ATTACHMENT I TO IPN-96-027

PROPOSED TECHNICAL SPECIFICATION CHANGE

REGARDING STEAM GENERATOR TUBE

INSPECTION INTERVAL

NEW YORK POWER AUTHORITY
INDIAN POINT 3 NUCLEAR POWER PLANT
DOCKET NO. 50-286
DPR-64

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 into 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.

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.

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. 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.

* Except that the surveillance related to the steam generator tube inspection due no later than July 1996, may be deferred until the next refueling outage but no later than May 31, 1997.

ATTACHMENT II TO IPN-96-027

SAFETY EVALUATION FOR THE

PROPOSED TECHNICAL SPECIFICATION CHANGE

REGARDING STEAM GENERATOR TUBE

INSPECTION INTERVAL

NEW YORK POWER AUTHORITY
INDIAN POINT 3 NUCLEAR POWER PLANT
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SAFETY EVALUATION OF PROPOSED TECHNICAL SPECIFICATION CHANGE REGARDING STEAM GENERATOR TUBE INSPECTION INTERVAL

SECTION I- Description of Change

This application for amendment to the Indian Point 3 Technical Specifications proposes to revise Section 4.9.A.4 of Appendix A of the Operating License. The proposed change would allow a one time extension to the steam generator tube inspection interval. This inspection is currently due in July 1996. The steam generator tube inspection is scheduled to be performed during the next refueling outage (RO 9) but no later than May 31, 1997. Though subject to change, RO 9 is currently scheduled to start in February 1997. Without this one-time extension, a forced outage will be necessary to perform the required inspection. The forced outage will increase the probability of events which are more likely to occur during transient operation (heatup and cooldown) than at full power steady state operation and subject plant equipment to further heatup/cooldown cycles resulting in unnecessary equipment wear. The proposed change would revise the Technical Specifications Section 4.9.A.4.c to include a footnote. The footnote would indicate that the surveillance related to the steam generator tube inspection due no later than July 1996, may be deferred until the next RO but no later than May 31, 1997.

SECTION II- Evaluation of Change

In accordance with Technical Specification 4.9.A.4.b, the steam generator tubes are currently required to be inspected at intervals no greater than 24 calendar months after the previous inspection. As per Technical Specification 4.9.A.4.c, this inspection interval may be increased to 40 months if the results of two consecutive inspections, excluding the preservice inspection, fall into Category C-1 (Category C-1 is less than 5 % of the tubes inspected are degraded and none of them is defective). The Indian Point 3 steam generators meet this criterion. The last steam generator tube inspection was conducted in May 1992. Based on a 50 month inspection interval, which includes a 25% extension allowance for surveillance interval per Technical Specification 1.12, the next inspection would have to be done by July 1996. This application for amendment proposes a one time change to the Technical Specification 4.9.A.4.c that would allow the interval for the steam generator tube inspection to exceed 40 months. The next inspection for the steam generator tubes would be scheduled to be performed during the next refueling outage (RO 9) but no later than May 31, 1997.

As a result of an extended Restart and Continuous Improvement outage from February 1993 to July 1995, and a subsequent forced outage from September 1995, the Authority estimates the start of the Indian Point 3 RO 9 in February 1997. This schedule is subject to change based on, among other things, IP3's return to service date. Under the current Technical Specification requirement, the due date for completion of the next steam generator tube inspection will occur prior to the 1997 outage. The proposed footnote would extend the

surveillance related to the steam generator tube inspection due no later than July 1996, to the next RO but no later than May 31, 1997.

Replacement steam generators were installed at Indian Point 3 in 1989. The replacement steam generators have only operated for a small fraction of their design service life. The IP3 steam generators employ advanced design features which make them highly resistant to tube degradation. All types of tube degradation require at least two of the following to occur; a susceptible material, a source of stress and/or a corrosive environment. The corrosion process is generally a function of temperature and time. In addition to using corrosion resistant materials, the steam generators' environment is carefully controlled to minimize corrosive effects.

The replacement steam generators are Westinghouse Model 44F and have operated for less than 3 Effective Full Power Years. In addition, since the last steam generator inspection in May 1992, the steam generators have operated for less than one year. The steam generator tube bundle consists of 3214 U-tubes, fabricated from thermally treated Alloy 690 (ASME-SB-163 Alloy UNS N06690 to Code Case N-20). The U-tube ends are hydraulically expanded to the full depth of the tubesheet. Based on tube material qualification testing and extensive laboratory research conducted by EPRI, this tube material and tube expansion process are highly resistant to tube degradation.

In addition to the tube material and tube expansion process features which make the Indian Point 3 steam generators highly resistant to tube degradation, a comprehensive chemistry control program is in place which protects the steam generator tubing. The chemistry control program maintains the water chemistry in both the primary and secondary sides of the steam generators to minimize the corrosive environment for the tubes. The primary water chemistry control program assures fuel and reactor coolant system integrity, which includes steam generator tubes on the primary side. The Indian Point 3 Facility Operating License, Section 2.I (Amendment 29) requires a secondary water chemistry monitoring program to inhibit steam generator tube degradation. The chemistry controls in place minimize the corrosive environment on the secondary side of the steam generators which assures a very low likelihood of tube degradation occurring. When the steam generators are not operating, there is a lay-up program which maintains the secondary side of the steam generators in an environment which minimizes the potential for corrosion. There have been no gross (greater than EPRI level 3) chemistry excursions since the new steam generators were installed. In addition, there have been visual inspections and corrosion product chemical analyses performed on the secondary side of the steam generators in both refueling outages since the replacement of the steam generators. The secondary side visual inspection and corrosion product chemical analyses indicated no problems and confirmed the effectiveness of the chemistry control program.

