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December 11, 2000

Re: Indian Point Unit No. 2
Docket No. 50-247
NL 00-147

US Nuclear Regulatory Commission
ATTN: Document Control Desk
Mail Station P1-137
Washington, DC 20555-0001

Subject: Proposed Technical Specification Amendment - Changes to Primary to Secondary Leakage Limits and Steam Generator Tube Inservice Surveillance Requirements

Transmitted herewith is an "Application for Amendment to the Operating License," sworn on December 11, 2000. This application requests an amendment to the Consolidated Edison Company of New York, Inc. (Con Edison), Indian Point Unit No. 2 Technical Specifications (TS).

Con Edison is currently replacing the steam generators at Indian Point Unit No. 2. This proposed license amendment deletes provisions in the existing TS that are not necessary and are no longer applicable after the installation of the replacement steam generators. These obsolete provisions are associated with the steam generator tube "denting" phenomena, unique requirements for F* steam generator tubes, and tube sleeving as a repair method for steam generator tubes. This proposed amendment also establishes a more conservative primary to secondary leakage limit that is consistent with industry guidelines. This amendment request is not related to the startup of the unit.

Separate from this request, Con Edison is preparing a license amendment request associated with steam generators that will adopt the regulatory framework for Nuclear Electric Institute (NEI) 97-06, "Steam Generator Program Guidelines."

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Attachment I to this letter provides the proposed changes to Specifications 3.1.F.2.a, "Primary to Secondary Leakage," and 4.13.A.3.f, "Steam Generator Tube Inservice Surveillance." Attachment II provides a copy of the proposed changes as markups on the affected existing Technical Specification (TS) pages, Attachment III provides the Safety Assessment.

The proposed changes have been reviewed by the Station Nuclear Safety Committee (SNSC) and the Nuclear Facility Safety Committee (NFSC). Both committees concur the proposed changes do not represent a significant hazards consideration as defined by 10 CFR 50.92(c).

Con Edison requests NRC approval of this request by May 31, 2001 and that the amendment provide an allowance for implementation within 31 days after receipt of the approved amendment.

In accordance with 10 CFR 50.91, a copy of this application and the associated attachments are being submitted to the designated New York State official.

No new regulatory commitments are made by Con Edison in this correspondence.

Should you or your staff have any questions regarding this submittal, please contact Mr. John F. McCann, Manager, Nuclear Safety and Licensing.

Very truly yours,

Enclosure and Attachments

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UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

In the Matter of CONSOLIDATED EDISON COMPANY OF NEW YORK, INC. (Indian Point Station, Unit No. 2)	))))	Docket No. 50-247

APPLICATION FOR AMENDMENT
TO OPERATING LICENSE

Pursuant to Section 50.90 of the Regulations of the Nuclear Regulatory Commission (NRC), Consolidated Edison Company of New York, Inc. (Con Edison), as holder of Facility Operating License No. DPR-26 hereby applies for amendment of the Technical Specifications (TS) contained in Appendix A of this license.

This Application for amendment to the Indian Point 2 Technical Specifications seeks to propose changes to Technical Specification (TS) 3.1.F.2.a, "Primary to Secondary Leakage," and 4.13.A.3.f, "Steam Generator Tube Inservice Surveillance."

The specific changes to the TS are as follows:

a) Change TS 3.1.F.2.a.1 to eliminate separate treatment of primary to secondary leakage for sleeved and non-sleeved tubes. Additionally, the primary to secondary leakage limit is reduced from 0.3 gallons per minute (432 gallons per day) to 150 gallons per day in any steam generator. The associated TS Basis discussion is modified to incorporate the changes to the specification,
b) Change definition in specification 4.13.A.1.d to delete verbiage associated with sleeved tubes,
c) Change definition in specification 4.13.A.1.e to delete verbiage associated with sleeves,
d) Change definition in specification 4.13.A.1.f to delete verbiage associated with sleeves,
e) Delete definition in specification 4.13.A.1.l regarding sleeving,
f) Delete specification 4.13.A.3.c regarding sampling requirements for sleeved tubes,
g) Change specification 4.13.A.4.1.a to delete verbiage regarding degraded or defective tube sleeves,
h) Delete specification 4.13.A.4.1.d regarding examination requirements for sleeved tubes,
i) Change specification 4.13.B.1.a to eliminate limit for minimum sleeve wall thickness,
j) Change specification 4.13.B.2 to eliminate verbiage regarding sleeves,
k) Delete definition in specification 4.13.A.1.j regarding F* Distance,
l) Delete definition in specification 4.13.A.1.k regarding F* Tube,
m) Delete specification 4.13.A.3.g regarding minimum sample size for F* Tubes and provisions regarding non-utilization of F* tube inspections for additional examinations,
n) Delete specification 4.13.B.1.c regarding acceptability of F* tubes,
o) Change Basis to TS 4.13 to delete paragraphs associated with F* repair criteria,

