitate the determination of prompt corrective action. As such, it is considered an acceptable compensatory measure.

2.1.15 Girth Weld Repair Conclusion

We conclude that the licensee's repair program is acceptable. The basis for this conclusion is as follows:

1. The repair program meets the fabrication requirements for the steam generators except for the very small remaining flaws. The augmented inspection program provides an acceptable alternative to removal of these cracks.

2. The radiographic examination (RT) of the weld joint prior to PWHT was more sensitive than the original radiography and covered more than twice the area adjacent to the weld. Ultrasonic testing (UT) also was performed. RT results combined with UT results provides reasonable assurance that the flaws in the girth welds are very small and are significantly smaller than the largest flaw in the original fabrication.

3. The flaws present have been shown to be insignificant by fracture mechanicals analysis; i.e., cracks of of a maximum size found after PWHT in steam generator 33 would not impair the integrity of the steam generators assuming a worst case cooldown transient.

4. The residual stresses in the weld area are lower than those in the original fabrication due to the higher PWHT temperatures.

5. The girth welds in the steam generators were liquid penetrant inspected and radioagraphed prior to PWHT. These tests did not reveal flaws. After PWHT, flaws were detected by liquid penetrant testing. Since PWHT does not create weld defects such as those found in steam generators 31 and 32, these flaws existed prior to PWHT. PWHT causes an oxide layer to be built up on flaws which allows liquid penetrant detection. Flaws in steam generators 31 and 32 are not expected to be significantly different from those found in the girth welds of #33 and #34 steam generators.

6. Augmented inservice inspections have been committed to by the licensee. The surveillance inspection areas chosen include sections of steam generator 34 with known liquid penetrant indications on the inside surface, the plug repair area of steam generator 32 and areas in steam generator 31 and 33 which had a high density of defects or have the wider and deeper weld repairs. The augmented inservice inspection by ultrasonic methods of the upper shell to transition cone girth weld in the four steam generators will provide additional assurance of adequate steam generator shell integrity.

7. Flaw growth has been predominantly attributed to corrosion. In order to improve secondary plant water chemistry the licensee has agreed to both short and long term measures as discussed in section 2.1.16. Based on this consideration, there is no reason to expect growth of minor flaws to unacceptable depths. Further assurance is provided by the mid cycle testing as described in item 6 above.

2.1.16 Corrosion and Secondary Water Chemistry Monitoring and Control

2.1.16.1 Introduction

During routine steam generator inspections in the fall of 1981, large numbers of tubes were identified with pitting defects greater than 40% through-wall. Many tubes had pits ranging in depth up to 65% through-wall. After staff evaluation, the plant was permitted continued operation.

While the plant was shut down for refueling in the spring of 1982, a leak was observed in the shell of one steam generator upper transition cone girth weld. Subsequent examinations of these welds on all four steam generators revealed that each generator had extensive indications of cracking. The repair of the cracks in the girth weld of the steam generators required an extensive period of plant downtime. Sleeving repair of pitted steam generator tubes was initiated at this time.

To assist in the review of the adequacy of these two repair programs, an evaluation was made of the environmental conditions within the steam generators during previous operations and the anticipated operating conditions after the repairs. The licensee provided data on the chemical parameters during previous operations and also provided a description of their ongoing program of modifying and upgrading of the instruments, equipment and components in the secondary water cycle.

2.1.16.2 Evaluation

By letter dated October 18 and 25, 1982; November 17, 1982; January 11 and 19, 1983 and May 3, 1983 the licensee described the proposed girth weld and tube sleeving repair methods and secondary water chemistry program.

This plant has a long history of condenser leakage problems resulting in a small continuing in-leaking of impurities even when major condenser leaks had not been identified. These inleakage occurences at the condenser have contributed to steam generator tube denting. The condenser and feedwater heater tubes are made of copper alloys, which have been corroded so that the sludge analysis in the steam generators shows concentrations of copper in excess of 45% with copper oxide as a major constituent. The presence of this and other constituents in the sludge indicates that oxygen control in the feedwater/steam generator train has been poor for a considerable length of time.

The licensee had been minimizing the amount of hydrazine present in their steam generators to minimize the potential for hydrazine discharges to the river water. The reduced use of hydrazine contributed to the corrosiveness of the steam generator environment. The influx of copper ions through the condenser and the feedwater train has also caused copper deposition on the Inconel tubing in an area roughly at the boiling line on the cold leg side of the steam generators. When the power level of the unit was reduced, the boiling line on the cold leg of the steam generators would have risen, as a result of the lower amount of steam being generated by the unit. Consequently, the areas on the tubes where the copper deposits were located were then submersed in all liquid phase in the cold leg. Coupling this with inleakage of chlorides, and the crevices that exist on the surface of the tubes between the Inconel and the copper deposits and on the steam generator shell, produced a site on the Inconel tubes and carbon steel shell for the pitting to initate. Once a pit has initiated in the presence of chloride ions, the pit will continue to grow so long as there is a supply of oxygen or an oxidizing environment, until the environment within the pit itself is flushed out.

The licensee had been implementing a boric acid treatment of feedwater injection since mid '79 in an attempt to reduce the rate of denting. After the September 1981 outage, this program was discontinued. Although denting has not been noted as a factor in the repair program, and the addition of boric acid is not believed to be a factor in the pitting corrosion, this treatment has been terminated pending further evaluation.

The licensee has had metallurgical evaluations of samples removed from both the degraded girth weld and the pitted steam generator tubes. We have reviewed these analyses and agree with the conclusions. In both cases, the metallurgical evaluation indicated that the pitting and degradation were caused, in part, by the environmental conditions in the steam generator. Pitting occurred extensively on the inside surfaces of the steam generators in the girth weld areas. Pits were found with and without cracks. However, no cracks were found independent of pits. There were a few cases where a small crack was found at the bottom of a pit and the crack tips lay entirely within the pit boundary.

Tube were removed from the steam generators and failure analyses made in the area of the copper deposits, which were in rings around the tubing. These showed bands of pits ranging in size from a few mils to 100 mils in diameter and ranging in depth up to 65% of the wall (32 mils).

