UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

METROPOLITAN EDISON COMPANY

JERSEY CENTRAL POWER AND LIGHT COMPANY

PENNSYLVANIA ELECTRIC COMPANY

DOCKET NO. 50-289

THREE MILE ISLAND NUCLEAR STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 47
License No. DPR-50

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Metropolitan Edison Company, Jersey Central Power and Light Company and Pennsylvania Electric Company (the licensees), dated November 12, 1976, as revised January 7, 1977, and supplemented by letter dated July 14, 1978, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.c.(2) of Facility Operating License No. DPR-50 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 47 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Morton B. Fairtile for
Robert W. Reid, Chief
Operating Reactors Branch #4
Division of Operating Reactors

Attachment:
Changes to the Technical Specifications

Date of Issuance: December 22, 1978

ATTACHMENT TO LICENSE AMENDMENT NO. 47

FACILITY OPERATING LICENSE NO. DPR-50

DOCKET NO. 50-289

Revise the Appendix A Technical Specifications as follows:

Remove Pages	Insert Pages
iii - v	iii - v
3-1 & 3-2	3-1 & 3-2
3-11 - 3-14	3-11 - 3-14
--	4-77 - 4-86
6-18	6-18

The changed areas on the revised pages are shown by marginal lines. Pages 3-11 and 3-14 are unchanged and are included for convenience only.

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3. LIMITING CONDITIONS FOR OPERATION

3.1 REACTOR COOLANT SYSTEM

3.1.1 OPERATIONAL COMPONENTS

Applicability

Applies to the operating status of reactor coolant system components.

Objective

To specify those limiting conditions for operation of reactor coolant system components which must be met to ensure safe reactor operations.

Specification

3.1.1.1 Reactor Coolant Pumps

a. Pump combinations permissible for given power levels shall be as shown in Specification Table 2.3.1.

b. Power operation with one idle reactor coolant pump in each loop shall be restricted to 24 hours. If the reactor is not returned to an acceptable RC pump operating combination at the end of the 24-hour period, the reactor shall be in a hot shutdown condition within the next 12 hours.

c. The boron concentration in the reactor coolant system shall not be reduced unless at least one reactor coolant pump or one decay heat removal pump is circulating reactor coolant.

3.1.1.2 Steam Generator

a. Both steam generators shall be operable whenever the reactor coolant average temperature is above 250°F.

3.1.1.3 Pressurizer Safety Valves

a. The reactor shall not remain critical unless both pressurizer code safety valves are operable with a lift setting of 2500 psig ±1%.

b. When the reactor is subcritical, at least one pressurizer code safety valve shall be operable if all reactor coolant system openings are closed, except for hydrostatic tests in accordance with ASME Boiler and Pressure Vessel Code, Section III.

Bases

The limitation on power operation with one idle RC pump in each loop has been imposed since the ECCS cooling performance has not been calculated in accordance with the Final Acceptance Criteria requirements specifically for this mode of reactor operation. A time period of 24 hours is allowed for operation with one idle RC pump in each loop to effect repairs of the idle pump(s) and to return the reactor to an acceptable combination of operating RC pumps. The 24 hours for this mode of operation is acceptable since this mode is expected to have considerable margin for the peak cladding temperature limit and since the likelihood of a LOCA within the 24-hour period is considered very remote.

A reactor coolant pump or decay heat removal pump is required to be in operation before the boron concentration is reduced by dilution with makeup water. Either pump will provide mixing which will prevent sudden positive reactivity changes caused by dilute coolant reaching the reactor. One decay heat removal pump will circulate the equivalent of the reactor coolant system volume in one-half hour or less.

The decay heat removal system suction piping is designed for 300°F and 370 psig; thus, the system can remove decay heat when the reactor coolant system is below this temperature.(2, 3)

Both steam generators must be operable before heatup of the Reactor Coolant System to insure system integrity against leakage under normal and transient conditions. Only one steam generator is required for decay heat removal purposes.

