

ENCLOSURE 1

REVISED PROPOSED INDIAN POINT
NUCLEAR GENERATING UNIT NO. 2
TECHNICAL SPECIFICATIONS

CONSOLIDATED EDISON COMPANY OF NEW YORK, INC.
INDIAN POINT UNIT NO. 2
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F. REACTOR COOLANT SYSTEM LEAKAGE AND LEAKAGE IN THE CONTAINMENT FREE VOLUME

Specifications

1. Leakage Detection And Removal Systems

- a. The reactor shall not be brought above cold shutdown unless the following leakage detection and removal systems are operable:
- (1) two containment sump pumps,
 - (2) two containment sump level monitors,
 - (3) a containment sump discharge line flow monitoring system,
 - (4) two recirculation sump level monitors,
 - (5) two reactor cavity level monitors,
 - (6) two of the following three systems:
 - (a) a containment atmosphere gaseous radioactivity monitoring system,
 - (b) a containment atmosphere particulate radioactivity monitoring system,
 - (c) the containment fan cooler condensate flow monitoring system.
- b. When the reactor is above cold shutdown, the requirements of Specification 3.1.F.1.a may be modified as follows:
- (1) One containment sump pump may be inoperable for a period not to exceed seven (7) consecutive days provided that, on a daily basis, the other containment sump pump is started and discharge flow is verified.

- (2) One of the two required containment sump level monitors may be inoperable for a period not to exceed seven (7) consecutive days.
 - (3) The containment sump discharge line flow monitoring system may be inoperable for a period not to exceed seven (7) consecutive days provided a detailed Waste Holdup Tank water inventory balance is performed daily.
 - (4) One of the two required recirculation sump level monitors may be inoperable for a period not to exceed fourteen (14) consecutive days.
 - (5) One of the two required reactor cavity level monitors may be inoperable for a period not to exceed thirty (30) consecutive days.
 - (6) Two of the three monitoring systems specified in Specification 3.1.F.1.a.(6) may be inoperable for a period not to exceed thirty (30) consecutive days. If either of the radioactivity monitoring systems specified in Specification 3.1.F.1.a.(6) is inoperable, grab samples of the containment atmosphere shall be obtained and analyzed daily.
- c. If the conditions of Specification 3.1.F.1.b cannot be met or an inoperable system(s) is not restored to operable status within the time period(s) specified therein, then either perform a visual inspection of containment once a shift, or place the reactor in the hot shutdown condition within the next 6 hours and, if the inoperability continues, place the reactor in the cold shutdown condition within the following 30 hours.

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) 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.

- (b) Leakage rates greater than 5.0 gpm but less than or equal to 5.0 gpm are acceptable if the latest measured rate has not exceeded the rate determined by the previous test by an amount that reduces the margin between the measured leakage rate and the maximum permissible rate of 5.0 gpm by 50% or greater.
 - (c) Leakage rates greater than 5.0 gpm are unacceptable.
- (2) If any RCS/RHR pressure isolation valve listed in Specification 3.1.F.2.b.(1) is determined to be inoperable based on the acceptance criteria presented therein, an orderly plant shutdown shall be initiated and the reactor shall be placed in the cold shutdown condition within 24 hours.

c. Total Reactor Coolant System Leakage

- (1) Whenever the reactor is above cold shutdown, reactor coolant system leakage shall be limited to:
- (a) No pressure boundary leakage,
 - (b) 1 gpm unidentified leakage, and
 - (c) 10 gpm identified leakage.
- (2) With any pressure boundary leakage, the reactor must be placed in hot shutdown within 6 hours and in cold shutdown within the following 30 hours.
- (3) If the Reactor Coolant System leakage exceeds the limits in either c.(1)(b) or c.(1)(c) above, the leakage rate must be reduced to within limits within 4 hours or the reactor must be placed in hot shutdown within the next 6 hours and in cold shutdown within the following 30 hours.

d. Leakage Into The Containment Free Volume

- (1) Whenever the reactor is above cold shutdown, the total leakage into the containment free volume from both reactor coolant and non-reactor coolant sources combined shall not exceed 10 gpm.
- (2) Notwithstanding the action which may be required by Specification 3.1.F.2.d.(3) below, with the combined leakage into the containment free volume greater than the above limit, the leakage rate must be reduced to within the specified limit within 12 hours or the reactor must be placed in cold shutdown within the following 36 hours.
- (3) If 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 shall be placed in a cold shutdown condition within the next 36 hours unless the water level(s) is reduced below the specified limit(s).
- (4) 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", immediately place the reactor in a subcritical condition and initiate an expeditious cooldown of the reactor to the cold shutdown condition.

