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NewYorkPower Authority

William J. Cahill, Jr.
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June 27, 1995
IPN-95-072

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
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SUBJECT: Indian Point 3 Nuclear Power Plant
Docket No. 50-286
License No. DPR-64
Response to NRC Generic Letter 95-03:
"Circumferential Cracking of Steam Generator Tubes"

References:
1. Generic Letter 95-03: "Circumferential Cracking of Steam Generator Tubes," U. S. Nuclear Regulatory Commission, April 28, 1995.
2. EPRI Report NP-6201, "PWR Steam Generator Examination Guidelines"

Dear Sir:

This letter provides the Authority's response to Generic Letter (GL) 95-03, "Circumferential Cracking of Steam Generator Tubes," dated April 28, 1995, (Reference 1) for the Indian Point 3 Nuclear Power Plant (IP3). This generic letter was addressed to all holders of operating licenses or construction permits for pressurized water reactors (PWRs). The generic letter requests all holders of an operating license for a PWR to submit to the Nuclear Regulatory Commission (NRC) a written response, per 10 CFR 50.54(f), within 60 days from the date of the generic letter.

The NRC issued this generic letter to (1) notify addressees about recent steam generator tube inspection findings and the safety significance of these findings, (2) request that all addressees implement the actions described in the generic letter, and (3) request that all addressees submit to the NRC a written response to the generic letter regarding implementation of the requested actions. In addition, the generic letter alerts licensees to the importance of performing comprehensive examinations

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of steam generator tubes using techniques and equipment capable of reliably detecting degradation to which the steam generator tubes may be susceptible.

This letter submits the safety assessment justifying continued operation of the Indian Point 3 Nuclear Power Plant based on the evaluations performed in accordance with NRC Requested Actions (1) and (2) and a summary of Indian Point 3 Nuclear Power Plant inspection plans, including a schedule for the next planned inspection, developed in accordance with NRC Requested Action (3) from Generic Letter 95-03.

Attachment I of this letter describes the implementation of the requested actions for Indian Point 3. These actions are based on generic industry guidance and additional Indian Point 3 plant-specific information to establish justification for continued operation.

Attachment II of this letter contains the commitments associated with this letter. If you have any questions regarding this matter, please contact Mr. Ken Peters.

Very truly yours,

William J. Cahill, Jr.
Chief Nuclear Officer
Nuclear Generation

STATE OF NEW YORK
COUNTY OF WESTCHESTER
Subscribed and Sworn to before me
this 27th day of June, 1995

Geraldine Strand
Notary Public

GERALDINE STRAND
Notary Public, State of New York
No. 4991272
Qualified in Westchester County
Commission Expires Jan 27, 1996

cc: See next page

att: as stated

att: as stated

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Resident Inspector's Office
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ATTACHMENT I to IPN-95-072

IMPLEMENTATION OF REQUESTED ACTIONS

PERTAINING TO

CIRCUMFERENTIAL CRACKING OF STEAM GENERATOR TUBES

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

The Authority's response to the NRC Requested Actions (1) and (2), below, submit the safety assessment justifying continued operation for the Indian Point 3 Nuclear Power Plant.

NRC REQUESTED ACTION:

(1) Evaluate recent operating experience with respect to the detection and sizing of circumferential indications to determine the applicability to Indian Point 3.

NYPA RESPONSE:

The Indian Point 3 steam generators are Westinghouse Model 44F with hydraulically expanded, thermally treated (TT), Alloy 690 tubing. A list of other plants with steam generators using a similar tube expansion process and tube material is provided in the table below. The Indian Point 3 steam generators use Alloy 690 TT tubing with a nominal tube size of .875 inch Outer Diameter x .050 inch nominal wall thickness.

Hydraulically Expanded
Alloy 690 Thermally Treated Tubes *

Plant/Steam Generator Model	Startup	First Time Circ. Cracking	Location	Tube Pull and Results
Cook Unit 2 / 54F	1989	none	N/A	N/A
Indian Point 3 / 44F	1989	none	N/A	N/A
North Anna Unit 1 / 54F	1993	none	N/A	N/A
North Anna Unit 2 / 54F	mid 1995	N/A	N/A	N/A
V. C. Summer / 75	1994	none	N/A	N/A

Note: All steam generators are replacement steam generators
*Information in table was provided by Westinghouse

As shown in the table, no circumferential cracking has initiated in plants with hydraulically expanded, thermally treated, Alloy 690 tubing. In addition, thermally treated Alloy 690 sleeves have been in service at other nuclear power plants since 1983 and based on information received from Westinghouse, there has been no reported degradation in these sleeves.

NRC REQUESTED ACTION:

(2) On the basis of the evaluation in Item (1) above, past inspection scope and results, susceptibility to circumferential cracking, threshold of detection, expected or inferred crack growth rates, and other relevant factors, develop a safety assessment justifying continued operation until the next scheduled steam generator tube inspections are performed.