One pressurizer code safety valve is capable of preventing overpressurization when the reactor is not critical since its relieving capacity is greater than that required by the sum of the available heat sources which are pump energy, pressurizer heaters, and reactor decay heat. (4) Both pressurizer code safety valves are required to be in service prior to criticality to conform to the system design relief capabilities. The code safety valves prevent overpressure for a rod withdrawal or feedwater line break accidents. (5) The pressurizer code safety valve lift set point shall be set at 2500 psig ±1% allowance for error and each valve shall be capable of relieving 280,800 lb/h of saturated steam at a pressure not greater than three percent above the set pressure.

References

(1) FSAR, Tables 9-10 and 4-3 through 4-7

(2) FSAR, Sections 4.2.5.1 and 9.5.2.3

(3) FSAR, Section 4.2.5.4

(4) FSAR, Sections 4.3.10.4 and 4.2.4

(5) FSAR, Section 4.3.7

Because of the time dependent nature of any adverse effects arising from chlorides, fluorides, or oxygen concentrations in excess of the limits, and because the condition can be corrected, it is unnecessary to shutdown immediately.

The oxygen, chloride, or fluoride limits specified are at least an order of magnitude below concentrations which could result in damage to materials found in the reactor coolant system even if maintained for an extended period of time.[4] Thus, the period of eight hours to initiate corrective action and the period of 24 hours to perform corrective action to restore the concentration within the limits have been established. The eight hour period to initiate corrective action allows time to ascertain that the chemical analysis are correct and to locate the source of contamination. If corrective action has not been effective at the end of 24 hours, then the reactor coolant system will be brought to the hot shutdown condition within 12 hours thereafter and corrective action will continue. If the normal operational limits are not restored within an additional 24 hour period the reactor shall be placed in cold shutdown condition within 24 hours thereafter.

The maximum limit of 1 ppm for the oxygen, chloride, or fluoride concentration that will not be exceeded was selected because these values have been shown to be safe at 550 F.[3] It is prudent to restrict operation to hot shutdown conditions, if these limits are reached.

REFERENCES

(1) FSAR, Section 4.1.2.7

(2) FSAR, Section 9.2.2

(3) Stress Corrosion of Metals, Logan

(4) Corrosion and Wear Handbook, D. J. DePaul, Editor

3.1.6 LEAKAGE

Applicability

Applies to reactor coolant leakage from the reactor coolant system and the makeup and purification system.

Objective

To assure that any reactor coolant leakage does not compromise the safe operation of the facility.

Specification

3.1.6.1 If the total reactor coolant leakage rate exceeds 10 gpm, the reactor shall be placed in hot shutdown within 24 hours of detection.

3.1.6.2 If unidentified reactor coolant leakage (excluding normal evaporative losses) exceeds one gpm or if any reactor coolant leakage is evaluated as unsafe, the reactor shall be placed in hot shutdown within 24 hours of detection.

3.1.6.3 If primary-to-secondary leakage through the steam generator tubes exceeds 1 gpm total for both steam generators, the reactor shall be placed in cold shutdown within 36 hours of detection.

3.1.6.4 If any reactor coolant leakage exists through a nonisolable fault in an RCS strength boundary (such as the reactor vessel, piping, valve body, etc., except the steam generator tubes), the reactor shall be shutdown, and cooldown to the cold shutdown condition shall be initiated within 24 hours of detection.

3.1.6.5 If reactor shutdown is required by Specification 3.1.6.1, 3.1.6.2, 3.1.6.3, or 3.1.6.4, the rate of shutdown and the conditions of shutdown shall be determined by the safety evaluation for each case and reported as required by Specification 6.9.2.

3.1.6.6 Action to evaluate the safety implication of reactor coolant leakage shall be initiated within four hours of detection. The nature, as well as the magnitude, of the leak shall be considered in this evaluation. The safety evaluation shall assure that the exposure of offsite personnel to radiation is within the guidelines of 10 CFR 20.