p) Delete the first sentence of specification 3.1.F.2.a.2 that provided requirements regarding leaks within multiple steam generators within a 20 day period and delete the second sentence of specification 3.1.F.2.a.2 that provided requirements regarding leaks within multiple steam generators attributable to the denting phenomena identified after the reactor is in cold shutdown,
q) Delete specification 4.13.A.2.e regarding profilometry tensile strain criterion for unscheduled steam generator examinations,
r) Delete specification 4.13.A.3.e regarding examination requirements for tube deformation ("dents"),
s) Change CTS 4.13.A.3.f to delete the exclusion for examinations for deformation by eddy current methodology and to delete a provision permitting use of a 540 mil probe with profilometry and tensile strain criteria for examination of tubes in rows 2 through 5,
t) Delete specification 4.13.A.4.2 regarding additional examinations for degradation caused by denting and modify 4.13.A.4.1 to delete exclusion statement for degradation examination for denting,
u) Change CTS 4.13.B.1.b to delete the allowance for tubes that will not pass a 610 mil probe based upon passage of a 540 mil probe combined with tensile strain criteria,
v) Change specification 4.13.C.2 to eliminate reporting requirement associated with increase in denting rate and significant changes in steam generator condition,
w) Delete specification 4.13.C.3 requiring submittal of an evaluation upon discovery of significant hour glassing (closure) of the upper support plate flow slots,
x) Change Basis to TS 4.13 to delete paragraphs associate with the denting phenomena and delete an obsolete statement,
y) Change to the Basis section for 3.1.F to correct left margin alignment for text following list item e, and
z) Delete obsolete footnote associated with specification 4.13.A.2.a regarding a historical scheduler extension for examinations.

The specific proposed Technical Specification Revisions are set forth in Attachment I to this Application. Attachment II provides a copy of the proposed changes as markups on the affected existing Technical Specification (TS) pages, Attachment III provides the Safety Assessment. This assessment demonstrates that the proposed changes do not represent a significant hazards consideration as defined in 10 CFR 50.92(c).

As required by 10 CFR 50.91(b)(1), a copy of this Application and our analysis concluding that the proposed changes do not constitute a significant hazards consideration have been provided to the appropriate New York State official designated to receive such amendments.

BY: [signature]
J. Baumstark
Vice President - Nuclear Engineering

Subscribed and sworn to
December 11, 2000.

Karen L. Lancaster
Notary Public

KAREN L. LANCASTER
Notary Public, State of New York
No. 60-4643659
Qualified In Westchester County
Term Expires 9/30/01

ATTACHMENT I

PROPOSED TECHNICAL SPECIFICATION CHANGES

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

ATTACHMENT II

PROPOSED TECHNICAL SPECIFICATION MARKED-UP PAGES

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

2. Operational Leakage Limits

a. Primary to Secondary Leakage

(1) Primary to secondary leakage through the steam generator tubes shall not exceed 150 gpd 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) DELETED

(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.

e. Water may also collect in the recirculation sump and/or the reactor cavity depending on the size and location of the leak. However, under most circumstances, the containment sump will be filled prior to the recirculation sump filling and both sumps will be filled prior to water level increasing on the containment floor (EL. 46') sufficient to initiate filling of the reactor cavity. Level monitoring of the recirculation sump is provided by two level instruments which actuate control room lights at discrete sump/containment water levels and provide an audible alarm for certain discrete levels within the recirculation sump. In addition, another level monitoring device provides a continuous level readout in the control room. Level monitoring of the reactor cavity is provided by a single analog continuous level indication in the control room and by two separate and independent level switches, each of which actuates an audible alarm in the control room.

Total reactor coolant leakage can be determined by means of periodic water inventory balances. If leakage is into another closed system, it will be detected by the plant radiation monitors and/or inventory balances. Determined leakage rates are an average over the applicable surveillance interval. Industry experience has shown that while a limited amount of leakage is expected from the RCS, the unidentified portion of this leakage can be reduced to a threshold value of less than 1 gpm. This threshold value is sufficiently low to ensure detection of additional leakage.

The 10 gpm identified leakage limitation provides allowance for a limited amount of leakage from known sources whose presence will not interfere with the detection of unidentified leakage by the leakage detection systems.

Pressure boundary leakage of any magnitude is unacceptable since it may be indicative of an impending gross failure of the pressure boundary. Therefore, the presence of any pressure boundary leakage requires the unit to be promptly placed in cold shutdown. Primary system leakage through packing, gaskets, seal welds or mechanical joints is not considered to be pressure boundary leakage.

The leakage limit and surveillance testing for RCS/RHR Pressure Isolation Valves provide added assurance of valve integrity, thereby reducing the probability of gross valve failure and consequent intersystem LOCA.

Leakage from the RCS/RHR Pressure Isolation Valves is identified leakage and will be considered as a portion of the allowed limit.

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 limitation of steam generator leakage between the reactor coolant system and the secondary coolant system. The allowable primary to secondary leakage rate of 150 gpd in any steam generator is based on industry operating experience.

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.

The above actions are necessary to (1) preclude accumulation of water inside containment so that if a LOCA were to occur safety-related equipment would not become submerged, (2) prevent the reactor cavity from becoming filled with water, (3) prevent the reactor vessel from being wetted while it is at an elevated temperature, and (4) prevent the immersion of the in-core instrument conduits. The amount of water estimated to be inside containment after actuation of the emergency core cooling system following a loss of coolant accident is approximately 423,000 gallons. This amount of water would, by itself, reach approximately EL. 50' 1". An additional 28,000 gallons (a total of approximately 451,000 gallons) would have to accumulate inside containment before any safety-related electrical component would be submerged (approximately EL. 50' 5"). The combined volume of the containment sump, the recirculation sump and the containment floor trenches is approximately 18,000 gallons. Since operator action is required by these specifications to shut the reactor down before these volumes are filled, sufficient margin between the water level inside containment following a loss of coolant accident and the level at which a safety-related electrical component may become submerged is maintained. Furthermore, since both sumps, the floor trenches and the containment floor up to EL. 46' 5 3/8" must be flooded (i.e., approximately 50,000 gallons) before the water level is sufficiently high to flood over the curb leading to the reactor cavity, the forementioned operator actions taken to preclude excessive flooding plus LOCA water levels will conservatively preclude flooding of the reactor cavity and subsequent wetting of the reactor vessel at an elevated temperature.