The reactor coolant temperature entering the steam generator tubes during operation on the hot leg side (T-hot) has been maintained at approximately 593 degrees F since the steam generators were replaced. At 593 degrees F, IP3 is considered a low-temperature plant (below the current industry demarcation of 600 degrees F, which distinguishes "low-

temperature" from "high temperature" operating plants when considering the impact of temperature on corrosion processes).

A baseline inspection of 100% of the replacement steam generator tubes was performed prior to initial operation in 1989. There have been two subsequent steam generator tube inspections performed. Both inspections included a 20% random sample of tubes in all 4 steam generators using the same type of bobbin coil probe used during the baseline inspection. All tubes were found to be in a "like new" condition with no service imperfections noted (zero degraded or defective tubes as defined by the Technical Specifications).

There are administrative controls in place to provide defense in depth, thus providing for the health and safety of the public. Indian Point 3 Technical Specification Section 3.1.F states the maximum steam generator tube leakage allowed. The Authority has imposed more restrictive administrative limits to ensure corrective actions are taken before a Technical Specification limit is reached. For example, administrative leakage limits imposed by the Authority are approximately 3 times more restrictive than those stated in the Technical Specifications; also, the Authority performs monitoring of all steam generator tube leakage and has administrative controls to ensure that the rate of change of the leak does not increase beyond prescribed limits.

During the next steam generator tube inspection, in addition to the normal eddy current inspection program, the Authority will perform an augmented inspection for the detection of circumferential cracking at the hot leg expansion transition areas, small radius (row 1 and 2 only) U-bend areas and dented location areas in response to Generic Letter 95-03. Sample size and sample expansion criteria for the augmented inspection will be in accordance with plant Technical Specifications. The augmented inspection will use enhanced inspection techniques suitable for the detection of circumferential cracks. The methods, equipment and criteria of the augmented inspection will be in accordance with the current revision of EPRI Report NP-6201, "PWR Steam Generator Examination Guidelines." If a rotating pancake coil is used, terrain plots will be used to analyze the rotating pancake coil eddy current data at locations susceptible to circumferential cracking.

In conclusion, the design and operating environment, the minimal service life to date and the relatively short time of operation above 200 degrees F since the last inspection of the Indian Point 3 steam generators, provide excellent assurance against tube failures over the life of the plant. Inspections performed on the replacement steam generator tubing and planned future tubing inspections ensure that tube degradation, if initiated, will be detected. Even if advanced tube degradation did go unnoticed, a number of systems are available to provide defense in depth thus providing for the health and safety of the public. Therefore, there is no adverse consequence of deferring the steam generator tube inspection until the next refueling outage.

Section III - No Significant Hazards Evaluation

Consistent with the criteria of 10 CFR 50.92, the enclosed application is judged to involve no significant hazards based on the following information:

(1) Does the proposed license amendment involve a significant increase in the probability or consequences of an accident previously evaluated?

Response:

The proposed license amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated. As stated in the Basis of the IP3 Technical Specifications, the program for inservice inspection of steam generator tubes regarding equipment, procedures, and sample selection is based upon the guidance and recommendations in Regulatory Guide 1.83 and NRC Generic Letter 85-02. The addition of the footnote to extend the surveillance due date will not increase the deviation from the guidance and recommendation stated above, and, therefore will not involve a significant increase in the probability or consequences of an accident previously evaluated.

(2) Does the proposed license amendment create the possibility of a new or different kind of accident from any accident previously evaluated?

Response:

The proposed license amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed change does not involve the addition of any new or different type of equipment, nor does it involve the operation of equipment required for safe operation of the facility in a manner different from those addressed in the Final Safety Analysis Report. Therefore, the proposed change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

(3) Does the proposed amendment involve a significant reduction in a margin of safety?

Response:

The proposed license amendment does not involve a significant reduction in a margin of safety. The proposed change does not adversely affect any safety related system or component operation or operability, instrument operation, or safety system setpoints and does not result in increased severity of any of the accidents considered in the safety analysis. This change has no adverse effect on any margin of safety and, therefore, does not create a significant reduction in a margin of safety.

Section IV - Impact of Change

This change will not adversely affect the following:
ALARA Program
Security and Fire Protection Programs
Emergency Plan
FSAR or SER Conclusions
Overall Plant Operations and the Environment

Section V - Conclusions

The incorporation of this change: a) will not significantly increase 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 create the possibility of an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report; c) will not reduce the margin of safety as defined in the bases for any technical specification; and d) involves no significant hazards considerations as defined in 10 CFR 50.92.

Section VI - References

1) NYPA letter to the NRC dated June 27, 1995 (IPN-95-072), "Response to Generic Letter 95-03: Circumferential Cracking of Steam Generator Tubes."
2) NYPA letter to the NRC dated January 17, 1996 (IPN-96-004), "Response to Request for Additional Information (RAI), Response to GL 95-03, Circumferential Cracking of Steam Generator Tubes."
3) IP3 FSAR
4) IP3 SER