We are also of the opinion that the secondary water chemistry control was not very stringent and the continued ingress of oxygen and chlorides contributed to the pitting in both the SG tubes and the shell and to propagation of cracking in the girth weld.

The licensee has estimated the growth rate of pitting over the 3.5 month operating period between the fall of 1981 steam generator inspection and the current outage inspection. A sample of 116 data point was used to quantify the change in defect size. A growth of 1.7% per month in defect size was calculated. The sleeving repair program was initiated to extend the operating life of the steam generator tubes.

The sleeving concept and design are based on observations to date that the tube degradation due to pitting attack has occurred on the cold leg of the tube bundle, confined to an area within approximately two feet above the tubesheet.

To provide an evaluation of the corrosion aspects of the sleeving repair program, we have reviewed the Westinghouse Report WCAP-10146 (Proprietary), Revision 1, dated September 1982, and entitled "Indian Point 3 Steam Generator Sleeving Report Prepared for Power Authority of the State of New York." We have reviewed the corrosion test program performed in support of sleeving repair programs referenced in this document.

As part of the test program, the behavior of the repair program materials was studied in pure water, in primary coolant, and in 10% caustic solutions, to simulate the continued hideout of caustic in the crevices and sludge on the secondary side of the steam generators. This work has shown that the thermal treatment to be given to the Inconel sleeves is effective in reducing the probability of caustic stress corrosion developing on these sleeves. It has also been shown that the small, controlled amount of cold work performed on the Inconel in attaching the sleeve to the S.G. tube was not sufficient to cause a significant increase in the suceptibility of the tube to stress corrosion cracking from the primary side water. This amount of cold work is significantly less than that which occurred where the tube was expanded into the lower portion of the tubesheet during the original fabrication. To date no cracking has developed in that area in Point Beach, San Onofre, or in any steam generator in the U.S. of similar design to those at Indian Point 3.

We have examined the results of model boiler tests in which heat treated Inconel 600 tubes were hydraulically expanded into simulated tubesheets. These tube/tube sheet models were exposed to severe caustic corrosive media, for accelerated time testing at steam generator hot leg operating temperatures. While steam generators do not operate in severe caustic environments, this environment provides a reasonable accelerated test time for determining susceptability to caustic corrosive degradation.

Extended test times in this environment did not produce corrosive attack upon the Inconel 600 tubes that had been thermally treated and hydraulically expanded into a simulated tube sheet. Based on these data, there is reasonable assurances that the sleeve material will be equal to or more corrosion resistant to Indian Point 3 environment than the original tubes. We find that the sleeving techniques and the material in the sleeves are acceptable from a corrosion resistance aspect.

By letters of January 11, 1983 and May 3, 1983, the licensee provided information on the secondary water chemistry monitoring and control program. We have reviewed the steam generators past water chemistry history and have evaluated the present monitoring and control program using the guidance provided in our April 21, 1983 letter.

The licensee has had an ongoing program of improving water chemistry monitoring and control. Over many years the plant has added sampling points and instruments.

They have also had a program of upgrading and modifying components of the secondary water cycle. The licensee has had a program of locating and repairing condenser leaks of cooling water and air. The sludge lancing programs performed during this outage has removed sludge with large amounts of reducible metal oxides such as Cu_2O, in an attempt to further control the availability of corrosion assisting elements. The lay up procedures should protect the SG from degradation during layup after the completion of the repairs. These procedures have been modified to mitigate the presence of oxygen in the SG. To improve the quality of the makeup water, the plant has installed a 160 gpm demineralizer and a vacuum degasifier.

The licensee has installed over a period of time, increasingly sophisticated leak detection instruments. Concurrent with the increased ability to detect condenser leakage, a program of repairing major leakage paths has been instituted.

The licensee's water chemistry monitoring and control program includes a clearly defined chain of authority and responsibility for analysis, interpretation, and corrective actions for secondary water chemistry control including action levels for power reduction. The authority responsible for interpreting secondary side water chemistry data is the Water Chemist who reports to the Chemistry Supervisor and the Shift Supervisor. Final responsibility for any course of action lies with the Superintendent of Power or his designated alternate, including authorization of certain deviations from normal practices. Specific water chemistry limits are defined for the condensate, feedwater, and steam generator blowdown, including sampling point locations and sampling schedules. These limits cover varying plant conditions, including normal power operation, cold shutdown, power operation following startup, and dry or wet layup. Procedures are outlined, and details mentioned by reference, which define corrective actions to be taken in response to out-of-specification conditions. Daily chemistry log sheets are reviewed systematically to help determine trends.

We anticipate that the improvements in the chemistry program and component modifications will further improve the chemistry control. However, the chemistry program parameter limits and the action levels for correction are less restrictive than our recommendations or those by the NSSS vendor and the Steam Generator Owners Group (SGOG). Significant improvements in water chemistry; i.e. the use of more restrictive parameter limits and more responsive action levels, are not anticipated until major components are modified or replaced.

Based on the available information pertaining to the steam generator water chemistry control, it is the staff position that pitting, stress corrosion cracking, and material degradation of the girth weld has been continuous since early in plant operations. The unit has had approximately 36 months of effective full power operation since starting commercial operation in 1976. Assuming that crack propagation occurred at a uniform rate during hot operation, a pre-repair propagation rate of 0.083-inch/month is postulated (3" thick/36 months). Stress corrosion cracking is influenced primarily by stress and environment. The post repair crack propagation rate is anticipated to be significantly slower due to the reduced residual stresses in the girth weld as a consequence of post repair heat treatment and the water chemistry program which is at least as restrictive as the prior programs.

Based on the reduced residual stresses, we have reasonable assurance that if SCC continues in the girth weld during the 9 months prior to the next inspection it will penetrate to a depth significantly less than 0.075 inches, as predicted by the pre-repair corrosion rate.

Assuming a worse case situation, the girth weld crack could have propagated through wall during the 18-month operating period since the major chloride intrusion when the turbine blade throw incident ruptured a tube in the condenser. This assumption gives a crack propagation rate of 0.17 in/month. As discussed above, the post-repair rate is expected to be less than before the repair program. Therefore, even in the worst case assumption, crack propagation should be significantly less than 50% through wall during the upcoming 9 months operating cycle. Therefore, there is reasonable assurance that the public health and safety will not be endangered.