<u>References</u>

UFSAR Sections 6.7, 11.2.3 and 14.2.4

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

e. % Degradation is an estimated % of the tube wall thickness affected or removed by degradation.

f. Defect is a degradation of such severity that it exceeds the plugging limit. A tube containing a defect is defective.

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

h. Hot-Leg Tube Examination 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. Cold-Leg Tube Examination 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.

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.

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. DELETED

d. 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.

e. DELETED

Examination shall be by eddy current techniques. A 700-mil diameter probe shall be used unless previous data indicates that a 700-mil diameter probe would not pass through the tube. If the 700-mil diameter probe cannot pass through the tube, the largest probe size that is expected to pass through the tube shall be used. In all case a probe with at least a 610-mil diameter shall be used.

4. Additional Examination Criteria

1. Degradation

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.

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

   AND

      the tube will permit passage of a 0.610" diameter probe.

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

C. REPORTS AND REVIEW OF RESULTS

   1. The proposed steam generator examination program shall be submitted for NRC staff review 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.

   3. DELETED

   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.

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.

A minor diameter of 0.610" is established as the criterion for continuing a tube inservice if denting is occuring. This criterion has been 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.

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

On these marked-up pages from the current Technical Specifications;

Additions are presented in bold italic, [***Addition***]

And

Deletions are presented in double strikethrough. [~~Deletion~~]

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 which does not contain tube sleeves. Primary to secondary leakage through the steam generator tubes and/or sleeves shall not exceed~~ 150 gpd in any steam generator~~containing sleeves~~. With any steam generator tube leakage greater than this limit, the reactor shall be brought to the cold shutdown condition within 24 hours.

(2) ***DELETED*** ~~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.

e. Water may also collect in the recirculation sump and/or the reactor cavity depending on the size and location of the leak. However, under most circumstances, the containment sump will be filled prior to the recirculation sump filling and both sumps will be filled prior to water level increasing on the containment floor (EL. 46') sufficient to initiate filling of the reactor cavity. Level monitoring of the recirculation sump is provided by two level instruments which actuate control room lights at discrete sump/containment water levels and provide an audible alarm for certain discrete levels within the recirculation sump. In addition, another level monitoring device provides a continuous level readout in the control room. Level monitoring of the reactor cavity is provided by a single analog continuous level indication in the control room and by two separate and independent level switches, each of which actuates an audible alarm in the control room.

Total reactor coolant leakage can be determined by means of periodic water inventory balances. If leakage is into another closed system, it will be detected by the plant radiation monitors and/or inventory balances. Determined leakage rates are an average over the applicable surveillance interval. Industry experience has shown that while a limited amount of leakage is expected from the RCS, the unidentified portion of this leakage can be reduced to a threshold value of less than 1 gpm. This threshold value is sufficiently low to ensure detection of additional leakage.

The 10 gpm identified leakage limitation provides allowance for a limited amount of leakage from known sources whose presence will not interfere with the detection of unidentified leakage by the leakage detection systems.

Pressure boundary leakage of any magnitude is unacceptable since it may be indicative of an impending gross failure of the pressure boundary. Therefore, the presence of any pressure boundary leakage requires the unit to be promptly placed in cold shutdown. Primary system leakage through packing, gaskets, seal welds or mechanical joints is not considered to be pressure boundary leakage.

The leakage limit and surveillance testing for RCS/RHR Pressure Isolation Valves provide added assurance of valve integrity, thereby reducing the probability of gross valve failure and consequent intersystem LOCA.

Leakage from the RCS/RHR Pressure Isolation Valves is identified leakage and will be considered as a portion of the allowed limit.

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 limitation of steam generator leakage between the reactor 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 ~~lower~~ allowable ***primary to secondary*** leak***age*** rate of 150 gpd ***in any*** ~~for a~~ steam generator ~~containing sleeved tubes~~ is based on industry operating experience.

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.

The above actions are necessary to (1) preclude accumulation of water inside containment so that if a LOCA were to occur safety-related equipment would not become submerged, (2) prevent the reactor cavity from becoming filled with water, (3) prevent the reactor vessel from being wetted while it is at an elevated temperature, and (4) prevent the immersion of the in-core instrument conduits. The amount of water estimated to be inside containment after actuation of the emergency core cooling system following a loss of coolant accident is approximately 423,000 gallons. This amount of water would, by itself, reach approximately EL. 50' 1". An additional 28,000 gallons (a total of approximately 451,000 gallons) would have to accumulate inside containment before any safety-related electrical component would be submerged (approximately EL. 50' 5"). The combined volume of the containment sump, the recirculation sump and the containment floor trenches is approximately 18,000 gallons. Since operator action is required by these specifications to shut the reactor down before these volumes are filled, sufficient margin between the water level inside containment following a loss of coolant accident and the level at which a safety-related electrical component may become submerged is maintained. Furthermore, since both sumps, the floor trenches and the containment floor up to EL. 46' 5 3/8" must be flooded (i.e., approximately 50,000 gallons) before the water level is sufficiently high to flood over the curb leading to the reactor cavity, the forementioned operator actions taken to preclude excessive flooding plus LOCA water levels will conservatively preclude flooding of the reactor cavity and subsequent wetting of the reactor vessel at an elevated temperature.

References

UFSAR Sections 6.7, 11.2.3 and 14.2.4

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. % Degradation is an estimated % of the tube ~~or sleeve~~ wall thickness affected or removed by degradation.

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

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

h. Hot-Leg Tube Examination 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. Cold-Leg Tube Examination 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. F* Distance 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. F* Tube is a tube:~~

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

~~l. Sleeving refers to tube repair achieved by 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.

