

ATTACHMENT 2 TO AEP:NRC:1166R

EXISTING TECHNICAL SPECIFICATION
PAGES MARKED TO REFLECT PROPOSED CHANGES

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REACTOR COOLANT SYSTEM

STEAM GENERATORS

TECHNICAL
NO CHANGE

FORMAT CHANGE ONLY

LIMITING CONDITION FOR OPERATION

3.4.5 Each steam generator shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.*

ACTION:

With one or more steam generators inoperable, restore the inoperable generator(s) to OPERABLE status prior to increasing T_{avg} above 200°F.

SURVEILLANCE REQUIREMENTS

4.4.5.0 Each steam generator shall be demonstrated OPERABLE by performance of the following augmented inservice inspection program and the requirement of Specification 4.0.5.

4.4.5.1 Steam Generator Sample Selection and Inspection - Each steam generator shall be determined OPERABLE during shutdown by selecting and inspecting at least the minimum number of steam generators specified in Table 4.4-1.

4.4.5.2 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 4.4-2. The inservice inspection of steam generator tubes shall be performed at the frequencies specified in Specification 4.4.5.3 and the inspected tubes shall be verified acceptable per the acceptance criteria of Specification 4.4.5.4. 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. Where experience in similar plants with similar water chemistry indicates critical areas to be inspected, then at least 50% of the tubes inspected shall be from these critical areas.
- b. The first sample of tubes selected for each inservice inspection (subsequent to the preservice inspection) of each steam generator shall include:
 1. All tubes that previously had detectable wall penetrations (greater than or equal to 20%) that have not been plugged or repaired by sleeving in the affected area.

*This Specification does not apply in Mode 4 while performing crevice flushing as long as Limiting Conditions for Operation for Specification 3.4.1.3 are maintained.

REACTOR COOLANT SYSTEM

SURVEILLANCE REQUIREMENTS

2. Tubes in those areas where experience has indicated potential problems.

3. A tube inspection (pursuant to Specification 4.4.5.4.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.

Insert A

4.

c. In addition to the sample required in 4.4.5.2.b.1 through 3, all tubes which have had the F* criteria applied will be inspected in the roll expanded region. The roll expanded region of these tubes may be excluded from the requirements of 4.4.5.2.b.1.

d. The tubes selected as the second and third samples (if required by Table 4.4-2) during each inservice inspection may be subjected to a partial tube inspection provided:

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

2. The inspections include those portions of the tubes where imperfections were previously found.

Insert B

e. ~~Implementation of the steam generator tube/tube support plate interim plugging criteria for one fuel cycle (Cycle 14) requires a 100% bobbin coil inspection for hot leg tube support plate intersections and cold leg intersections down to the lowest cold leg tube support plate with known outer diameter stress corrosion cracking (ODSCC) indications.~~

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

<u>Category</u>	<u>Inspection Results</u>
C-1	Less than 5% of the total tubes inspected 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 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 are degraded tubes or more than 1% of the inspected tubes are defective.

INSERT "A"

- b. 4. Tubes left in service as a result of application of the tube support plate plugging criteria shall be inspected by bobbin coil probe during all future refueling outages.

INSERT "B"

- e. Implementation of the steam generator tube/tube support plate plugging criteria for one fuel cycle (cycle 15) requires a 100 percent bobbin coil inspection for hot-leg tube support plate intersections and cold-leg intersections down to the lowest cold-leg tube support plate with known outside diameter stress corrosion cracking (ODSCC) indications. The determination of tube support plate intersections having ODSCC indications shall be based on the performance of at least a 20 percent random sampling of tubes inspected over their full length.

REACTOR COOLANT SYSTEM

SURVEILLANCE REQUIREMENTS (Continued)

Note: In all inspections, previously degraded tubes must exhibit significant (greater than or equal to 10%) further wall penetrations to be included in the above percentage calculations.