2.1.16.3 Corrosion and Secondary Water Chemistry Conclusion

Based on the above evaluation, we conclude that:

1. The corrosion test programs performed in support of the sleeving operations are adequate to provide reasonable assurance that the sleeving process will not induce accelerated attack on the tube itself and that the sleeving material is more resistant to stress cracking than the original tubing;

2. The licensee's secondary water chemistry monitoring and control program parameter limits and action levels for corrective action are less restrictive than our recommendations of those of the NSSS vendor and the SGOG. However, it has the capability of reducing the observed rate of SG degradation for plant operations during the upcoming fuel cycle by reducing the availability of oxygen;

3. The expected improved secondary water chemistry control during the upcoming operating period will, it is anticipated, reduce the rate of pit growth and girth weld crack growth sufficiently to provide reasonable assurance that public health and safety will not be endangered.

Based upon the above, the staff concludes that the secondary water chemistry and control program is acceptable for operation during the upcoming fuel cycle.

2.2 Reactor Physics

2.2.1 Introduction

Reactor physics is unaffected by steam generator girth weld repairs. Therefore, the following evaluation addresses steam generator tube sleeving/plugging and the effect on reactor physics.

Sleeves inserted into steam generator tubes create a higher pressure drop and consequently decrease flow in the reactor cooling system (RCS). Since this is an adverse affect on core cooling, the operation of the reactor has to be reanalyzed to ensure that excessively high temperatures, which could damage the reactor, will not be obtained during normal operating, transient, or accident conditions.

The nominal thickness of the Inconel 600 sleeves, which are to be inserted into the .775" I.D. steam generator tubes, is .039". Putting this sleeve into a tube will reduce the flow area about 27 percent. Westinghouse in its report on the sleeving of the steam generator tubes at Indian Point Unit 3 (Reference 1) states, however, that the equivalent loss in flow is only 5 percent of the normal flow through a tube. This is due to the increase in velocity of the flow through a sleeved tube. With a 5 percent loss in flow due to a sleeve, 20 tubes can be sleeved before the flow through a steam generator is reduced the same amount as by one fully plugged tube.

Of the four steam generators at Indian Point Unit 3, No. 3 has the greater number of degraded tubes. Two hundred and eighty five of its 3,260 tubes have already been plugged. Eddy current tests have shown that 24 more need to be plugged. Also an additional 998 tubes have reached the plugging limit (i.e., 40% wall degradation) and the licensee has requested sleeving these tubes. Assuming a 5 percent reduction in each tube's flow when sleeved, this would be equivalent to 50 fully plugged tubes. Thus the total equivalent number of plugged tubes in steam generator No. 31 will be 359, which is 11 percent of the 3,260 tubes.

Westinghouse performed tests under simulated steam generator conditions to determine the leak rates through the sleeved tube joints (Reference 2). These tests showed that the anticipated leak rate through the joints of 1494 sleeved tubes per steam generator is only a small fraction of the Technical Specification limit, which is 13.7 GPM for the plant, or an average of 3.425 GPM per steam generator, and has an insignificant effect on core cooling.

2.2.2 Evaluation

As stated in Refernece 3, Westinghouse performed a safety study, which evaluated the effects of the reduced RCS flow through steam generators due to 12 percent of the tubes being plugged in each of the four steam generators. This safety study was submitted to the NRC in support of license amendment No. 40. In NRC's evaluation (Reference 4) of this study it was stated that:

> "The licensee has analyzed the proposed increase in steam generator tube plugging with respect to transients and accidents analyzed in the Safety Analysis Report. They have concluded that the incorporation of these modifications: a) will not change the probability nor the consequences of an accident or malfunction of equipment important to safety as previously evaluated in the Safety Analysis Report; b) will not increase the possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report; and c) will not reduce the margin for the safety as defined in the basis for any Technical Specification."

2.2.3 Conclusion

It is concluded that the plugging of no more than 309 tubes along with the sleeving of no more than 1640 tubes or any combination equivalent to the plugging of 391 tubes in any of the four steam generators at Indian Point 3 will not reduce the RCS flow more than the reduction which the NRC approved for license Amendment No. 40. On this basis we conclude that this reduction in steam generator flow will not result in a significant increase in the probability or consequences of accidents previously considered and does not involve a significant decrease in safety margin. Therefore, the reduction in steam generator flow which will be caused by the sleeving and plugging proposed in Reference 1 is approved.

2.3 Worker Dose Mitigation Program

2.3.1 Evaluation

The Power Authority of the State of New York (PASNY) took into account ALARA considerations for each of the activities involved in the full-scale steam generator sleeving program and steam generator girth weld repair at Indian Point Unit 3 (IP-3). ALARA activities specifically directed to reduction of occupational radiation doses included: decontamination of steam generators; installation of shielding as appropriate to reduce radiation exposures to repair personnel; remote control of the sleeving processes and personnel training in full-size mock-ups. PASNY verified that the training program was in accordance with Regulatory Guide 8.27, 8.29 and 8.13 or equivalent. In addition, welders, grinders and girth weld workers received training on the girth weld process and potential problems to familiarize themselves with the job. All personnel assigned to the project received special offsite training at a Westinghouse and a Peakskill Training facility utilizing full scale mock-up sleeving equipment.

Administrative control of personnel exposures were effected by planning of maintenance procedures for the job, in order to minimize the number of personnel used to perform the various tasks involving relatively high doses and dose rates. Nozzle cover shielding was used to reduce doses to workers on or near the nozzle cover. TV surveillance of personnel during tasks were used to identify areas and activities involving high exposures and thus to initiate suitable dose-reducing actions.

PASNY described the provisions for special local ventilation in the steam generator repair area. Each steam generator was ventilated through the hot leg manway for cold side work. This maintained a negative pressure in the working manway to prevent airborne radioactivity on the steam generator platform. Each steam generator was ventilated by providing suction and supply via the secondary side manways and flexible ducting.

