ATTACHMENT A

PROPOSED TECHNICAL SPECIFICATIONS

CONSOLIDATED EDISON COMPANY OF NEW YORK, INC.
INDIAN POINT UNIT NO. 2
DOCKET NO. 50-247
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2. Operational Leakage Limits

a. Primary to Secondary Leakage

(1) Primary to secondary leakage through the steam generator tubes shall not exceed 0.3 gpm in any steam generator. With any steam generator tube leakage greater than this limit, the reactor shall be brought to the cold shutdown condition within 24 hours.

(2) If leakage from two or more steam generators in any 20-day period is observed or determined, the reactor shall be brought to the cold shutdown condition within 24 hours and Nuclear Regulatory Commission approval shall be obtained before resuming reactor operation. If tube leaks attributable to the tube denting phenomena are observed in two or more steam generators after the reactor is in cold shutdown, Nuclear Regulatory Commission approval shall be obtained before resuming reactor operation.

(3) Whenever the reactor is shut down in order to investigate steam generator tube leakage and/or to plug or otherwise repair a leaking tube, the NRC shall be informed before any tube is either plugged or repaired, or if no tube is either plugged or repaired, before the steam generator is returned to service.

b. RCS/RHR Pressure Isolation Valves Leakage

(1) Whenever the reactor is above cold shutdown, leakage through each of the RCS/RHR pressure isolation valves 897A, B, C and D, and 838A, B, C and D shall satisfy the following acceptance criteria:

(a) Leakage rates of less than or equal to 1.0 gpm are acceptable.

coolant system and the secondary coolant system. Leakage in excess of 0.3 gpm for any steam generator will require plant shutdown and the leaking tube(s) will be located and either plugged or repaired.

The 10 gpm limit for combined reactor coolant and non-reactor coolant leakage into the containment free volume provides allowance for a limited amount of leakage from sources other than the reactor coolant system within containment while conservatively limiting total leakage into the containment free volume to the same limit (i.e., 10 gpm) for identified reactor coolant leakage alone. This leakage is within the capabilities of the leakage detection and waste processing system and will not interfere with the detection of independent unidentified reactor coolant system leakage.

For those circumstances where high energy line failures occur inside containment resulting in flooding of the containment building sumps and/or floor, automatic actuation of reactor protection, safety injection and/or containment spray systems places the plant in a safe condition and, in some cases, provides intended flooding of the containment building. However, for those circumstances resulting from leakage or failure of low energy systems such as service water or component cooling inside containment, operator action is necessary to prevent accumulation of water on the containment floor to undesirable levels.

If the water level in the containment sump reaches EL. 45′, or the water level in the recirculation sump reaches EL. 35′, or the water level in the reactor cavity reaches EL. 20′, the reactor is placed in cold shutdown within the next 36 hours. If the water level in the containment sump increases above EL. 45′ and the water level in the recirculation sump increases above EL. 39′ 9″, or the water level in the reactor cavity increases above EL. 20′ 5″, the operator will immediately bring the reactor subcritical and initiate an expeditious cooldown of the plant.

4.13 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.

Specifications

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

A. INSPECTION REQUIREMENTS

1. Definitions

a. Imperfection is a deviation from the dimension, finish, or contour required by drawing or specification.

b. Deformation is a deviation from the initial circular cross-section of the tubing. Deformation includes the deviation from the initial circular cross-section known as denting.

c. Degradation means service-induced cracking, wastage, pitting, wear or corrosion (i.e., service-induced imperfections).

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

e. <u>% Degradation</u> is an estimated % of the tube or sleeve wall thickness affected or removed by <u>degradation</u>.

f. <u>Defect</u> is a degradation of such severity that it exceeds the <u>plugging limit</u>. A tube or sleeve containing a <u>defect</u> is <u>defective</u>.

g. <u>Plugging Limit</u> is the degradation depth at or beyond which the tube must be plugged or repaired.

h. <u>Hot-Leg Tube Examination</u> is an examination of the hot-leg side tube length. This shall include the length from the point of entry at the hot-leg tube sheet around the U-bend to the top support of the cold leg.

i. <u>Cold-Leg Tube Examination</u> is an examination of the cold-leg side tube length. This shall include the tube length between the top support of the cold leg and the face of the cold-leg tube sheet.

j. <u>F* Distance</u> is the distance of the expanded portion of a tube which provides a sufficient length of undegraded tube expansion to resist pullout of the tube from the tubesheet. The F* distance is equal to 1.25 inches and is measured down from the bottom of the roll transition.

k. <u>F* Tube</u> is a tube:

a) With degradation equal to or greater than 40% below the F* distance, and b) which has no indication of imperfections within the F* distance, and c) that remains in service.