4.4.5.3 Inspection Frequencies - The above required inservice inspections of steam generator tubes shall be performed at the following frequencies:

- a. The first inservice inspection shall be performed after 6 Effective Full Power Months but within 24 calendar months of initial criticality. Subsequent inservice inspections shall be performed at intervals of not less than 12 nor more than 24 calendar months after the previous inspection. If two consecutive inspections following service under AVT conditions, not including the preservice inspection, result in all inspection results falling into the C-1 category or if two consecutive inspections demonstrate that previously observed degradation has not continued and no additional degradation has occurred, the inspection interval may be extended to a maximum of once per 40 months.
- b. If the results of inservice inspection of a steam generator conducted in accordance with Table 4.4-2 at 40 month intervals fall in Category C-3, the inspection frequency 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 4.4.5.3.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 4.4-2 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 3.4.6.2.
 2. A seismic occurrence greater than the Operating Basis Earthquake.
 3. A loss-of-coolant accident requiring actuation of the engineered safeguards.
 4. A main steam line or feedwater line break.

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- ~~d. Tubes left in service as a result of application of the tube support plate interim plugging criteria shall be inspected by bobbin coil probe during all future refueling outages.~~

REACTOR COOLANT SYSTEM

SURVEILLANCE REQUIREMENTS (Continued)

4.4.5.4 Acceptance Criteria

a. As used in this Specification:

1. Imperfection means an exception to the dimensions, finish or contour of a tube or sleeve from that required by fabrication drawings 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 or sleeve.
3. Degraded Tube or Sleeve means an imperfection greater than or equal to 20% of the nominal wall thickness caused by degradation.
4. Percent Degradation means the amount of the tube wall thickness affected or removed by degradation.
5. Defect means an imperfection of such severity that it exceeds the repair limit.
6. Repair/Plugging Limit means the imperfection depth at or beyond which the tube or sleeved tube shall be repaired or removed from service. Any tube which, upon inspection, exhibits tube wall degradation of 40 percent or more of the nominal tube wall thickness shall be plugged or repaired prior to returning the steam generator to service. This definition does not apply to the portion of the tube in the tubesheet below the F* distance for F* tubes. Any sleeve which, upon inspection, exhibits wall degradation of 29 percent or more of the nominal wall thickness shall be plugged prior to returning the steam generator to service. In addition, any sleeve exhibiting any measurable wall loss in sleeve expansion transition or weld zones shall be plugged. ~~This definition does not apply for tubes experiencing outer diameter stress corrosion cracking confirmed by bobbin probe inspection to be within the thickness of the tube support plates. See 4.4.5.4.a.10 for the plugging limit for use within the thickness of the tube support plate.~~
7. Unserviceable describes the condition of a tube or sleeve 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 4.4.5.3.c, above.
8. Inspection determines the condition of the steam generator tube or sleeve from the point of entry (hot leg side) completely

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REACTOR COOLANT SYSTEM

SURVEILLANCE REQUIREMENTS (Continued)

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around the U-bend to the top support of the cold leg. For a tube in which the tube support plate elevation interim plugging limit has been applied, the inspection will include all the hot leg intersections and all cold leg intersections down to, at least, the level of the last crack indication. ..

Insert D

9. Sleeving a tube is permitted only in areas where the sleeve spans the tubesheet area and whose lower joint is at the primary fluid tubesheet face.

10. ~~The Tube Support Plate Interim Plugging Criteria is used for disposition of a steam generator tube for continued service that is experiencing outer diameter initiated stress corrosion cracking confined within the thickness of the tube support plates. For application of the tube support plate interim plugging limit, the tube's disposition for continued service will be based upon standard bobbin probe signal amplitude. The plant-specific guidelines used for all inspections shall be amended as appropriate to accommodate the additional information needed to evaluate tube support plate signals with respect to the above voltage/depth parameters. Pending incorporation of the voltage verification requirements in ASME standard verifications, an ASME standard calibrated against the laboratory standard will be utilized in the Donald C. Cook Nuclear Plant Unit 1 steam generator inspections for consistent voltage normalization.~~