The major source of radiation dose rate inside the steam generator channel head was a tenacious layer of "oxide" which included deposited activated corrosion products. In order to remove this deposited activity from the inside of the channel head and thereby reduce dose rates in this region, PASNY used a Westinghouse mechanical decontamination process involving a slurry compound in a high pressure water spray. A manipulatory arm inside the channel head with jet nozzles was operated remotely from a low dose rate area.

PASNY made use of the experience gained in prior channel head decontamination in planning for the proposed tube sleeving activities. Data were available for Point Beach (Unit 1), Takahama (Unit 1), San Onofre (Unit 1), and Turkey Point (Unit 3). In particular, PASNY considered information on mechanisms used in prior decontaminations, and provided information relevant to projected occupational radiation exposures.

Based on experience from sleeving projects at other plants, PASNY had estimated an average dose of 217 man-rems for sleeving each steam generator or a total of 868 man-rems for completing the IP-3 sleeve installation. An additional 371 man-rems was estimated to be expended for the steam generator girth weld program. The collective dose of 1240 man-rems includes all occupational dose resulting from the sleeving operation and girth weld operations including all site and contractor support personnel. A breakdown of each task by estimated dose rates, man-hours, and man-rems has been provided.

The tasks of steam generator sleeving and the steam generator girth weld repair have now been completed. A total dose of 737 man-rems were expended for completing the IP-3 sleeve installation and a total of 387 man-rems were expended for the steam generator girth weld program. This resulted in a collective dose of 1124 man-rems, substantively lower than the estimated dose.

By letter dated October 18, 1982, as supplemented by letters dated January 19, 1983, May 2, 1983 and May 3, 1983, PASNY committed as part of the technical specification change request to conduct a mid-cycle inspection of both steam generator tubes and girth welds during fuel cycle 4. The applicant has committed to submitting his testing program 30 days prior to implementation, for NRC approval. This program will be reviewed by the staff to ensure that these doses are within the guidelines of Regulatory Guide 8.8.

2.3.2 Conclusion

The Indian Point Report identified the programs necessary to maintain personnel doses ALARA, consistent with the guidelines of Regulatory Guide 8.8. The collective dose of 1124 man-rems expended in the completion of the tasks verified that the IP3 program as proposed and implemented was conservative.

3.0 CONCLUSION

3.1 No Significant Hazards Consideration Determination

In summary, the staff concludes that as a result of the weld repairs to the shell all but very small indications have been repaired and these have been demonstrated by fracture mechanics analyses to be insignificant with respect to structural integrity of the steam generators. Adequate surveillance of the girth weld will assure that any crack growth will be detected before it can become significant. Moreover, the reduction in residual stress in the girth weld area will reduce the tendency toward cracking in this area. On this basis we conclude that the structural integrity of the steam generator shell has been restored and will be retained over the next operating cycle, and that monitoring programs will provide early identification of potential degradation well before it can have a significant effect on structural integrity.

For the steam generator tubes the staff concludes that the sleeving process is a well developed known process which produces sleeves that restore the structural integrity of the sleeved tube as a primary pressure boundary. Plugging limits for pitted tubes have been established to provide the same margin as that provided by the existing Technical Specifications for other types of tube degradation. Moreover, burst test data demonstrates that such tubes still have a high margin against bursting. In addition, increased eddy current surveillance including midcycle testing will assure that tube integrity will remain adequate. Accordingly, the staff concludes that the structural integrity of the steam generator tubes, as a primary system boundary, has been restored to the original design basis.

After the completion of steam generator girth weld repairs welds were ultrasonically (UT) inspected. These same welds will be UT tested after about 9 months of plant operation as well as during the next refueling outage. The results of completed tests in conjunction with the midcycle test provide trend information which allows early identification of any degradation. In addition, there will be a midcycle test of the steam generator tubes as well as a steam generator tube inspection during the next refueling outage. These tests will likewise provide early indication of degradation. The licensee's commitment to perform these tests and to provide timely documentation to the NRC will allow the determination of prompt corrective action, if appropriate.

The steam generator repair program has adequately restored the structural integrity of both the tubes and shell to the original design basis. Therefore, this amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated, does not create the possibility of an accident of a type different from any evaluated previously and does not involve a significant reduction in a margin of safety. On this basis, the NRC staff concludes that this amendment does not involve a significant hazards consideration.

3.2 Environmental Consideration

We have determined that the amendments do not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendments involve an action which is insignificant from the standpoint of environmental impact and, pursuant to 10 CFR §51.5(d)(4), that an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of the amendments.

3.3 Conclusion

We have concluded, based on the considerations discussed above, that: there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

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

ENVIRONMENTAL IMPACT APPRAISAL BY THE OFFICE OF NUCLEAR REACTOR REGULATION

POWER AUTHORITY OF THE STATE OF NEW YORK

INDIAN POINT NUCLEAR GENERATING PLANT, UNIT NO. 3

STEAM GENERATOR SLEEVING/PLUGGING

AND GIRTH WELD REPAIR PROGRAMS

DOCKET NO. 50-286

1.0 Introduction

During 1982 two major steam generator repair programs were undertaken at the Indian Point Nuclear Generating Plant, Unit No. 3 (IP-3). These two programs consisted of sleeving, and, if necessary, plugging steam generator tubes and the removal and replacement of steam generator secondary side upper girth welds. This Environmental Impact Appraisal evaluates the significance of the occupational exposure incurred during the now completed repair work.

2.0 Radiological Assessment

2.1 Environmental Significance of Occupational Exposure

By letter dated January 19, 1983, the Power Authority of the State of New York (licensee) has estimated that the occupational exposure from the proposed steam generator repair at Indian Point Nuclear Generating Plant, Unit No. 3 will be about 1240 person-rems per reactor unit*. Based on the staff's review of the licensee's report, the staff concludes that the licensee's estimate of 1240 person-rems to the workforce is a reasonable estimate of the expected dose.

To determine the relative environmental significance of the estimated occupational dose for the repair, the staff has compared this dose for the repair with the reported doses experienced at modern pressurized water reactors (PWRs). In addition, the staff has also compared the estimated risk to nuclear power plant workers to published risks for other occupations.