1. Sleeving refers to tube repair achieved either by mechanical sleeving, as described by Westinghouse Report WCAP-12223, or laser welded sleeving, as described by Westinghouse Report WCAP-13583 and 13088. Sleeving is used to maintain a tube in service or return a previously plugged tube to service.

2. Extent and Frequency of Examination

a. Steam generator examinations shall be conducted not less than 12 months nor later than twenty four calendar months after the previous examination.

b. Scheduled examinations shall include each of the four steam generators in service.

c. Unscheduled steam generator examinations shall be required in the event there is a primary to secondary leak exceeding technical specifications, a seismic occurrence greater than an operating basis earthquake, a loss-of-coolant accident requiring actuation of engineered safeguards, or a major steamline or feedwater line break.

d. Unscheduled examinations may include only the steam generator(s) affected by the leak or other occurrence.

e. In case of an unscheduled steam generator examination, the profilometry tensile strain criterion shall be the same as contained in the approved program for the last scheduled steam generator inspection.

3. Basic Sample Selection and Examination

a. At least 12% of the tubes in each steam generator to be examined shall be subjected to a hot-leg examination.

b. At least 25% of the tubes inspected in Specification 4.13.A.3.a above shall be subjected to a cold-leg examination.

c. Tubes selected for examination shall include, but not be limited to, tubes in areas of the tube bundle in which degradation has been reported, either at Indian Point 2 in prior examinations, or at other utilities with similar steam generators.

d. Examination for deformation ("dents") shall be either by eddy current or by profilometry.

e. Examination for degradation other than deformation shall be by eddy current techniques, using a 700-mil diameter probe. If the 700-mil diameter probe cannot pass through the tube, a 610-mil diameter probe shall be used. For examination of the U-bends and cold-legs of tubes in rows 2 through 5, a 540-mil diameter probe may be used, provided it is justified by profilometry measurement within the tensile strain criterion.

f. In addition to the minimum sample size as determined by Table 4.13-1, all F* tubes shall be inspected within the pertinent tubesheet region. The results of F* tube inspections are not to be utilized as a basis for additional inspections per Table 4.13-1.

4. <u>Additional Examination Criteria</u>

1. Degradation Not Caused by Denting

a. If 5% of more of the tubes examined in a steam generator exhibit degradation or if any of the tubes examined in a steam generator are defective, additional examinations shall be required as specified in Table 4.13-1.

b. Tubes for additional examination shall be selected from the affected area of the tube array and the examination may be limited to that region of the tube where degradation or defective tube(s) were detected.

c. The second and third sample inspections in Table 4.13-1 may be limited to the partial tube inspection only, concentrating on tubes in the areas of the tube sheet array and on the portion of the tube where tubes with imperfections were found.

2. Degradation Caused by Denting

a. Additional examinations, for degradation caused by denting, shall be performed as described in the most recent steam generator examination program approved by the NRC.

B. ACCEPTANCE CRITERIA AND CORRECTIVE ACTION

1. Tubes shall be considered acceptable for continued service if:

a. depth of degradation is less than:

- 40% of the tube wall thickness, or
- 23% of the sleeve wall thickness

AND

b. the tube will permit passage of a 0.540" diameter probe and the strain in the tube wall (if measured) is less than the tensile strain criterion as specified in the approved examination program, or the tube will permit passage of a 0.610" diameter probe in the absence of strain measurement.

c. the tube is an F* tube and meets a. and b. above outside of the F* region.

2. Tubes that are not considered acceptable for continued service shall be plugged or repaired.

C. REPORTS AND REVIEW AND APPROVAL OF RESULTS

1. The proposed steam generator examination program shall be submitted for NRC staff review and concurrence at least 60 days prior to each scheduled examination.

2. The results of each steam generator examination shall be submitted to NRC within 45 days after the completion of the examination. A significant increase in the rate of denting or significant change in steam generator condition shall be reportable immediately.

3. An evaluation which addresses the long term integrity of small radius U-bends beyond row 1 shall be submitted within 60 days of any finding of significant hour-glassing (closure) of the upper support plate flow slots.

4. Restart after the scheduled steam generator examination need not be subject to NRC approval.

Basis

Inservice examination of steam generator tubing is essential if there is evidence of mechanical damage or progressive deterioration in order to assure continued integrity of the tubing. Inservice examination 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 examination was performed on each tube in each steam generator by eddy current techniques prior to service in order to establish a baseline condition for the tubing. No significant baseline imperfections were identified. In addition, prior to the discontinuance of phosphate treatment and the institution of all-volatile treatment (AVT), a baseline inspection was conducted in March, 1975 before the resumption of power operation.