~~* Examinations scheduled for 1999 only, shall be conducted during the 2000 Refueling Outage which will commence no later than June 3, 2000. The scheduled examinations will be completed prior to return to service from the 2000 Refueling Outage.~~

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 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. ***DELETED*** ~~At least 20% of a random sample of tubes containing sleeves shall be subjected to an examination throughout the sleeved portion of the tube.~~

d. 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.

e. ***DELETED*** ~~Examination for deformation ("dents") shall be either by eddy current or by profilometry.~~

f. Examination ~~for degradation other than deformation~~ shall be by eddy current techniques. A 700-mil diameter probe shall be used unless previous data indicates that a 700-mil diameter probe would not pass through the tube. If the 700-mil diameter probe cannot pass through the tube, the largest probe size that is expected to pass through the tube shall be used. In all cases a probe with at least a 610-mil diameter shall be used~~, except for the examination of the U-bends and the cold legs of tubes in rows 2 through 5. For these examinations, a 540-mil diameter probe may be used, provided it is justified by a profilometry measurement within the tensile strain criterion~~.

~~g. 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. Additional Examination Criteria

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 ~~with the exception of degraded or defective tube sleeves~~.

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.

~~d. If a tube sleeve exhibits degradation of greater than 23% or is otherwise defective, an additional 20% (minimum) of the unsampled sleeves shall be examined. If a sleeve exhibits degradation of greater than 23% or is otherwise defective in the second sample, all remaining sleeves shall be examined.~~

~~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.~~

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 the F* region.~~

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

C. REPORTS AND REVIEW OF RESULTS

   1. The proposed steam generator examination program shall be submitted for NRC staff review 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. ***DELETED*** ~~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.~~

~~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~~ **A** minor diameter of 0.610" ~~was~~ **is** established as the criterion for continuing a tube inservice **if denting is occurring**. This criterion ~~was~~ **has been** used successfully for several years at Indian Point Unit 2 and at other plants, and ~~appears to be~~ **is** 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 degradation 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.~~

~~Testing performed as documented in BAW-10195 P demonstrates the maximum postulated leakage under accident conditions for repair of 100% of the tube ends using the F* criteria is well below the allowable leakage limits for Indian Point 2 steam generators. If, in the future, steam generator tubes are allowed to remain in service by the use of F* and, in addition, other tube acceptance criteria, then the aggregate maximum postulated accident leakage must be below the allowable leakage limits for Indian Point 2 steam generators.~~

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

ATTACHMENT III

SAFETY ASSESSMENTS

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

Introduction

In this Attachment, separate assessments are provided, one for each of the following five categories of changes:

1. Change to Primary to Secondary Leakage Limit

   This category includes the following discrete change to the TS.

   a) Change TS 3.1.F.2.a.1 to eliminate separate treatment of primary to secondary leakage for sleeved and non-sleeved tubes. Additionally, the primary to secondary leakage limit is reduced from 0.3 gallons per minute (432 gallons per day) to 150 gallons per day for any steam generator. The associated TS Basis discussion is modified to incorporate the changes to the specification.

2. Deletion of Provisions Associated With Steam Generator Tube Sleeving Repair Method

   This category includes the following discrete changes to the TS.

   b) Change definition in specification 4.13.A.1.d to delete verbiage associated with sleeved tubes,
   c) Change definition in specification 4.13.A.1.e to delete verbiage associated with sleeves,
   d) Change definition in specification 4.13.A.1.f to delete verbiage associated with sleeves,
   e) Delete definition in specification 4.13.A.1.l regarding sleeving,
   f) Delete specification 4.13.A.3.c regarding sampling requirements for sleeved tubes,
   g) Change specification 4.13.A.4.1.a to delete verbiage regarding degraded or defective tube sleeves,
   h) Delete specification 4.13.A.4.1.d regarding examination requirements for sleeved tubes,
   i) Change specification 4.13.B.1.a to eliminate limit for minimum sleeve wall thickness, and
   j) Change specification 4.13.B.2 to eliminate verbiage regarding sleeves.

3. Deletion of Provisions Associated With Steam Generator F* Tube Classification

   This category includes the following discrete changes to the TS.

   k) Delete definition in specification 4.13.A.1.j regarding F* Distance,
   l) Delete definition in specification 4.13.A.1.k regarding F* Tube,
   m) Delete specification 4.13.A.3.g regarding minimum sample size for F* Tubes and provisions regarding non-utilization of F* tube inspections for additional examinations,
   n) Delete specification 4.13.B.1.c regarding acceptability of F* tubes, and
   o) Change Basis to TS 4.13 to delete paragraphs associate with F* repair criteria.

4. Deletion of Provisions Associated With Steam Generator Tube Denting Phenomenon

This category includes the following discrete changes to the TS.

p) Delete the first sentence of specification 3.1.F.2.a.2 that provided requirements regarding leaks within multiple steam generators within a 20 day period and delete the second sentence of specification 3.1.F.2.a.2 that provided requirements regarding leaks within multiple steam generators attributable to the denting phenomena identified after the reactor is in cold shutdown,
q) Delete specification 4.13.A.2.e regarding profilometry tensile strain criterion for unscheduled steam generator examinations,
r) Delete specification 4.13.A.3.e regarding examination requirements for tube deformation ("dents"),
s) Change CTS 4.13.A.3.f to delete the exclusion for examinations for deformation by eddy current methodology and to delete a provision permitting use of a 540 mil probe with profilometry and tensile strain criteria for examination of tubes in rows 2 through 5,
t) Delete specification 4.13.A.4.2 regarding additional examinations for degradation caused by denting and modify 4.13.A.4.1 to delete exclusion statement for degradation examination for denting,
u) Change CTS 4.13.B.1.b to delete the allowance for tubes that will not pass a 610 mil probe based upon passage of a 540 mil probe combined with tensile strain criteria,
v) Change specification 4.13.C.2 to eliminate reporting requirement associated with increase in denting rate and significant changes in steam generator condition,
w) Delete specification 4.13.C.3 requiring submittal of an evaluation upon discovery of significant hour glassing (closure) of the upper support plate flow slots, and
x) Change Basis to TS 4.13 to delete paragraphs associate with the denting phenomena and delete an obsolete statement.