3.1.6.7 If reactor shutdown is required per Specification 3.1.6.1, 3.1.6.2, 3.1.6.3, or 3.1.6.4, the reactor shall not be restarted until the leak is repaired or until the problem is otherwise corrected.

3.1.6.8 When the reactor is critical and above 2 percent power, two reactor coolant leak detection systems of different operating principles shall be in operation for the Reactor Building with one of the two systems sensitive to radioactivity. The systems sensitive to radioactivity may be out-of-service for no more than 72 hours provided a sample is taken of the Reactor Building atmosphere every eight hours and analyzed for radioactivity and two other means are available to detect leakage.

3.1.6.9 Loss of reactor coolant through reactor coolant pump seals and system valves to connecting systems which vent to the gas vent header and from which coolant can be returned to the reactor coolant system shall not be considered as reactor coolant leakage and shall not be subject to the consideration of Specifications 3.1.6.1, 3.1.6.2, 3.1.6.3, 3.1.6.4, 3.1.6.5, 3.1.6.6 or 3.1.6.7, except that such losses when added to leakage shall not exceed 30 gpm. If leakage plus losses exceeds 30 gpm, the reactor shall be placed in hot shutdown within 24 hours of detection.

Bases

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 radioactive contamination and required cleanup or, in the case of reactor coolant, it could develop into a still more serious problem and, therefore, the first indications of such leakage will be followed up as soon as practical. The unit's makeup system has the capability to makeup considerably more than 30 gpm of reactor coolant leakage.

Water inventory balances, monitoring equipment, radioactive tracing, boric acid crystalline deposits, and physical inspections can disclose reactor coolant leaks.

Although some leak rates on the order of gallons per minute may be tolerable from a dose point of view, it is recognized that leaks in the order of drops per minute through any of the barriers of the primary system could be indicative of materials 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. Therefore, the nature and location of the leak, as well as the magnitude of the leakage, must be considered in the safety evaluation.

When reactor coolant leakage occurs to the Reactor Building, it is ultimately conducted to the Reactor Building sump. Although the reactor coolant is safely contained, the gaseous components in it escape to the Reactor Building atmosphere. There, the gaseous components become a potential hazard to plant personnel, during inspection tours within the Reactor Building, and to the general public whenever the Reactor Building atmosphere is periodically purged to the environment.

When reactor coolant leakage occurs to the Auxiliary Building, it is collected in the Auxiliary Building sump. The gases escaping from reactor coolant leakage within the Auxiliary Building will be collected in the Auxiliary and Fuel Handling Building exhaust ventilation system and discharged to the environment via the unit's Auxiliary and Fuel Handling Building vent. Since the majority of this leakage occurs within confined, separately ventilated cubicles within the Auxiliary Building, it incurs very little hazard to plant personnel.

When reactor coolant leakage occurs to the nuclear services closed cooling water system, the leakage, both gaseous and liquid, is contained because the nuclear services closed cooling water system surge tank is a closed tank that is maintained above atmospheric pressure. The leakage would be detected by the nuclear services closed cooling water system monitor and by purge tank liquid level, both of which alarm in the control room. Since the nuclear services closed cooling water system's only potential contact with reactor coolant is in the sample coolers, it is considered not to be a hazard. However, if reactor coolant leakage to this receptor occurred and the surge tank's relief valve discharged, radioactive gases could be discharged to the environment via the unit's auxiliary and fuel handling building vent.

When reactor coolant leakage occurs to the intermediate cooling closed cooling water system, the leakage is indicated by both the intermediate cooling water monitor (RM-L9) and the intermediate cooling closed cooling water surge tank liquid level indicator, both of which alarm in the control room. Reactor coolant leakage to this receptor ultimately could result in radioactive gas leaking to the environment via the unit's auxiliary and fuel handling building vent by way of the atmospheric vent on the surge tank.