Wastage-type defects are unlikely with the all-volatile treatment (AVT) of secondary coolant; however, even if this type of defect occurs, the steam generator tube examination will identify tubes with significant degradation from this effect.

The results of steam generator tube burst and collapse tests have demonstrated that tubes having wall thickness of 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. An allowance of 10% for tube degradation that may occur between inservice tube examinations added to the 40% degradation depth provided in the acceptance criteria provides an adequate margin to assure that tubes considered acceptable for continued operation will not have a minimum tube wall thickness of less than the acceptable 50% of normal tube wall thickness (i.e. 0.025 inch) during the service life-time of the tubes. Steam generator tube examinations of other operating plants have demonstrated the capability to reliably detect wastage type defects that have penetrated 20% of the original 0.050 inch wall thickness.

Examination of samples of tubes and support plates removed from steam generators have revealed that "denting" is caused by the accretion of steel corrosion products in the tube/support plate annuli. As these corrosion products are more voluminous than the support plate material from which they are derived, a compressive force is exerted on the tubes in the plane of the support plates, resulting in deformation of the tubes. If the deformation results in an ovalization of the tubes, the resulting strain is low and there is no risk of development of stress corrosion cracking in the tubes. However, if the deformation results in an irregular tube shape, the resulting strain may be high enough for the tube to become susceptible to stress corrosion cracking inservice, and it should be preventively repaired. Beginning with the steam generator examination to be conducted during the Cycle 5/6 Refueling Outage, the tensile strain criterion for profilometry shall be 25%. The 25% strain criterion is based on a review of data currently available from operating steam generators, and will be revised as necessary as more experience is gained with the evaluation of this measurement. In the future, this criterion may be revised, either higher or lower, based on steam generator examination results. The profilometry criterion to be used for any steam generator examination shall be established in the most recent program approved by NRC.

A first report on the R&D work leading to the development of profilometry, entitled "Profilometry of Steam Generator Tubes" dated August, 1980, was forwarded to the NRC by Con Edison. Additional R&D work has improved the accuracy of the profilometer and the calculation of strain in a deformed tube.

Before the development of profilometry, a minor diameter of 0.610" was established as the criterion for continuing a tube inservice. This criterion was used successfully for several years at Indian Point Unit 2 and at other plants, and appears to be sufficiently conservative so that it can be continued in the absence of more accurate strain determination by means of profilometry.

A sound roll expansion throughout the F* distance provides a tube to tubesheet interface that ensures the requirements of Regulatory Guide 1.121 are met regardless of the severity of any tube imperfections below the F* distance. The F* distance of 1.25 inches is comprised of 1.01 inches of sound roll that ensures tube integrity requirements are met plus 0.24 inches which allows for eddy current measurement uncertainty. The testing and analysis supporting the F* distance is documented in B&W Nuclear Technologies Qualification Report No. BAW-10195P.

This program for inservice inspection of steam generator tubes exceeds the requirements of Regulatory Guide 1.83, Revision 1, dated July 1975.

Table 4.13-1

Steam Generator Tube Inspection

First Sample Inspection			Second Sample Inspection		Third Sample Inspection	
Minimum Size	Result	Action	Result	Action	Result	Action
12% tubes per steam generator hot leg plus 3% tubes per steam generator cold leg	C-1	- - - - - -	- - -	- - - - - -	- - >	Go to power.
	C-2	Plug or repair defective tubes. Inspect additional 6% tubes in this S.G.	C-1 -	- - - - - -	- - >	Go to power.
			C-2	Plug or repair defective tubes. Inspect additional 12% tubes in this S.G.	C-1->	Go to power.
					C-2->	Plug or repair defective tubes. Go to power.
					C-3->	Go to first sample. C-3 action.
			C-3	Go to first sample. C-3 action.		
	C-3	Inspect all tubes this S.G. Plug or repair defective tubes. Inspect 6% tubes in each other S.G. if not included in the examination program.	All other S.G.s C-1 -	- - - - - - - - - - - -	- - - >	Go to power.
			Some S.G.s C-2 but no add'l C-3	Go to second sample. C-2 action.		
			Add'l S.G. C-3	Inspect all tubes in all S.G.s. Plug or repairs defective tubes.	- - >	Report to NRC. NRC approval req'd prior to startup.

ATTACHMENT B

SAFETY ASSESSMENT

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SAFETY ASSESSMENT

Background:

Approval of the proposed change will provide Con Edison with the option of repairing a steam generator tube that contains a defect. Steam generator tube sleeving is an industry proven repair technique. The proposed change also defines the F* distance as the length of sound tube-to-tubesheet contact that assures tube and system integrity requirements are met.