5. Administrative Changes

This category includes the following discrete changes to the TS.

y) Change to the Basis section to 3.1.F to correct left margin alignment for text following list item e, and
z) Delete obsolete footnote associated with specification 4.13.A.2.a regarding a historical scheduler extension for examinations.

CHANGE TO PRIMARY TO SECONDARY LEAKAGE LIMITS

SECTION I – Description of Change

Current Technical Specification (CTS) 3.1.F.2.a.1 provides separate primary to secondary leakage limits for steam generators (SG) containing sleeved tubes and steam generators not containing sleeved tubes. The leakage limit for a steam generator containing sleeved tubes is 150 gallons per day (gpd) and the leakage limit for SGs not containing sleeves is 0.3 gallons per minute (gpm) (432 gpd). The proposed change deletes the separate leakage limit for SGs containing sleeved tube and reduces the limit for SGs primary to secondary leakage from 432 gpd to 150 gpd for any SG. The associated Basis is modified to reflect the change to the specification.

SECTION II – Evaluation of Change

The deletion of the separate primary to secondary leakage limit for sleeved tubes and the adoption of the 150 gpd primary to secondary leakage limit for any SG are more restrictive requirements. Consolidated Edison (Con Edison) is currently replacing the SGs at Indian Point No. 2 (IP2). There is no need for a separate primary to secondary leakage limit that is applicable for SG containing sleeved tubes. The adoption of the 150 gpd limit for any SG is based on industry operating experience gained from tube degradation mechanisms which result in tube leakage. This leakage criteria along with other performance criteria in the SG Program provide reasonable assurance that any flaw leaking this amount will not propagate to a SG Tube Rupture under the stress conditions of a Loss of Coolant Accident (LOCA) or a main steam line rupture prior to detection by leakage monitoring methods and commencement of plant shutdown. If leaked through many flaws, the flaws are very small and the above assumption is conservative. The associated Basis change is necessary to reflect the changes to the specifications.

SECTION III – No significant Hazards Evaluation

Con Edison has evaluated this proposed change and determined that it does not involve a significant hazard. According to 10 CFR 50.92(c), a proposed amendment to an operating license involves no significant hazards consideration if operation of the facility in accordance with the proposed amendment would not:

1. involve a significant increase in the probability of occurrence or consequences of an accident previously evaluated;

2. create the possibility of a new or different kind of accident from any previously analyzed; or

3. involve a significant reduction in a margin of safety.

The proposed change revises the Technical Specification (TS) to reduce the limit for allowable primary to secondary leakage for any SG and to eliminate the separate primary to secondary leakage limits for SGs with tube sleeves.

The determination that the criteria set forth in 10 CFR 50.92 are met for this amendment request is indicated below.

1. Does the change involve a significant increase in the probability of occurrence or consequences of an accident previously evaluated?

The proposed reduction in primary to secondary leakage limit and the elimination of the limit for SGs containing sleeved tubes does not affect accident initiators or precursors. The proposed change establishes a primary to secondary leakage limit that is equivalent to the lesser of the primary to second leakage limits currently established for SG with and without SG tube sleeves. Reducing the primary to secondary leakage limit does not increase the probability of an accident. The proposed does not increase primary to secondary leakage limits. Therefore, the consequences of an accident are not increased. Therefore, the probability of occurrence or the consequences of accidents previously evaluated are not significantly increased.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed changes do not modify any plant equipment. Therefore, the proposed changes do not degrade the reliability of systems, structures, or components or create a new accident initiator or precursor. No new failure modes are created. Therefore, the change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the change involve a significant reduction in a margin of safety?

The proposed change establishes one limit for primary to secondary limit that is the same as the most restrictive of the two primary to secondary leakage limits that currently exists. The proposed change does not increase the allowable primary to secondary leakage limit.

Since the primary to secondary leakage limit is not increased, the margin of safety will not be reduced. The proposed change still requires verification that primary to secondary leakage is within the limit at the existing frequency. Since the primary to secondary leakage limit is not increased, dose rates at the site boundary will not be increased. Therefore, the proposed activity does not involve a significant reduction in a margin of safety.

In summary, based upon the above evaluation, Con Edison has concluded that the proposed amendment involves no significant hazards consideration.

DELETION OF PROVISIONS ASSOCIATED WITH STEAM GENERATOR TUBE SLEEVING REPAIR METHOD

SECTION I – Description of Changes

CTS 4.13.A.1.d provides a definition of degraded tube that includes sleeved tubes. The proposed change deletes sleeved tube from this definition. CTS 4.13.A.1.e provides a definition of % degradation that includes sleeve wall thickness. The proposed change deletes sleeve wall thickness from this definition. CTS 4.13.A.1.f provides a definition of defect that includes tube sleeves. The proposed change deletes tube sleeves from this definition. CTS 4.13.A.1.l provides a definition of sleeving. The proposed change deletes this definition.

CTS 4.13.A.3.c provides sampling and selection criterion for sleeved tubes. The proposed change deletes this sampling and selection criterion.

CTS 4.13.A.4.1.a provides criteria that establish additional examination requirements based upon the results of initial tube examinations. This additional examination criterion excludes sleeved tubes since the requirements for sleeved tubes are specified elsewhere. CTS 4.13.A.4.1.d specifies the additional examination criteria for sleeved tubes. The CTS 4.13.A.4.1.d criteria for additional examination of sleeved tubes as well as the CTS 4.13.A.4.1.a exclusion statement for non-sleeved tubes are deleted.

