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.

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 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. <u>Basic Sample Selection and Examination</u>

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

f. 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 size probe 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.

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

   b. 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 occurring. This criterion has been used successfully for several years at Indian Point Unit 2 and at other plants, and is 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.