Basis

Water inventory balances, monitoring equipment, radioactive tracing, boric acid crystalline deposits, and physical inspections can disclose reactor coolant leaks. Any leak of radioactive fluid, whether from the reactor coolant system primary boundary or not, can be a serious problem with respect to in-plant radioactivity contamination and cleanup or it could develop into a still more serious problem; therefore, first indications of such leakage will be followed up as soon as practicable.

Although some leak rates on the order of gpm may be tolerable from a dose point of view, especially if they are to closed systems, it must be recognized that leaks on the order of drops per minute through any pressure boundary of the primary system could

be indicative of material failure such as by stress corrosion cracking. If depressurization, isolation and/or other safety measures are not taken promptly, these small leaks could develop into much larger leaks, possibly into a gross pipe rupture.

If leakage is to the containment, it may be identified by one or more of the following methods:

- a. The containment air particulate monitor is sensitive to low rates. The rates of reactor coolant leakage to which the instrument is sensitive are within the recommended sensitivity guidelines of Regulatory Guide 1.45.
- b. The containment radiogas monitor.
- c. A leakage detection system collects and measures moisture condensed from the containment atmosphere by cooling coils of the main air recirculation units including leaks from the cooling coils themselves. This system provides a dependable and accurate means of measuring the total leakage from these sources. Condensate flows from approximately 1 gpm to 15 gpm per detector can be measured by this system. Condensate flows greater than 15 gpm can be determined using weir calibration curves. Condensate flows less than 1 gpm may be determined by periodic observation of the water accumulation in the standpipes of the condensate collection system.
- d. Leakage detection via the containment sump level and discharge flow monitoring systems will determine leakage losses from all fluid systems to the containment free volume. Water collecting on the containment floor will normally be delivered to the containment sump via the containment floor trench system. Level monitoring of the containment sump is in part 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 containment sump. In addition, another level monitoring device provides a continuous level readout in the control room. When the water level in the containment sump reaches predetermined levels, one or both containment sump pumps will automatically start and pump the fluid out of containment to the liquid waste disposal system. Flow in the containment sump pump discharge line from containment to the Waste Holdup Tank is monitored on a continuous basis. Thus, monitoring of both flow indication systems will provide a positive means for determining leakage into the containment free volume.

- 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 leak rate of 150 gpd 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. 2' 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

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 and 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.

- c. Unscheduled steam generator examination 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. At least 20% of a random sample of tubes containing sleeves shall be subjected to an examination throughout the sleeved portion of the tube and the portion of the associated tube in the tube to sleeve joint region.
- 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. 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, 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.
- 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% or 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. Tube sleeve inspection program expansion will be in accordance with Section 4.13 A.4.1.d (below).
- 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 or associated sleeve to tube joint region exhibits degradation of greater than 23% of wall thickness or is otherwise defective, an additional 20% (minimum) of the unsampled sleeves shall be examined. If a sleeve or associated sleeve to tube joint region exhibits degradation of

greater than 23% of wall thickness otherwise defective in the second sample, all remaining sleeves and sleeve to tube joint regions 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 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 the F* region.

2. Tubes or sleeves 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 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.

Sleeving is a tube repair method which inserts a sleeve within the parent steam generator tube to allow the tube to remain in service for heat transfer purposes. Sleeve inservice inspection inspects not only the sleeve but additionally the sleeve to tube joint area. This ensures the integrity of the sleeve as well as the parent tube in the region of the tube potentially affected by the sleeve or the sleeve installation process.

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