When reactor coolant leakage occurs to either of the decay heat closed cooling water systems, the leakage is indicated by the affected system's radiation monitor (RM-L2 or RM-L3 for system A and B, respectively) and surge tank liquid level indicator, all four of which alarm in the control room. Reactor coolant leakage to this receptor ultimately could result in radioactive gas leaking to the environment via the unit's auxiliary and fuel handling building vent by way of the atmospheric vent on the surge tank of the affected system.

Assuming the existence of the maximum allowable activity in the reactor coolant, a reactor coolant leakage rate of less than one gpm unidentified leakage within the reactor or auxiliary building or any of the closed cooling water systems indicated above, is a conservative limit on what is allowable before the guide lines of 10 CFR 20 would be exceeded. This is shown as follows: if the specific activity of the reactor coolant is $130/\overline{E}$ μCi/ml and the gaseous portion of it (as identified by Table 11-2) is discharged to the environment via the unit's auxiliary and fuel handling building vent, the yearly whole body dose resulting from this activity at the site boundary, using an annual average $\chi/Q = 4.5 \times 10^{-6}$ sec/m^3, is 0.34 rem. This may be compared with the 10 CFR 20 guideline of 0.5 rem/year whole body dose.

When the reactor coolant leaks to the secondary sides of either steam generator, all the gaseous components and a very small fraction of the ionic components are carried by the steam to the main condenser. The gaseous components exit the main condenser via the unit's vacuum pump which discharges to the condenser vent past the condenser off-gas monitor. The condenser off-gas monitor will detect any radiation, above background, within the condenser vent.

However, buildup of radioactive solids in the secondary side of a steam generator and the presence of radioactive ions in the condensate can be tolerated to only a small degree. Therefore, the appearance of activity in the condenser off-gas, or any other possible indications of primary to secondary leakage such as water inventories, condensate demineralizer activity, etc., shall be considered positive indication of primary to secondary leakage and steps shall be taken to determine the source and quantity of the leakage.

4.19 OTSG TUBE INSERVICE INSPECTION

Applicability

This Technical Specification applies to the inservice inspection of the OTSG tube portion of the reactor coolant pressure boundary.

Objective

The objective of this inservice inspection program is to provide assurance of continued integrity of the tube portion of the Once-Through Steam Generators, while at the same time minimizing radiation exposure to personnel in the performance of the inspection.

Specification

Each steam generator shall be demonstrated OPERABLE by performance of the following augmented inservice inspection program and the requirements of Specification 3.1.6.3.

4.19.1 Steam Generator Sample Selection and Inspection Methods

a. Each steam generator shall be determined OPERABLE during shutdown by selecting and inspecting at least the minimum number of steam generators specified in Table 4.19.1 at the frequency specified in 4.19.3.

b. Inservice inspection of steam generator tubing shall include nondestructive examination by eddy-current testing or other equivalent techniques. The inspection equipment shall be calibrated to provide a sensitivity that will detect defects with a penetration of 20 percent or more of the minimum allowable as-manufactured tube wall thickness.

4.19.2 Steam Generator Tube Sample Selection and Inspection

The steam generator tube minimum sample size, inspection result classification, and the corresponding action required shall be as specified in Table 4.19.2. The inservice inspection of steam generator tubes shall be performed at the frequencies specified in Specification 4.19.3 and the inspected tubes shall be verified acceptable per the acceptance criteria of Specification 4.19.4. The tubes selected for each inservice inspection shall include at least 3% of the total number of tubes in all steam generators; the tubes selected for these inspections shall be selected on a random basis except:

a. The first sample of tubes selected for each inservice inspection (subsequent to the preservice inspection) of each steam generator shall include:
   1. All nonplugged tubes that previously had detectable wall penetrations (>20%).

   2. At least 50% of the tubes inspected shall be in those areas where experience has indicated potential problems.

   3. A tube inspection (pursuant to Specification 4.19.4.a.8) shall be performed on each selected tube. If any selected tube does not permit the passage of the eddy current probe for a tube inspection, this shall be recorded and an adjacent tube shall be selected and subjected to a tube inspection.