Most of the doses to nuclear plant workers result from external exposure to radiation emitted by radioactive materials outside of the body, rather than from internal exposure due to inhaled or ingested radioactive materials. Experience has shown that the total annual dose to nuclear plant workers

*Unless otherwise noted, all estimates in §2.0 of the quantities of radionuclides released, the exposure estimates, and health risk estimates are on a per reactor unit basis.

varies substantially from reactor to reactor and from year to year. Recently licensed 1000-MWe PWRs are designed in accordance with the post-1975 regulatory requirements and guidelines that place increased emphasis on maintaining occupational exposure at nuclear power plant "as low as reasonably achievable" ("ALARA"). These requirements and guidelines are outlined respectively in 10 CFR Part 20, Standard Review Plan Chapter 12 (NUREG-0800), and Regulatory Guide 8.8, "Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations Will be as Low as is Reasonably Achievable."

The licensee's proposed implementation of these requirements and guidelines for the repair work has been reviewed by the NRC staff, and the results of that review is reported in the staff's Safety Evaluation Report.

Table 1 shows the occupational dose history for Indian Point Unit 3. With the 1982 addition of 741 person-rems for steam generator repair programs, the average annual dose for the plant will increase by about 16.9% from the average of 569 person-rems (five year average) to about 665 person-rems (six year average). Furthermore, if the 1983 collective dose is estimated to be 569 person-rems (the five year average dose prior to the repair) plus 449 person-rems (the estimated dose for completing the repairs) or 1018 person-rems, then the seven year average (1972-1983) would be 716 person-rems. The seven year average annual dose would be about a 26% increase over the five year average annual dose of about 570 person-rems.

Average collective occupational dose information for 239 PWR reactor years of operation is available for those plants operating between 1974 and 1980. (The year 1974 was chosen as a starting date because the dose data for years prior to 1974 are primarily from reactors with average rated capacities below 500 MWe). These data indicate that the average reactor annual collective dose at PWRs has been about 440 person-rems, with some plants experiencing an average plant lifetime annual collective dose to date as high as 1300 person-rems (NUREG-0713, Vol. 2). These dose averages are based on widely varying yearly doses at PWRs. For example, for the period mentioned above, annual collective doses for PRWs have ranged from 18 to 5262 person-rems per reactor. However, the average annual dose per nuclear plant worker of about 0.8 rem (ibid) has not varied significantly during this period. The worker dose limit, established by 10 CFR Part 20, is 3 rems/quarter (if the average dose over the worker lifetime is being controlled to 5 rems/yr) or 1.25 rems/quarter (if it is not).

The wide range of annual collective doses experienced at PWRs in the United States results from a number of factors such as the amount of required maintenance and the amount of reactor operations and inplant surveillance. Because these factors can vary widely and unpredictably, it is impossible to determine in advance a specific year-to-year annual occupational radiation dose for a particular plant over its operating lifetime. There may on occasion be a need for relatively high (with respect to the average annual collective dose) collective occupational doses, even at plants with radiation protection programs designed to ensure that occupational radiation doses will be kept ALARA.

The average annual dose of about 0.8 rem per nuclear-plant worker at operating PWRs has been well within the limits of 10 CFR Part 20. However, for impact evaluation, the NRC staff has estimated the risk to nuclear-power-plant workers and compared it in Table 2 to published risks for other occupations. Based on these comparisons, the staff concludes that the risk to nuclear-plant workers from plant operation is comparable to the risks associated with other occupations.

Table 1. Annual collective occupational dose at Indian Point Unit 3

Year	Reported Collective occupational dose* (person-rems/reactor)
1977	535
1978	1003
1979	636
1980	308
1981	364
1982	1148**
Average ('77-'81)	569
Average ('77-'82)	665

*USNRC, "Occupational Radiation Exposure at Commerical Nuclear Power Reactors, 1981," NUREG-0713, Vol. 3, November 1982. For the years 1977 and 1978, the annual doses from Indian Point Unit 3 were combined with those from Unit 2 and were reported as a single dose (Unit 1 was defueled in 1975). For these two years, the doses shown in Table 1 for Unit 3 were obtained by dividing the reported doses by two.

**Steam Generator Repair Program	741
Other Occupational Dose	407
TOTAL DOSE	1148

Table 2 . Incidence of job-related mortalities

Occupational Group	Mortality Rates (premature deaths per 10^5 person-years)
Underground metal miners*	~1300
Uranium miners*	420
Smelter workers*	190
Mining**	61
Agriculture, forestry, and fisheries**	35
Contract construction**	33
Transportation and public utilities**	24
Nuclear-plant worker***	23
Manufacturing**	7
Wholesale and retail trade**	6
Finance, insurance, and real estate**	3
Services**	3
Total private sector**	10

*The President's Report on Occupational Safety and Health, "Report on Occupational Safety and Health by the U.S. Department of Health, Education, and Welfare," E. L. Richardson, Secretary, May 1972.

**U.S. Bureau of Labor Statistics, "Occupational Injuries and Illness in the United States by Industry, 1975," Bulletin 1981, 1978.

***The nuclear-plant workers' risk is equal to the sum of the radiation-related risk and the nonradiation-related risk. The estimated occupational risk associated with the industry-wide average radiation dose of 0.8 rem is about 11 potential premature deaths per 10^5 person-years due to cancer, based on the risk estimators described in the following text. The average non-radiation-related risk for seven U.S. electrical utilities over the period 1970-1979 is about 12 actual premature deaths per 10^5 person-years as shown in Figure 5 of the paper by R. Wilson and E. S. Koehl, "Occupational Risks of Ontario Hydro's Atomic Radiation Workers in Perspective," presented at Nuclear Radiation Risks, A Utility-Medical Dialog, sponsored by the International Institute of Safety and Health in Washington, D.C., September 22-23, 1980. (Note that the estimate of 11 radiation-related premature cancer deaths describes a potential risk rather than an observed statistic.)