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1. ~~A tube can remain in service if the signal amplitude of a crack indication is less than or equal to 2.0 volt, regardless of the depth of tube wall penetration, if, as a result, the projected end-of-cycle distribution of crack indications is verified to result in primary-to-secondary leakage less than 12.6 gpm in the faulted loop during a postulated steam line break event. The methodology for calculating expected leak rates from the projected crack distribution must be consistent with WCAP-13187, Rev. 0, and as prescribed in draft NUREG-1477.~~
2. ~~A tube should be plugged or repaired if the signal amplitude of the crack indication is greater than 2.0 volt except as noted in 4.4.5.4.a.10.3 below.~~
3. ~~A tube can remain in service with a bobbin coil signal amplitude greater than 2.0 volt but less than or equal to 3.6 volts, if a rotating pancake probe inspection does not detect degradation. Indications of degradation with a bobbin coil signal amplitude greater than 3.6 volts will be plugged or repaired.~~

11. F* Distance is the distance from the bottom of the hardroll transition toward the bottom of the tubesheet that has been conservatively determined to be 1.11 inches (not including eddy current uncertainty).
12. F* Tube is a tube with degradation, below the F* distance, equal to or greater than 40%, and not degraded (i.e., no indications of cracking) within the F* distance.

INSERT "C"

This definition does not apply to tube support plate intersections for which the voltage-based plugging criteria are being applied. Refer to 4.4.5.4.a.10 for the plugging limit applicable to these intersections.

INSERT "D"

10. Tube Support Plate Repair Limit is used for the disposition of a steam generator tube for continued service that is experiencing outside diameter stress corrosion cracking (ODSCC) confined within the thickness of the tube support plates. At tube support plate intersections, the repair limit is based on maintaining steam generator tube serviceability as described below:
 - a. Degradation attributed to ODSCC within the bounds of the tube support plate with bobbin voltage less than or equal to 2.0 volts will be allowed to remain in service.
 - b. Degradation attributed to ODSCC within the bounds of the tube support plate with a bobbin voltage greater than 2.0 volts will be repaired or plugged except as noted in 4.4.5.4.a.10.c below.
 - c. Indications of potential degradation attributed to ODSCC within the bounds of the tube support plate with a bobbin voltage greater than 2.0 volts but less than or equal to 5.6 volts may remain in service if a rotating pancake coil inspection does not detect degradation. Indications of ODSCC degradation with a bobbin voltage greater than 5.6 volts will be plugged or repaired.

REACTOR COOLANT SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

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- b. The steam generator shall be determined OPERABLE after completing the corresponding actions (plugging or sleeving all tubes exceeding the repair limit and all tubes containing through-wall cracks) required by Table 4.4-2.
- c. Steam generator tube repairs may be made in accordance with the methods described in either WCAP-12623 or CEN-313-P.

4.4.5.5 Reports

- a. Following each inservice inspection of steam generator tubes, if there are any tubes requiring plugging or sleeving, the number of tubes plugged or sleeved in each steam generator shall be reported to the Commission within 15 days.
- b. The complete results of the steam generator tube inservice inspection shall be included in the Annual Operating Report for the period in which this inspection was completed. 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 or sleeved.
- c. Results of steam generator tube inspections which fall into Category C-3 and require prompt notification of the Commission shall be reported pursuant to Specification 6.9.1 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.

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- d. ~~The results of inspections performed under 4.4.5.2 for all tubes in which the tube support plate interim plugging criteria has been applied or that have defects below the F* distance and were not plugged shall be reported to the Commission within 15 days following the inspection. The report shall include:~~
 - 1. ~~Listing of applicable tubes.~~
 - 2. ~~Location (applicable intersections per tube) and extent of degradation (voltage).~~

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- e. ~~The results of steam line break leakage analysis performed under T/S 4.4.5.4.4.10 will be reported to the Commission prior to restart for cycle 14.~~

INSERT "E"

4.4.5.5. Reports

- d. For implementation of the voltage-based repair criteria to tube support plate intersections, notify the staff prior to returning the steam generators to service should any of the following conditions arise:
 1. If estimated leakage based on the actual measured end-of-cycle voltage distribution would have exceeded the leak limit (for the postulated main steam line break utilizing "Standard Review Plan - NUREG - 0800" assumptions) during the previous operating cycle.
 2. If circumferential crack-like indications are detected at the tube support plate intersections.
 3. If significant indications are identified that extend beyond the confines of the tube support plate.
 4. If the calculated conditional burst probability, as calculated per WCAP-14277, exceeds 1×10^{-2} , notify the NRC and provide an assessment of the safety significance of the occurrence.