   4. Tubes in the following groups may be excluded from the first random sample if all tubes in a group in both steam generators are inspected. No credit will be taken for these tubes in meeting minimum sample size requirements.

      (1) Group A-1: Tubes within one, two or three rows of the open inspection lane.

      (2) Group A-2: Tubes having a drilled opening in the 15th support plate.

b. The tubes selected as the second and third samples (if required by Table 4.19.2) during each inservice inspection may be subjected to a partial tube inspection provided:

   1. The tubes selected for these second and third samples include the tubes from those areas of the tube sheet array where tubes with imperfections were previously found.

   2. The inspection includes those portions of the tubes where imperfections were previously found.

The results of each sample inspection shall be classified into one of the following three categories:

Category	Inspection Results
C-1	Less than 5% of the total tubes inspected in a steam generator are degraded tubes and none of the inspected tubes are defective.
C-2	One or more tubes, but not more than 1% of the total tubes inspected in a steam generator are defective, or between 5% and 10% of the total tubes inspected are degraded tubes.

C-3 More than 10% of the total tubes inspected in a steam generator are degraded tubes or more than 1% of the inspected tubes are defective.

NOTES: (1) In all inspections, previously degraded tubes must exhibit significant (>10%) further wall penetrations to be included in the above percentage calculations.

(2) Where special inspections are performed pursuant to 4.19.2.a.4, defective or degraded tubes found as a result of the inspection shall be included in determining the Inspection Results Category for that special inspection but need not be included in determining the Inspection Results Category for the general steam generator inspection.

4.19.3 Inspection Frequencies

The required inservice inspections of steam generator tubes shall be performed at the following frequencies:

a. The first (baseline) inspection was performed after 6 effective full power months but within 24 calendar months of initial criticality. The subsequent inservice inspections shall be performed not more than 24 calendar months after the previous inspection. If the results of two consecutive inspections for a given group of tubes* encompassing not less than 18 calendar months all fall into the C-1 category or demonstrate that previously observed degradation has not continued and no additional degradation has occurred, the inspection interval for that group may be extended to a maximum of once per 40 months.

b. If the results of the inservice inspection of a steam generator conducted in accordance with Table 4.19-2 at 40 month intervals for a given group of tubes* fall into Category C-3, the inspection frequency for that group shall be increased to at least once per 20 months. The increase in inspection frequency shall apply until the subsequent inspections satisfy the criteria of Specification 4.19.3.a; the interval may then be extended to a maximum of once per 40 months.

c. Additional, unscheduled inservice inspections shall be performed on each steam generator in accordance with the first sample inspection specified in Table 4.19-2 during the shutdown subsequent to any of the following conditions:

   1. Primary-to-secondary tubes leaks (not including leaks originating from tube-to-tube sheet welds) in excess of the limits of Specification 3.1.6.3.

*A group of tubes means: (a) All tubes inspected pursuant to 4.19.2.a.4, or
(b) All tubes in a steam generator less those inspected pursuant to 4.19.2.a.4.

2. A seismic occurrence greater than the Operating Basis Earthquake.

3. A loss of coolant accident requiring actuation of the engineering safeguards, or

4. A major main steam line or feedwater line break.

4.19.4 Acceptance Criteria

a. As used in this Specification:

1. Imperfection means an exception to the dimensions, finish or contour of a tube from that required by fabrication drawing or specifications. Eddy current testing indications below 20% of the nominal tube wall thickness, if detectable, may be considered as imperfections.

2. Degradation means a service-induced cracking, wastage, wear or general corrosion occurring on either inside or outside of a tube.

3. Degraded Tube means a tube containing imperfections ≥20% of the nominal wall thickness caused by degradation.

4. % Degradation means the percentage of the tube wall thickness affected or removed by degradation.