CTS 4.13.B.1.a includes criteria for tubes that are acceptable for use. These criteria include a criterion regarding sleeve wall thickness. CTS 4.13.B.2 specifies the required action for tubes or sleeves that are not acceptable for use. The proposed change deletes the CTS 4.13.B.1 criterion associated with acceptable sleeve wall thickness and the CTS 4.13.B.2 required action for unacceptable sleeves.

SECTION II – Evaluation of Changes

The use of sleeving was an approved repair method for the original IP2 SGs. The provisions that are proposed for deletion are associated with the use of the sleeving repair method. Con Edison is currently replacing the SGs at IP2.

The use of tube sleeves has not been approved as a repair method for the IP2 replacement SGs. The change deletes TS requirements that are not applicable to the IP2 replacement SGs. Therefore, this change proposes deletion of the sleeving provisions in these specifications.

SECTION III – No significant Hazards Evaluation

Con Edison has evaluated this proposed change and determined that it does not involve a significant hazard. According to 10 CFR 50.92(c), a proposed amendment to an operating license involves no significant hazards consideration if operation of the facility in accordance with the proposed amendment would not:

1. involve a significant increase in the probability of occurrence or consequences of an accident previously evaluated;
2. create the possibility of a new or different kind of accident from any previously analyzed; or
3. involve a significant reduction in a margin of safety.

The proposed change deletes the TS allowance and associated provisions that permit use of sleeving as a repair method for the SG tubes. The NRC has not approved the use of tube sleeving as a repair method for the IP2 replacement steam generators. The change deletes TS requirements that are not applicable to the IP2 replacement SGs.

The determination that the criteria set forth in 10 CFR 50.92 are met for this amendment request is indicated below.

1. Does the change involve a significant increase in the probability of occurrence or consequences of an accident previously evaluated?

The proposed deletion of the SG tube sleeving provisions does not affect accident initiators or precursors. The proposed change deletes the TS provisions that are not approved for the replacement SGs. Deletion of an unapproved repair method from the TS does not increase the probability of an accident and the proposed change does not increase primary to secondary leakage limits. Consequently, the consequences of an accident are not significantly increased. Therefore, the probability of occurrence or the consequences of accidents previously evaluated are not significantly increased.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed changes do not impact or interface with plant safety related equipment. Therefore, the proposed changes do not degrade the reliability of systems, structures, or components or create a new accident initiator or precursor. No new failure modes are created. Therefore, the change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the change involve a significant reduction in a margin of safety?

The proposed change deletes the TS provisions that are not approved for the replacement SGs. The proposed change does not increase the allowable primary to secondary leakage limit. Since the primary to secondary leakage limit is not increased, the margin of safety will not be reduced. Therefore, the proposed activity does not involve a significant reduction in a margin of safety.

In summary, based upon the above evaluation, Con Edison has concluded that the proposed amendment involves no significant hazards consideration.

DELETION OF PROVISIONS ASSOCIATED WITH STEAM GENERATOR F* TUBE CLASSIFICATION

SECTION I – Description of Changes

CTS 4.13.A.1.j provides a definition of F* Distance. CTS 4.13.A.1.k provides a definition of F* Tube. The proposed change deletes both of these definitions.

CTS 4.13.A.3.g specifies specific examination requirements for F* Tubes in addition to the examination requirements specified in Table 4.13-1. This specification also excludes results of the F* examinations as a basis for additional inspections per Table 4.13-1. The proposed change deletes these provisions.

CTS 4.13.B.1.c provides an additionally acceptance criteria for continued use of F* Tubes. The proposed change deletes these criteria.

The eighth and ninth paragraphs to the Basis for TS 4.13 that provide information associated with the F* repair criteria are also deleted.

SECTION II – Evaluation of Changes

The use of the F* Criteria as a repair method was approved for the original IP2 SGs. The provisions that are proposed for deletion are requirements that are specifically associated with the use of F* Criteria as a repair method. The repair method using the F* criteria provides an alternative method for repairing SG tube defects found in the tube expansion region within the tubesheet.

The use of F* Criteria as a repair method has not been approved for the IP2 replacement SGs. The change deletes TS requirements that are not applicable to the IP2 replacement SGs. Therefore, this change proposes deletion of the F* Criteria as a repair method in these specifications as well as the associated TS Basis paragraphs.

SECTION III – No significant Hazards Evaluation

Con Edison has evaluated this proposed change and determined that it does not involve a significant hazard. According to 10 CFR 50.92(c), a proposed amendment to an operating license involves no significant hazards consideration if operation of the facility in accordance with the proposed amendment would not:

1. involve a significant increase in the probability of occurrence or consequences of an accident previously evaluated;

2. create the possibility of a new or different kind of accident from any previously analyzed; or

3. involve a significant reduction in a margin of safety.

The proposed change revises the TS to delete the TS allowance and associated provisions that

permit use of the F* criteria as a repair method for the SG tubes. The use of the F* criteria as a repair method has not been approved for the IP2 replacement SGs. The change deletes TS requirements that are no longer needed or applicable to the IP2 replacement SGs.

The determination that the criteria set forth in 10 CFR 50.92 are met for this amendment request is indicated below.

1. Does the change involve a significant increase in the probability of occurrence or consequences of an accident previously evaluated?