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REACTOR COOLANT SYSTEM

OPERATIONAL LEAKAGE

LIMITING CONDITION FOR OPERATION

3.4.6.2 Reactor Coolant System leakage shall be limited to:

- a. No PRESSURE BOUNDARY LEAKAGE,
- b. 1 GPM UNIDENTIFIED LEAKAGE, ¹⁵
- c. 600 gallons per day total primary-to-secondary leakage through all steam generators and 150 gallons per day through any one steam generator for Fuel Cycle ~~(X)~~
- d. 10 GPM IDENTIFIED LEAKAGE from the Reactor Coolant System,
- e. Seal line resistance greater than or equal to $2.27E-1$ ft/gpm² and,
- f. The leakage from each Reactor Coolant System Pressure Isolation Valves specified in Table 3.4-0 shall be limited to 0.5 gpm per nominal inch of valve size up to a maximum of 5 gpm, at a Reactor Coolant System average pressure within 20 psi of the nominal full pressure value.

APPLICABILITY: MODES 1, 2, 3 and 4.*

ACTION:

- a. With any PRESSURE BOUNDARY LEAKAGE, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.
- b. With any Reactor Coolant System leakage greater than any one of the above limits, excluding PRESSURE BOUNDARY LEAKAGE, reduce the leakage rate to within limits within 4 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.
- c. With any reactor coolant system pressure isolation valve(s) leakage greater than the above limit, declare the leaking valve inoperable and isolate the high pressure portion of the affected system from the low pressure portion by the use of a combination of at least two closed valves, one of which may be the OPERABLE check valve and the other a closed de-energized motor operated valve. Verify the isolated condition of the closed de-energized motor operated valve at least once per 24 hours, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

* Specification 3.4.6.2.e is applicable with average pressure within 20 psi of the nominal full pressure value.

REACTOR COOLANT SYSTEM

BASES

3/4.4.5 STEAM GENERATORS TUBE INTEGRITY

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 that lead to corrosion. 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 plant 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 parameter limits, localized corrosion may likely result in stress corrosion cracking. The extent of cracking during plant operation would be limited by the limitation of steam generator tube leakage between the primary coolant system and the secondary coolant system. The allowable primary-to-secondary leak rate is 150 gallons per day per steam generator for one fuel cycle (Cycle 16). Axial or circumferentially oriented cracks having a primary-to-secondary leakage less than this limit during operation will have an adequate margin of safety to withstand the loads imposed during normal operation and by postulated accidents. Leakage in excess of this limit will require plant shutdown and an inspection, during which the leaking tubes will be located and plugged or repaired. A steam generator while undergoing crevice flushing in Mode 4 is available for decay heat removal and is operable/operating upon reinstatement of auxiliary or main feed flow control and steam control.

Wastage-type defects are unlikely with the all volatile treatment (AVT) of secondary coolant. However, even if a defect of similar type should develop in service, it will be found during scheduled inservice steam generator tube examinations. Plugging or sleeving will be required for all tubes with imperfections exceeding the repair limit which is defined in Specification 4.4.5.4.a. 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.

Tubes experiencing outer diameter stress corrosion cracking within the thickness of the tube support plates are plugged or repaired by the criteria of 4.4.5.4.a.10.

REACTOR COOLANT SYSTEM

BASES

3/4.4.5 STEAM GENERATORS TUBE INTEGRITY (Continued)

Whenever the results of any steam generator tubing inservice inspection fall into Category C-3, these results will be promptly reported to the Commission pursuant to Specification 6.9.1 prior to resumption of plant operation. Such cases will be considered by the Commission 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.

TECHNICAL NO ^ CHANGE

REACTOR COOLANT SYSTEM

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BASES

Degraded steam generator tubes may be repaired by the installation of sleeves which span the section of degraded steam generator tubing. A steam generator tube with a sleeve installed meets the structural requirements of tubes which are not degraded.