5. Defect means an imperfection of such severity that it exceeds the plugging limit. A tube containing a defect is defective.

6. Plugging Limit means the imperfection depth at or beyond which the tube shall be removed from service because it may become unserviceable prior to the next inspection and is equal to 40% of the nominal tube wall thickness, unless higher limits are shown to be acceptable by analysis and approved by the NRC.

7. Unserviceable describes the condition of a tube if it leaks or contains a defect large enough to affect its structural integrity in the event of an Operating Basis Earthquake, a loss of coolant accident, or a steam line or feedwater line break as specified in 4.19.3.c, above.

8. Tube Inspection means an inspection of the steam generator tube from the bottom of the upper tubesheet completely to the top of the lower tubesheet, except as permitted by 4.19.2.b.2, above.

b. The steam generator shall be determined OPERABLE after completing the corresponding actions (plugging including all tubes exceeding the plugging limit and all tubes containing throughwall cracks) required by Table 4.19.2.

4.19.5 Reports

a. Following the completion of each inservice inspection of steam generator tubes, the number of tubes plugged in each steam generator shall be reported to the NRC within 15 days.

b. The complete results of the steam generator tube inservice inspection shall be reported to the NRC within 3 months following completion of the inspection. This report shall include:

1. Number and extent of tubes inspected.

2. Location and percent of wall-thickness penetration for each indication of an imperfection.

3. Identification of tubes plugged.

c. Results of steam generator tube inspections which fall into Category C-3 and require prompt notification of the NRC shall be reported pursuant to Specification 6.9.2 prior to resumption of plant operation. The written followup of this report shall provide a description of investigations conducted to determine cause of the tube degradation and corrective measures taken to prevent recurrence.

Bases

The Surveillance Requirements for inspection of the steam generator tubes ensure that the structural integrity of this portion of the RCS will be maintained.

The program for inservice inspection of steam generator tubes is based on a modification of Regulatory Guide 1.83, Revision 1. Inservice inspection of steam generator tubing is essential in order to maintain surveillance of the conditions of the tubes in the event that there is evidence of mechanical damage or progressive degradation due to design, manufacturing errors, or inservice conditions. Inservice inspection 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.

The Unit is expected to be operated in a manner such that the secondary coolant will be maintained within those chemistry limits found to result in negligible corrosion of the steam generator tubes. If the secondary coolant chemistry is not maintained within these chemistry limits, localized corrosion may likely result in stress corrosion cracking.

The extent of steam generator tube leakage due to cracking would be limited by the secondary coolant actitivy, Specification 3.1.6.3.

The extent of cracking during plant operation would be limited by the limitation of total steam generator tube leakage between the primary coolant system and the secondary coolant system (primary-to-secondary leakage = 1 gpm). Leakage in excess of this limit will require plant shutdown and an unscheduled inspection, during which the leaking tubes will be located and plugged.

Wastage-type defects are unlikely with proper chemistry treatment of the secondary coolant. However, even if a defect would develop in service, it will be found during scheduled inservice steam generator tube examinations. Plugging will be required for degradation equal to or in excess of 40% of the tube nominal wall thickness, unless higher limits are shown to be acceptable by analysis and are reviewed and approved by the NRC. Steam generator tube inspections of operating plants have demonstrated the capability to reliably detect degradation that has penetrated 20% of the original tube wall thickness.

Where experience in similar plants with similar water chemistry, as documented by USNRC Bulletins/Circulars, indicate critical areas to be inspected, at least 50% of the tubes inspected should be from these critical areas. First sample inspections sample size may be modified subject to NRC review and approval.

Whenever the results of any steam generator tubing inservice inspection fall into Category C-3 on the 1st sample inspection, (See Table 4.19.2), these results will be promptly reported to the NRC pursuant to Specification 6.9.2 prior to resumption of plant operation. Such cases will be considered by the NRC on a case-by-case basis and may result in a requirement for analysis, laboratory examinations, tests, additional eddy current inspection, and revision of the Technical Specifications, if necessary.