The proposed deletion of the F* criteria and associated provisions does not affect accident initiators or precursors. The proposed change deletes the TS provisions that are not approved for the replacement SGs. Deletion of an unapproved repair method from the TS does not increase the probability of an accident. The proposed change does not increase primary to secondary leakage limits. Consequently, the consequences of an accident are not significantly increased. Therefore, the probability of occurrence or the consequences of accidents previously evaluated are not significantly increased.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed changes do not impact or interface with plant safety related equipment. Therefore, the proposed changes do not degrade the reliability of systems, structures, or components or create a new accident initiator or precursor. No new failure modes are created. Therefore, the change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the change involve a significant reduction in a margin of safety?

The proposed change deletes the TS provisions that are not approved for the replacement SGs. The proposed change does not increase the allowable primary to secondary leakage limit. Since the primary to secondary leakage limit is not increased, the margin of safety will not be reduced. Therefore, the proposed activity does not involve a significant reduction in a margin of safety.

In summary, based upon the above evaluation, Con Edison has concluded that the proposed amendment involves no significant hazards consideration.

DELETION OF PROVISIONS ASSOCIATED WITH STEAM GENERATOR TUBE DENTING PHENOMENON

SECTION I – Description of Changes

CTS 3.1.F.2.a.2 imposes restrictions upon unit operation for leaks within multiple SGs within a 20 day period and requirements regarding leaks within multiple SGs attributable to the denting phenomena identified after the reactor is in cold shutdown. The proposed change deletes these requirements.

CTS 4.13.A.2.e requires profilometry tensile strain criteria for unscheduled examinations to be the same as contained in the program for the last scheduled inspection. CTS 4.13.A.3.e allows examination requirements for tube deformation ("dents") to be either eddy current or profilometry. The proposed change deletes these provisions. With the deletion of profilometry measurements, tube examination will be eddy current examination.

CTS 4.13.A.3.f includes a provision permitting use of a 540 mil probe with profilometry and tensile strain criteria for examination of U-bends and tubes in rows 2 through 5 for degradation other than deformation. The proposed change deletes this provision and eliminates the exclusion for deformation.

CTS 4.13.A.4.2 requires additional examinations for degradation caused by denting to be the same as described in the most recent SG program. CTS 4.13.B.1.b provides an allowance for tubes that will not pass a 610 mil probe based upon passage of a 540 mil probe combined with tensile strain criteria. CTS 4.13.A.4.1 provides an exclusion to the examination criteria degradation caused by denting. The proposed change deletes these provisions.

CTS 4.13.C.2 imposes reporting requirement associated with increase in denting rate or significant changes in SG condition, and CTS 4.13.C.3 requires submittal of an evaluation upon any discovery of significant hour glassing (closure) of the upper support plate flow slots. The proposed change deletes these reporting requirements.

The last sentence in the second paragraph of the Basis for TS 4.13 provides a historical reference to a 1975 SG baseline inspection that was performed prior to discontinuance of phosphate water treatment. This statement becomes obsolete with the ongoing replacement of the SGs and is deleted in the proposed change. The fifth and sixth paragraphs of the Basis for TS 4.13 that provide information associated with the denting phenomena being deleted from the specifications are also deleted.

SECTION II – Evaluation of Changes

Each of the above CTS requirements that are being deleted is associated with the SG denting phenomena. Denting (mechanical deformation of the SG tube) is a process where rapid corrosion of carbon steel tube support plates occurs at the tube-to-tube support annulus. It is the result of boiling in tube-to-support crevices causing concentration of impurities that, in turn, cause rapid corrosion of the carbon steel tube support structures. The corrosion products fill the annulus and squeeze the tube, causing dents. This process can ultimately lead to cracking from tube support dents,

deformation, and cracking at U-bends, cracking at tube support plate ligaments, and tube support plate flow slot distortions.

The deleted provisions in CTS 3.1.F.2.a.2 are requirements that were adopted to limit unit operation upon discovery of multiple tube leaks in more than one SG. CTS 4.13.A.4.2 requires additional examinations for degradation caused by denting be performed as described in the most recent SG examination program. CTS 4.13.C.2 imposes reporting requirements associated with an increase in the denting rate or significant changes in SG condition. CTS 4.13.C.3 requires submittal of an evaluation upon discovery of significant hour glassing (closure) of the upper support plate flow slots. These requirements were adopted due to concerns and uncertainties associated with the scope and magnitude of SG tube degradation resulting from denting. In the past denting was one of the dominant SG tube degradation mechanism being observed. Con Edison is currently replacing the SGs at Indian Point No. 2 (IP2). The replacement SGs include design enhancements (corrosion resistant support plate material and quatrefoil support plates) that substantially reduce the potential for occurrence of the SG denting phenomena. After installation of the replacement SGs the requirements of CTS 3.1.F.2.a.2, CTS 4.13.A.4.2, CTS 4.13.C.2, and the exclusion statement in 4.13.A.4.1 are not necessary.

The provisions of CTS 4.13.A.2.e define profilometry tensile strain criteria for unscheduled examinations. The tensile strain criteria establish limits that required severely dented tubes to be removed from service. CTS 4.13.A.3.e allows examination of dented tubes to be either profilometry or eddy current. The provision of CTS 4.13.A.3.f permits use of a 540 mil probe with profilometry and tensile strain criteria for examination of u-bends and tubes in rows 2 through 5. CTS 4.13.B.1.b provides an allowance for tubes that will not pass a 610 mil probe based upon passage of a 540 mil probe combined with tensile strain criteria. Profilometry measurements can be used to determine the geometry of the tube deformation and the magnitude of tensile strain in the dented tubes. The denting phenomena has been effectively controlled or eliminated in part by improvements in water chemistry. Improved water chemistry and the enhanced design features of the replacement SGs further reduce the potential for occurrence of the SG denting phenomena. The allowance to inspect dented tubes using profilometry is not currently used at IP2. Dented tubes are readily detectable using eddy current methodology. Therefore retention of the provisions for profilometry, tensile strain criteria and use of a 540 mil probe is unnecessary.