To determine the basis for the sleeve plugging limit, the minimum sleeve wall thickness was calculated in accordance with Draft Regulatory Guide 1.121 (August 1976). In addition, a combined allowance of 20 percent of wall thickness is assumed for eddy current testing inaccuracies and continued operational degradation per Draft Regulatory Guide 1.121 (August 1976).

The following sleeve designs have been found acceptable by the NRC staff:

1. Westinghouse Mechanical Sleeves (WCAP-12623)
2. Combustion Engineering Leak Tight Sleeves (CEN-313-P)

Descriptions of other future sleeve designs shall be submitted to the NRC for review and approval in accordance with 10CFR50.90 prior to their use in the repair of degraded steam generator tubes. The submittals related to other sleeve design shall be made at least 90 days prior to use.

REACTOR COOLANT SYSTEM

BASES

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3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE

3/4.4.6.1 LEAKAGE DETECTION SYSTEMS

The RCS leakage detection systems required by this specification are provided to monitor and detect leakage from the Reactor Coolant Pressure Boundary. These detection systems are consistent with the recommendations of Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems," May 1973.

3/4.4.6.2 OPERATIONAL LEAKAGE

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 early detection of additional leakage.

The 10 GPM IDENTIFIED LEAKAGE limitations 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.

The limitation on seal line resistance ensures that the seal line resistance is greater than or equal to the resistance assumed in the minimum safeguards LOCA analysis. This analysis assumes that all of the flow that is diverted from the boron injection line to the seal injection line is unavailable for core cooling.



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REACTOR COOLANT SYSTEM

BASES

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Maintaining an operating leakage limit of 150 gpd per steam generator (600 gpd total) for Fuel Cycle (X) will minimize the potential for a large leakage event during steam line break under LOCA conditions. Based on the NDE uncertainties, bobbin coil voltage distribution and crack growth rate from the previous inspection, the expected leak rate following a steam line rupture is limited to below 12.6 gpm which will limit the calculated offsite doses to within 10 percent of 10 CFR 100 guidelines. Leakage in the intact loops is limited to 150 gpd. If the projected end of cycle distribution of crack indications results in primary-to-secondary leakage greater than 12.6 gpm in the faulted loop during a postulated steam line break event, additional tubes must be removed from service in order to reduce the postulated primary-to-secondary steam line break leakage to below 12.6 gpm.

PRESSURE BOUNDARY LEAKAGE of any magnitude is unacceptable since it may be indicative of an impending gross failure of the pressure boundary. Should PRESSURE BOUNDARY LEAKAGE occur through a component which can be isolated from the balance of the Reactor Coolant System, plant operation may continue provided the leaking component is promptly isolated from the Reactor Coolant System since isolation removes the source of potential failure.

3/4.4.7 CHEMISTRY

The limitations on Reactor Coolant System chemistry ensure that corrosion of the Reactor Coolant System is minimized and reduces the potential for Reactor Coolant System leakage or failure due to stress corrosion. Maintaining the chemistry within the Steady State Limits provides adequate corrosion protection to ensure the structural integrity of the Reactor Coolant System over the life of the plant. The associated effects of exceeding the oxygen, chloride, and fluoride limits are time and temperature dependent. Corrosion studies show that operation may be continued with contaminant concentration levels in excess of the Steady State Limits, up to the Transient Limits, for the specified limited time intervals without having a significant effect on the structural integrity of the Reactor Coolant System. The time interval permitting continued operation within the restrictions of the Transient Limits provides time for taking corrective actions to restore the contaminant concentrations to within the Steady State Limits.

REACTOR COOLANT SYSTEM

BASIS

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The surveillance requirements provide adequate assurance that concentrations in excess of the limits will be detected in sufficient time to take corrective action.

3/4.4.8 SPECIFIC ACTIVITY

The limitations on the specific activity of the primary coolant ensure that the resulting 2 hour doses at the site boundary will not exceed an appropriately small fraction of Part 100 limits following a steam generator tube rupture accident in conjunction with an assumed steady state primary-to-secondary steam generator leakage rate of 1.0 GPM. The values for the limits on specific activity represent interim limits based upon a parametric evaluation by the NRC of typical site locations. These values are conservative in that specific site parameters of the Cook Nuclear Plant site, such as site boundary location and meteorological conditions, were not considered in this evaluation. The NRC is finalizing site specific criteria which will be used as the basis for the reevaluation of the specific activity limits of this site. This reevaluation may result in higher limits.