TABLE 4.19-1

MINIMUM NUMBER OF STEAM GENERATORS TO BE

INSPECTED DURING INSERVICE INSPECTION

Preservice Inspection	None
No. of Steam Generators per Unit	Two
First Inservice Inspection	Two
Second & Subsequent Inservice Inspections	One[1]

TABLE NOTATION:

1. The Inservice Inspection may be limited to one steam generator on a rotating schedule encompassing 6% of the tubes in that steam generator if the results of the first and subsequent inspections indicate that both steam generators are performing in a like manner. Note that under some circumstances, the operating conditions in one steam generator may be found to be more severe than those in the other steam generator. Under such circumstances the sample sequence shall be modified to inspect the most severe conditions.

TABLE 4.19-2

STEAM GENERATOR TUBE INSPECTION(2)

1ST SAMPLE INSPECTION			2ND SAMPLE INSPECTION		3RD SAMPLE INSPECTION	
Sample Size	Result	Action Required	Result	Action Required	Result	Action Required
A minimum of S Tubes per S. G.(1)	C–1	None	N/A	N/A	N/A	N/A
	C–2	Plug defective tubes and inspect additional 2S tubes in this S. G.	C–1	None	N/A	N/A
			C–2	Plug defective tubes and inspect additional 4S tubes in this S. G.	C–1	None
					C–2	Plug defective tubes
					C–3	Perform action for C–3 result of first sample
			C–3	Perform action for C–3 result of first sample	N/A	N/A
	C–3	Inspect all tubes in this S. G., plug defective tubes and inspect 2S tubes in other S.G. Prompt notification to NRC pursuant to specification 6.9.2.	Other S.G. is C-1	None	N/A	N/A
			Other S.G. is C-2.	Perform action for C–2 result of second sample	N/A	N/A
			Other S.G. is C-3	Inspect all tubes in each S. G. and plug defective tubes. Prompt notification to NRC pursuant to specification 6.9.2.	N/A	N/A

Notes: (1) $S = 3 \frac{N}{n} \%$ Where N is the number of steam generators in the unit, and n is the number of steam generators inspected during an inspection

(2) For tubes inspected pursuant to 4.19.2.a.4: No action is required for C-1 results. For C-2 results in one or both steam generators plug defective tubes. For C-3 results in one or both steam generators, plug defective tubes and provide prompt notification of NRC pursuant to specification 6.9.2.

4.20 REACTOR BUILDING AIR TEMPERATURE

Applicability

This specification applies to the average air temperature of the primary containment during power operations.

Objective

To assure that the temperatures used in the safety analysis of the reactor building are not exceeded.

Specification

4.20.1 When the reactor is critical, the reactor building temperature will be checked once each twenty-four (24) hours. If any detector exceeds 130°F (120°F below elevation 320) the arithmetic average will be computed to assure compliance with Specification 3.17.1.

6.9 REPORTING REQUIREMENTS (cont'd)

6.9.3 Unique Reporting Requirements

A. Special reports shall be submitted to the Director of the Office of Inspection and Enforcement Regional Office within the time period specified for each report. These reports shall be submitted covering the activities identified below:

Tests	Submittal Dates
(1) Containment Structural Integrity Test	
(a) Tendon Surveillance Program	Within 3 months after performance of surveillance program.
(b) Ring Girder Inspection Program	Within 3 months after performance of each inspection.
(2) Steam Generator Tube Inspection Program (See Section 4.19.5)	Within 3 months after completion of inspection.
(3) Containment Integrated Leak Rate Test	Within 6 months after completion of test.
(4) Inservice Inspection Program	Within 6 months after five years of operation.

FOOTNOTES

1. These reporting requirements apply only to Appendix A technical specifications.

2. A single submittal may be made for a multiple unit station. The submittal should combine those sections that are common to all units at the station.

3. This tabulation supplements the requirements of §20.407 of 10 CFR Part 20.