Con Edison is currently replacing the SGs at IP2. The replacement SGs include design enhancements (corrosion resistant support plate material, and quatrefoil support plates) that substantially reduce the potential for occurrence of the SG denting phenomena. The replacement of the SGs as well as improvements in the IP2 SG Program substantially reduces concerns regarding dented SG tubes. The proposed change deletes TS requirements that are no longer needed or applicable to the replacement IP2 SGs. Therefore this change proposes deletion of the TS provisions associated with the denting phenomena as well as the associated TS Basis text.

SECTION III – No significant Hazards Evaluation

Con Edison has evaluated this proposed change and determined that it does not involve a significant hazard. According to 10 CFR 50.92(c), a proposed amendment to an operating license involves no significant hazards consideration if operation of the facility in accordance with the proposed amendment would not:

1. involve a significant increase in the probability of occurrence or consequences of an accident previously evaluated;

2. create the possibility of a new or different kind of accident from any previously analyzed; or

3. involve a significant reduction in a margin of safety.

The proposed change revises the TS to delete the TS the requirements and associated provisions for examinations for the SG tube denting phenomena. Con Edison is currently replacing the SGs at IP2. The replacement SGs include design enhancements (corrosion resistant support plate material, and quatrefoil support plates) that substantially reduce the potential for occurrence of the SG denting phenomena. The change deletes TS requirements that are no longer needed or applicable to the IP2 SGs.

The determination that the criteria set forth in 10 CFR 50.92 are met for this amendment request is indicated below.

1. Does the change involve a significant increase in the probability of occurrence or consequences of an accident previously evaluated?

The proposed deletion of the requirements and associated provisions regarding SG tube denting does not significantly affect accident initiators or precursors. The proposed change deletes from the TS provisions that are not necessary for the replacement SGs. Deletion of the SG tube denting examination requirements from the TS does not increase the probability of an accident. The proposed change does not increase primary to secondary leakage limits. Therefore, the consequences of an accident are not increased. Therefore, the probability of occurrence or the consequences of accidents previously evaluated are not significantly increased.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed changes do not impact or interface with plant safety related equipment. Therefore, the proposed changes do not degrade the reliability of systems, structures, or components or create a new accident initiator or precursor. No new failure modes are created. Therefore, the change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the change involve a significant reduction in a margin of safety?

The proposed change deletes the TS provisions that are not applicable for the replacement SGs. The proposed change does not increase the allowable primary to secondary leakage limit. Since the primary to secondary leakage limit is not increased, the margin of safety will not be reduced. Therefore, the proposed activity does not involve a significant reduction in a margin of safety.

In summary, based upon the above evaluation, Con Edison has concluded that the proposed amendment involves no significant hazards consideration.

RELATED ADMINISTRATIVE CHANGES

SECTION I – Description of Changes

The left margin for the CTS Basis paragraphs that follow list item "e" on CTS page 3.1F-7 is incorrectly aligned with the preceding indented list. The left margin for these TS Basis paragraphs is moved to align with the left side margin. (Note: No revision bars associated with this change are included on the retyped TS pages since it involves page presentation only. There are no changes to the text content.)

CTS 4.13.A.2.a has an associated footnote that provides a one time scheduler extension for examinations scheduled in 1999. This footnote is deleted.

SECTION II – Evaluation of Changes

These changes are administrative in nature and have no technical impact. The realignment of the text margin improves readability and presentation but does not change any requirement. The footnote to CTS 4.13.A.2.a is obsolete. Deletion of the historical footnote does not alter any current requirements.

SECTION III – No significant Hazards Evaluation

Con Edison has evaluated this proposed change and determined that it does not involve a significant hazard. According to 10 CFR 50.92(c), a proposed amendment to an operating license involves no significant hazards consideration if operation of the facility in accordance with the proposed amendment would not:

1. involve a significant increase in the probability of occurrence or consequences of an accident previously evaluated;

2. create the possibility of a new or different kind of accident from any previously analyzed; or

3. involve a significant reduction in a margin of safety.

The proposed changes involve revising the presentation format for several basis pages and deleting an obsolete scheduler extension footnote. The presentation change does not alter any requirements in the specification. The deletion of the obsolete footnote eliminates a provision that is no longer needed or applicable to the replacement IP2 SGs.

The determination that the criteria set forth in 10 CFR 50.92 are met for this amendment request is indicated below.

1. Does the change involve a significant increase in the probability of occurrence or consequences of an accident previously evaluated?

The proposed administrative changes do not affect accident initiators or precursors. The proposed changes correct the presentation of several TS Basis pages and delete an obsolete

scheduler extension footnote. Correcting the page presentation and deleting an obsolete footnote do not increase the probability of an accident. The proposed change does not increase primary to secondary leakage limits. Consequently, the consequences of an accident are not significantly increased. Therefore, the probability of occurrence or the consequences of accidents previously evaluated are not significantly increased.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed changes do not impact or interface with plant safety related equipment. Therefore, the proposed changes do not degrade the reliability of systems, structures, or components or create a new accident initiator or precursor. No new failure modes are created. Therefore, the change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the change involve a significant reduction in a margin of safety?

The proposed administrative changes do not affect accident initiators or precursors. The proposed change corrects the presentation of several TS Basis pages and deletes an obsolete scheduler extension footnote. The proposed changes do not increase the allowable primary to secondary leakage limit. Since the primary to secondary leakage limit is not increased, the margin of safety will not be reduced. Therefore, the proposed activity does not involve a significant reduction in a margin of safety.

In summary, based upon the above evaluation, Con Edison has concluded that the proposed amendment involves no significant hazards consideration.