In estimating the health effects resulting from occupational radiation exposures as a result of this repair, the NRC staff used somatic (cancer) and genetic risk estimators that are based on widely accepted scientific information. Specifically, the staff's estimates are based on information compiled by the National Academy of Science's Advisory Committee on the Biological Effects of Ionizing Radiation (BEIR I). The estimates of the risks to workers and the general public are based on conservative assumptions (that is, the estimates are probably higher than the actual number). The following risk estimators were used to estimate health effects: 135 potential deaths from cancer per million person-rems and 258 potential cases of all forms of genetic disorders per million person-rems. The cancer-mortality risk estimates are based on the "absolute risk" model described in BEIR I. Higher estimates can be developed by use of the "relative risk" model along with the assumption that risk prevails for the duration of life. Use of the "relative risk" model would produce risk values up to about four times greater than those used in this report. The staff regards the use of the "relative risk" model values as a reasonable upper limit of the range of uncertainty. The lower limit of the range would be zero because health effects have not been detected at doses in this dose-rate range. The number of potential nonfatal cancers would be approximately 1.5 to 2 times the number of potential fatal cancers, according to the 1980 report of the National Academy of Science's Advisory Committee in the Biological Effects of Ionizing Radiation (BEIR III).

Values for genetic risk estimators range from 60 to 1500 potential cases of all forms of genetic disorders per million person-rems (BEIR I). The value of 258 potential cases of all forms of genetic disorders is equal to the sum of the geometric means of the risk of specific genetic defects and the risk of defects with complex etiology.

The preceding values for risk estimators are consistent with the recommendations of a number of recognized radiation-protection organizations, such as the International Commission on Radiological Protection (ICRP 1977), the National Council on Radiation Protection and Measurement (NCRP 1975), the National Academy of Sciences (BEIR III), and the United Nations Scientific Committee on the Effects of Atomic Radiation (UNSCEAR 1982).

The risk of potential fatal cancers in the exposed work-force population at the Indian Point Unit 3 facility and the risk of potential genetic disorders in all future generations of this work-force population, is estimated as follows: multiplying the plant-worker-population dose (about 1240 person-rems) by the risk estimators, the staff estimates that about 0.17 cancer deaths may occur in the total exposed population and about 0.32 genetic disorders may occur in all future generations of the same exposed population. The value of 0.17 cancer deaths means that the probability of one cancer death over the lifetime of the entire work force as a result of the repair is about 1 chance in 6. The value of 0.32 genetic disorder means that the probability of one genetic disorder in all future generations of the entire work force as a result of the repair is about 1 chance in 3.

The significance of these risk estimates can be determined by comparing them to the natural incidence of cancer death and genetic abnormalities. Multiplying the estimated exposed worker population (~1240 persons assuming an average dose of 1 rem/worker) by the current incidence of actual cancer fatalities (~.20%) about 250 cancer deaths are expected due to natural causes (American Cancer Society, 1978). The risk of potential genetic disorders attributable to exposure of the workforce is a risk borne by the progeny of the entire population and is thus properly considered as part of the risk to the general public. Since BEIR III indicates that the mean persistence of the two major types of genetic disorders is about 5 generations and 10 generations, in the following analysis, the risk of potential genetic disorders from the repair is conservatively compared with the risk of actual genetic ill health in the first five generations, rather than the first ten generations. Multiplying the estimated population within 50 miles of the plant (~19,000,000 persons in the year 1980) by the current incidence of actual genetic ill health in each generation (~11%), about 10,000,000 genetic abnormalities are expected in the first five generations of the 50 mile population due to natural causes (BEIR III).

In summary, the NRC Staff has drawn the following conclusions regarding occupational radiation dose. The licensee's estimate of about 1240 person-rems/reactor for the repair at Indian Point Unit 3 is reasonable. This dose falls within the normal range of annual occupational doses observed in recent years at operating reactors. Although the doses resulting from the steam generator repair will increase the annual occupational dose average of 569 person-rems to approximately 720 person-rems per unit, this is still well below the 1300 person-rems per unit annual average which is an upper bound dose average of PWR's experiencing high levels of special maintenance work. The licensee has taken appropriate steps to ensure that occupational doses will be maintained within the limits of 10 CFR Part 20 and ALARA. The additional health risks due to these doses over normal risks are quite small, less than one percent of normal risk to the project work force as a whole. The risk to an average individual in the work force will be lower than the risk incurred from participation in many commonplace activities. For the foregoing reasons, the staff concludes that the environmental impact due to occupational exposure will not significantly affect the quality of the human environment.

2.2 Public Radiation Exposure

This section contains conservative estimates of the impacts on the public from the proposed steam generator repair project. The major sources of radiation and environmental pathways were considered in preparing this section. Public radiation exposure from the Indian Point Unit 3 steam generator repair can be estimated by comparing the estimated quantities of radioactive effluents from the steam generator repair with annual average releases and dose estimates from normal operations at Indian Point.

The licensee has estimated the amount of radioactivity that will be released in liquid and gaseous effluents as a result of the repair. Those estimates are presented in Table 3. The staff has reviewed the licensee's estimates and concluded that they are reasonable estimates. Table 3 also presents effluent releases for the years 1979, 1980 and 1981 from the plant and the FES annual average release estimates for normal operations. The expected releases from the repair are much less than both the FES estimates and the plant's actual annual releases for normal operations.

On the basis of this comparison, the staff concludes that the offsite environmental impact that may occur during the period of this procedure will be significantly smaller than that which occurs during normal operation.

The staff has estimated the doses to individual members of the public as well as the population as a whole in the area surrounding Indian Point based on the radioactive effluents which the licensee estimated for the repair (summarized in Table 3) and on the dose estimates in the FES. In the FES the staff estimated that the doses to the total body and any organ of the maximally exposed individual to either radioactive airborne effluents or radioactive liquid effluents would be less than about 5 millirems. Since the radioactive effluents from the repair are estimated to be less than 1% of the effluents from routine operations, the staff estimates that the doses to the total body and any organ of the maximally exposed individual to effluents from the repair will be much less than 1 millirem. This dose is equivalent to a very small fraction of the limits of 40 CFR Part 190. The annual limits of 40 CFR Part 190 are 25 millirems to the total body or any organ except the thyroid and 75 millirems to the thyroid. In a similar manner, the doses to the population of 19,000,000 persons within 50 miles of the plant are estimated to be less than 1 person-rems to the total body from exposure to airborne and liquid radioactive effluents from the repair.