Sleeving is the process of placing a smaller, shorter tube (sleeve) inside an existing tube. This sleeve extends through and for some length above the tube sheet. When the sleeve tube is sealed to the original tube at both ends of the sleeve, it effectively forms a new primary/secondary coolant barrier. Thus, if a defect were to exist in the lower region of an existing tube, the sleeving process as described would be a viable alternative to plugging the entire tube.

The advantage to sleeving is that existing tubes remain in service since reactor coolant can still flow on the inside of the sleeve and throughout the remainder of the existing steam generator tube. Thus, the sleeving option extends the life of the steam generators by allowing tubes which possess defects to be repaired and remain in service.

The sleeving repair processes here evaluated are mechanical sleeving and laser welded sleeving as described and validated in WCAP-12223 (mechanical sleeving) and WCAP-13583 and 13088 (laser welded sleeving). Proprietary and non-proprietary versions of these reports, will be provided under separate cover.

The proposed addition of an F* distance will provide an acceptance criterion for tubes with defects in the rolled region of the tube. The F* distance of 1.25 inches is based on engineering analysis and physical testing conducted by B&W Nuclear Technologies and documented in Qualification Report BAW-10195P. Proprietary and non-proprietary versions of this report will be provided under separate cover.

The F* distance defines a length of continuous roll expansion for which the structural integrity of the tube is assured during normal operation and postulated accident conditions as required by Regulatory Guide 1.121, "Basis for Plugging Degraded PWR Steam Generator Tubes". The designated 1.25 inches of sound roll expansion was arrived at by incorporating the length of roll required for structural integrity (1.01 inches) plus an allowance for eddy current testing uncertainty (0.24 inches at a 99% confidence level).

Basis for No Significant Hazards Determination:

In accordance with the requirements of 10 CFR 50.92, the proposed Technical Specification change is deemed to involve no significant hazards considerations because operation of Indian Point Unit No. 2 would not:

1) Involve a significant increase in the probability or consequences of an accident previously evaluated since the integrity of the steam generator tubes after sleeving will be equivalent to that of the original tubes. The sleeve, sleeve joint, and F* joint have been analyzed and tested for design, operating, and faulted condition loadings in accordance with NRC Regulatory guide 1.121 safety factors. The potential for a tube rupture is not increased with sleeving or F*. At worst case, a tube leak would occur, resulting in a small primary to secondary leak. Primary to secondary leakage occurring from within the sleeved or F* portions of the tube is bounded by the steam generator tube rupture scenario evaluated in the Final Safety Analysis Report. In addition, the steam generator tube remains capable of performing its required heat transfer function. Placing a sleeve in the steam generator tube or leaving a tube in service with a defect in a portion of the tube that provides no function results in a more efficient steam generator than plugging an affected tube. Thus, the consequences of any accident previously evaluated are not increased because the structural integrity and the heat transfer capability of the steam generators are not significantly altered by the proposed change.

2) Create the possibility of a new or different kind of accident from any accident previously evaluated because both the structural integrity and the heat transfer capability of the steam generators will not be significantly affected by the use of either of the sleeving processes or the implementation of the F* criteria. Testing and previous experience indicate that any primary to secondary leakage would be well below technical specification limits. In addition, in the unlikely event the defective tube failed completely at the defect, the remaining sleeve end or F* joint would restrain tube movement due to the sleeve end geometry or length of expanded contact within the tubesheet bore. Therefore, there is no threat to adjacent tubes and no other plant systems will be affected by this change. Thus, there is no potential for a new or different kind of accident.

3) Involve a significant reduction in a margin of safety. The heat transfer capabilities of Indian Point 2 Steam Generators will be improved by utilizing the proposed sleeving process or implementing the F* criteria rather than the currently required tube plugging and subsequent loss of heat transfer area. The proposed change will allow a repaired (sleeved) tube or a tube with a tube end defect below the F* distance to remain in service, rather than completely blocking the tube's flow with plugs. Because the structural integrity of the tubes will be unaltered, the net effect of implementing the proposed change, rather than the currently required plugging procedure, will be an increase in the heat transfer characteristics of the steam generator. Westinghouse has done an evaluation of selected LOCA and non-LOCA transients to verify that use of sleeves resulting in a plugging equivalency at the current plant limit will not have an adverse affect on the thermal-hydraulic performance of the plant. Therefore, the margin of safety is not reduced.

The proposed changes have been reviewed by both the Station Nuclear Safety Committee and the Con Edison Nuclear Facilities Safety Committee. Both Committees concur that the proposed changes do not represent a significant hazards consideration.