NYPA RESPONSE:

The steam generators at Indian Point 3 do not pose a threat to the health and safety of the public and are safe to operate until the next scheduled steam generator tube inspections are performed.

NRC Generic Letter 95-03 addressed the issue of circumferential cracking of steam generator tubes. There are two types of tube degradation which can lead to circumferential cracking of steam generator tubes. On the primary side of the steam generators, Primary-side Stress Corrosion Cracking (PWSCC) is the tube degradation form which can lead to circumferential cracking. On the secondary side of the steam generators, secondary side Stress Corrosion Cracking (SCC) is the tube degradation form which can lead to circumferential cracking. Both types of tube degradation require a susceptible material, a source of stress, and a corrosive environment. The corrosion process is generally a function of temperature and time.

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 the type of cracking identified in Generic Letter 95-03. In addition to using corrosion resistant materials, the steam generator fabrication process was designed to minimize stress levels in the tubes. The steam generators' environment is carefully controlled to minimize corrosive effects, and the temperature at which the steam generators operate is below the temperature at which the corrosion effect is accelerated.

Enhanced inspection techniques will be used in the next refuel outage to ensure early detection of circumferential tube cracks. However, this tube material has shown excellent service performance to date and cracking is not expected over the design life of the steam generators.

The replacement steam generators are Westinghouse Model 44F and have operated for less than 3 Effective Full Power Years. The steam generator tube bundle consists of 3214 thermally treated U-tubes, fabricated from Alloy 690 (ASME-SB-163 Alloy UNS N06690 to Code Case N-20). The U-tube ends are hydraulically expanded the full depth of the tubesheet. In experiments conducted by EPRI, thermally treated Alloy 690 has not cracked as a result of PWSCC, and is substantially more resistant to secondary side SCC than other tube materials. A comprehensive evaluation of recent steam generator operating experience has been performed by Westinghouse. This evaluation concludes there are four other nuclear plants in the United States with hydraulically expanded, thermally treated Alloy 690 tube material, and none of these plants have experienced any tube cracks (either circumferential or axial). The oldest steam generators with hydraulically expanded, thermally treated Alloy 690 tubing, have been in service since 1989 (see table on page 1).

In addition to the above design features which make the Indian Point 3 steam generators highly resistant to the type of cracking described in Generic Letter 95-03, a comprehensive chemistry control program is in place which protects the steam generator tubing. This 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 tubing on the primary side. The Indian Point 3 Facility Operating License, Section (2) (I) 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 secondary side SCC 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 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 chemistry control program effectiveness.

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. This is below the threshold value of 600 degrees F normally associated with increased incidence of PWSCC and secondary side SCC.

A baseline inspection, of 100% of the replacement steam generator tubes, was performed after installation and prior to initial operation in 1989. There have been two subsequent steam generator tubing inspections performed. Both inspections included a 20% random sample of tubes in all 4 steam generators using the same type bobbin coil probe used during the baseline inspection. All tubes were found to be in a like new condition with no service imperfections noted. Future inspections will utilize enhanced inspection techniques, for the early detection of circumferential cracks, although this cracking is not expected.

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 with the reactor critical and at power. The New York Power Authority (NYPA) 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 NYPA are approximately 3 times more restrictive than those stated in the Technical Specification; also, NYPA 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.

In conclusion, the design and operating environment 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 circumferential cracking 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, it is safe to continue plant operation until the next scheduled steam generator tube inspections.

NRC REQUESTED ACTION:

(3) Develop plans for the next steam generator tube inspections as they pertain to the detection of circumferential cracking. The inspection plans should address but not be limited to, scope (including sample expansion criteria, if applicable), methods, equipment, and criteria (including personnel training and qualification).

NYPA RESPONSE:

The New York Power Authority will perform an augmented inspection for the detection of circumferential cracking of the hot leg expansion transition areas during the next scheduled steam generator tube inspection. Sample size and sample expansion criteria, 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." The next steam generator tube inspection is required by Technical Specifications to be performed by June 1996. However, our schedule for the next steam generator tube inspection is currently being evaluated.

ATTACHMENT II to IPN-95-072

AUTHORITY COMMITMENTS

RELATED TO

CIRCUMFERENTIAL CRACKING OF STEAM GENERATOR TUBES

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

COMMITMENTS ASSOCIATED WITH IPN-95-072

Commitment Number	**Commitment**	**Due Date**
IPN-95-072-01	Perform an augmented inspection for the detection of circumferential cracking on a sample of the hot leg expansion transition areas. The sample size and sample expansion criteria will be in accordance with the Technical Specifications, and will be performed during the next scheduled steam generator tube inspection.	Next scheduled steam generator tube inspection
IPN-95-072-02	Enhanced inspection techniques, in accordance with EPRI Report NP-6201 will be used in the next scheduled steam tube inspection to ensure early detection of circumferential tube cracks.	Next scheduled steam generator tube inspection