Table 3. Radioactive effluents from steam generator repairs and normal operations at Indian Point Unit 3

Type of radioactive effluent	Repair, Ci/reactor	Normal operations, Ci/yr/reactor			
		Measured			
		1979	1980	1981	FES Estimates*
Gaseous					
Noble Gases	Negligible**	9,000.	1,100.	13,000.	2,700.
Iodines & Particulates	Negligible**	0.42	0.024	0.044	0.68
Tritium Negligible**		5.	8.7	4.2	***
Liquid					
Mixed fission and activation products	Negligible**,#	1.9	2.9	5.6	5.
Tritium Negligible**		470.	430.	240.	350.

*FES estimates are taken from Tables V-29 and 32 of NUREG-75/002.
**Below lower limits of detectability of plant instrumentation.
***No value given in the referenced report.
#It is estimated that approximately 10 to 15 curies of radioactive materials, primarily Co-58 and Co-60, will be removed during decontamination and honing procedures; however, none of this material is expected to appear in plant effluents, but will be solidified for disposal as solid radioactive wastes.

By comparison, every year the same population of about 19,000,000 will receive a cumulative total body dose of about 1,900,000 person-rems from natural background radiation (about 0.1 rem per year per person). Thus, the population total body dose from the repair is less than one millionth of the annual dose due to natural background. On this basis, the Staff concludes that the doses to individuals in unrestricted areas and to the population within 50 miles due to exposure to effluents from the repair will not be environmentally significant.

In summary, the estimated radioactive releases resulting from the repair are much less than those due to normal plant operation. The doses due to these releases are small compared to the limits of 40 CFR Part 190 and to the annual doses from natural background radiation. Therefore, the radiological impact of the repair will not significantly affect the quality of the human environment.

3.0 Conclusion

Based on the Staff's review of the proposed steam generator repair, the Staff concludes that:

(1) The estimated total occupational exposure of 1240 person-rems/reactor for the repair is within the expected range of doses incurred at light water power reactors in a year.

(2) The risks to the workers involved in the repair are comparable to the risks associated with other occupations.

(3) The licensee has taken appropriate steps to ensure that occupational dose will be maintained as low as is reasonably achievable and within the limits of 10 CFR Part 20.

(4) The estimated doses to the general public are:

- (a) much less than those incurred during normal operation of Indian Point Unit 3, and
- (b) negligible in comparison to the dose members of the public receive each year from exposure to natural background radiation.

On the basis of the foregoing analysis, it is concluded that there has been no environmental impact attributable to the repair work other than that which was previously predicted and described in the Commission's FES for the Indian Point Nuclear Generating Plant, Unit No. 3.

Dated: MAY 27 1983

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4.0 References

Material used in the preparation of this section includes the U.S. Nuclear Regulatory Commission publications listed immediately below by publication number as well as the other documents listed following in alphabetical order.

U.S. Nuclear Regulatory Commission NUREG Reports

NUREG-0713, B. G. Brooks, "Occupational Radiation Exposure at Commercial Nuclear Power Reactors 1980," Volume 2, December 1981.

NUREG-0800, "Radiation Protection," in: "Standard Review Plan," Chapter 12, July 1981. (Formerly issued as NUREG-75/087.)

U.S. Nuclear Regulatory Commission Regulatory Guides

1.109, Revision 1, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I," October 1977.

1.113, Revision 1, "Estimating Aquatic Dispersion of Effluent from Accidental and Routine Reactor Releases for the Purpose of Implementing Appendix I," April 1977.

8.8, Revision 3, "Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations Will Be as Low as Is Reasonably Achievable," June 1978.

Other References

Advisory Committee on the Biological Effects of Ionizing Radiations (BEIR I), "The Effects on Populations of Exposure to Low Levels of Ionizing Radiation," National Academy of Sciences/National Research Council, November 1972.

Advisory Committee on the Biological Effects of Ionizing Radiations (BEIR III), "The Effects on Populations of Exposure to Low Levels of Ionizing Radiation," National Academy of Sciences/National Research Council, July 1980.

American Cancer Society, "Cancer Facts and Figures 1979," 1978.

International Commission on Radiological Protection, ICRP, "Recommendations of the International Commission on Radiological Protection," ICRP Publication 26, January 1977.

National Council on Radiation Protection and Measurements, NCRP, "Review of the Current State of Radiation Protection Philosophy," NCRP Report No. 43, January 1975.

Power Authority of the State of New York (PASNY), Indian Point Unit 3 Steam Generator Sleeving Report prepared by Westinghouse Electric Corporation, Rev. 1, October 1982.

Power Authority of the State of New York (PASNY), "Indian Point 3 Nuclear Power Plant, Response to NRC Request for Information Concerning Steam Generator Girth Weld Repair Program," letter dated January 19, 1983.

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UNITED STATES NUCLEAR REGULATORY COMMISSION

POWER AUTHORITY OF THE STATE OF NEW YORK

DOCKET NO. 50-286

NOTICE OF ISSUANCE OF AMENDMENT TO

FACILITY OPERATING LICENSE AND FINAL NO SIGNIFICANT HAZARDS

CONSIDERATION DETERMINATION AND OPPORTUNITY FOR HEARING

AND NEGATIVE DECLARATION

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 47 to Facility Operating License No. DPR-64, issued to the Power Authority of the State of New York (the licensee), which revised Technical Specifications for operation of the Indian Point Nuclear Generating Unit No. 3 (the facility) located in Buchanan, Westchester County, New York. The amendment is effective as of the date of issuance.

This amendment involves three principal sets of changes, all relating to resumption of operation after steam generator repairs at the facility. The first set adds requirements for surveillance of steam generator upper girth welds governing operation after repair of cracking in certain steam generator shell upper girth welds. The second set modifies steam generator tube surveillance provisions, permits operation with steam generator tubes repaired by sleeving, and provides limits on degradation of sleeves. The third set of changes imposes secondary water chemistry monitoring requirements.

Before issuance of the license amendment, the Commission will have made findings required by the Atomic Energy Act of 1954, as amended (the Act), and the Commission's regulations.

The Commission has made a determination that the amendment request involves no significant hazards consideration. Under the Commission's standards in 10 CFR 50.92, this means that operation of the facility in accordance with the license amendment would not (1) involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety.

The Commission has determined that the issuance of the amendment will not result in any significant environmental impact and that pursuant to 10 CFR §51.5(d)(4) an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of the amendment. The Commission has also prepared an environmental impact appraisal for the completed repair work and has concluded that there has been no environmental impact attributable to the repair work other than that which was previously predicted and described in the Commission's Final Environmental Statement for the facility.

The Commission has provided guidance concerning the application of these standards by providing certain examples, which was published in the FEDERAL REGISTER on April 6, 1983 (48 FR 14864). None of the examples, relating to whether significant hazards considerations are likely or unlikely, appears to be directly applicable to this amendment. Consequently. the Commission has determined that the application does not involve a significant hazards consideration, since the applicant proposes compensatory measures to provide a level of safety in operation with the repaired steam generators commensurate with that of a facility that had not experienced the need to repair steam generators anticipated when the facility was initially licensed to operate.

The Commission did not seek public comments on this determination since it had planned to issue this amendment prior to the effective date of its new regulations governing procedures for no significant hazards determinations. Under preexisting practice notice of amendments which did not involve significant hazards were issued after the amendment's effective date. See 48 FR 14877. Since failure to issue this amendment before the expiration of a public comment period would result in a shutdown of the Indian Point 3 facility (See. 10 CFR §50.91(a)(5)), the Commission has determined the amendment should be issued without prior notice and opportunity for hearing or for public comment. However, the Director of Technological Development Programs, New York State Energy Office was advised of the subject of the licensee's request and of the NRC's actions.

By July 6, 1983, the licensee may file a request for a hearing with respect to issuance of the amendment to the subject facility operating license and any person whose interest may be affected by this proceeding and who wishes to participate as a party in the proceeding must file a written petition for leave to intervene. Request for a hearing and petitions for leave to intervene shall be filed in accordance with the Commission's "Rules of Practice for Domestic Licensing Proceedings" in 10 CFR Part 2. If a request for a hearing or petition for leave to intervene is filed by the above date, the Commission or an Atomic Safety and Licensing Board, designated by the Commission or by the Chairman of the Atomic Safety and Licensing Board Panel, will rule on the request and/or petition and the Secretary or the designated Atomic Safety and Licensing Board will issue a notice of hearing or an appropriate order.

As required by 10 CFR §2.714, a petition for leave to intervene shall set forth with particularity the interest of the petitioner in the proceeding and how that interest may be affected by the results of the proceeding. The petition should specifically explain the reasons why intervention should be permitted with particular reference to the following factors: (1) the nature of the petitioner's right under the Act to be made a party to the proceeding; (2) the nature and extent of the petitioner's property, financial, or other interest in the proceeding; and (3) the possible effect of any order which may be entered in the proceeding on the petitioner's interest. The petition should also identify the specific aspect(s) of the subject matter of the proceedings as to which petitioner wishes to intervene. Any person who has filed a petition for leave to intervene or who has been admitted as a party may amend the petition without requesting leave of the Board up to fifteen (15) days prior to the first prehearing conference scheduled in the proceeding, but such an amended petition must satisfy the specificity requirements described above.

Since the Commission has made a final determination that the amendment involves no significant hazards consideration, if a hearing is requested, it will not stay the effectiveness of the amendment. Any hearing held would take place while the amendment is in effect.

Not later than fifteen (15) days prior to the first prehearing conference scheduled in the proceeding, a petitioner shall file a supplement to the petition to intervene which must include a list of the contentions which are sought to be litigated in the matter, and the bases for each contention set forth with reasonable specificity. Contentions shall be limited to matters within the scope of the amendment under consideration. A petitioner who fails to file such a supplement which satisfies these requirements with respect to at least one contention will not be permitted to participate as a party.

Those permitted to intervene become parties to the proceeding, subject to any limitations in the order granting leave to intervene, and have the opportunity to participate fully in the conduct of the hearing, including the opportunity to present evidence and cross-examine witnesses.

A request for a hearing or a petition for leave to intervene must be filed with the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Docketing and Service Branch, or may be delivered to the Commission's Public Document Room, 1717 H Street, N.W. Washington, D.C., by the above date. Where petitions are filed during the last ten (10) days of the notice period, it is requested that the petitioner promptly so inform the Commission by a toll-free telephone call to Western Union at (800) 325-6000 (in Missouri (800) 342-6700). The Western Union operator should be given Datagram Identification Number 3737 and the following message addressed to Steven A. Varga, Chief, Operating Reactors Branch No. 1, Division of Licensing: petitioner's name and telephone number; date petition was mailed; plant name; and publication date and page number of this FEDERAL REGISTER notice. A copy of the petition should also be sent to the Executive Legal Director, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, and to Charles M. Pratt, Assistant General Counsel, Power Authority of the State of New York, 10 Columbus Circle, New York, New York 10019, attorney for the licensee.

Nontimely filings of petitions for leave to intervene, amended petitions, supplemental petitions and/or requests for hearing will not be entertained absent a determination by the Commission, the presiding officer or the Atomic Safety and Licensing Board designated to rule on the petition and/or request, that the petitioner has made a substantial showing of good cause for the granting of a late petition and/or request; that determination will be based upon a balancing of the factors specified in 10 CFR 2.714(a)(1)(i)-(v) and 2.714(d).

For further details with respect to this action, see (1) the submittal dated October 18, 1982, as supplemented by letters dated January 19, 1983, May 2, 1983 and May 3, 1983, (2) Amendment No. 47 to License No. DPR-64, (3) the Commission's related Safety Evaluation, (4) the Commission's related letter dated May 27, 1983, and (5) the Commission's Environmental Impact Appraisal. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N.W., Washington, D.C. and at the White Plains Public Library, 100 Martine Avenue, White Plains, New York. A copy of items (2), (3), (4) and (5) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Director, Division of Licensing.

Dated at Bethesda, Maryland, this 27th day of May 1983.

FOR THE NUCLEAR REGULATORY COMMISSION

Steven A. Varga

Steven A. Varga, Chief
Operating Reactors Branch No. 1
Division of Licensing