Offsite doses following a main steam line break are limited to 10 percent of the 10 CFR 100 guideline. The restriction is based on a Cook Nuclear Plant site-specific radiological evaluation that assumes a post-accident primary-to-secondary leak rate of 120 gpm in the faulted loop and a primary coolant specific activity concentration corresponding to 13 fuel defects (approximately 4.6 microCuries/gram dose equivalent I-131), rather than a specific activity of 1.0 microCuries dose equivalent I-131.

Reducing T_{avg} to less than 500°F prevents the release of activity should a steam generator tube rupture since the saturation pressure of the primary coolant is below the lift pressure of the atmospheric steam relief valves. The surveillance requirements provide adequate assurance that excessive specific activity levels in the primary coolant will be detected in sufficient time to take corrective action. Information obtained on iodine spiking will be used to assess the parameters associated with spiking phenomena. A reduction in frequency of isotopic analyses following power changes may be permissible if justified by the data obtained.

NO TECHNICAL CHANGE



ATTACHMENT 3 TO AEP:NRG:1166R

PROPOSED REVISED
TECHNICAL SPECIFICATION PAGES



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3/4 LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS
3/4.4 REACTOR COOLANT SYSTEM

STEAM GENERATORS

LIMITING CONDITION FOR OPERATION

3.4.5 Each steam generator shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.*

ACTION:

With one or more steam generators inoperable, restore the inoperable generator(s) to OPERABLE status prior to increasing T_{avg} above 200°F.

SURVEILLANCE REQUIREMENTS

4.4.5.0 Each steam generator shall be demonstrated OPERABLE by performance of the following augmented inservice inspection program and the requirement of Specification 4.0.5.

4.4.5.1 Steam Generator Sample Selection and Inspection - Each steam generator shall be determined OPERABLE during shutdown by selecting and inspecting at least the minimum number of steam generators specified in Table 4.4-1.

4.4.5.2 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 4.4-2. The inservice inspection of steam generator tubes shall be performed at the frequencies specified in Specification 4.4.5.3 and the inspected tubes shall be verified acceptable per the acceptance criteria of Specification 4.4.5.4. 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. Where experience in similar plants with similar water chemistry indicates critical areas to be inspected, then at least 50% of the tubes inspected shall be from these critical areas.
- b. The first sample of tubes selected for each inservice inspection (subsequent to the preservice inspection) of each steam generator shall include:
 1. All tubes that previously had detectable wall penetrations (greater than or equal to 20%) that have not been plugged or repaired by sleeving in the affected area.

* This Specification does not apply in Mode 4 while performing crevice flushing as long as Limiting Conditions for Operation for Specification 3.4.1.3 are maintained.

3/4 LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS
3/4.4 REACTOR COOLANT SYSTEM

SURVEILLANCE REQUIREMENTS (continued)

2. Tubes in those areas where experience has indicated potential problems.
 3. A tube inspection (pursuant to Specification 4.4.5.4.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 left in service as a result of application of the tube support plate plugging criteria shall be inspected by bobbin coil probe during all future refueling outages.
- c. In addition to the sample required in 4.4.5.2.b.1 through 3, all tubes which have had the F* criteria applied will be inspected in the roll expanded region. The roll expanded region of these tubes may be excluded from the requirements of 4.4.5.2.b.1.
- d. The tubes selected as the second and third samples (if required by Table 4.4-2) during each inservice inspection may be subjected to a partial tube inspection provided:
1. The tubes selected for the samples include the tubes from those areas of the tube sheet array where tubes with imperfections were previously found.
 2. The inspections include those portions of the tubes where imperfections were previously found.
- e. Implementation of the steam generator tube/tube support plate plugging criteria for one fuel cycle (Cycle 15) requires a 100 percent bobbin coil inspection for hot-leg tube support plate intersections and cold-leg intersections down to the lowest cold-leg tube support plate with known outside diameter stress corrosion cracking (ODSCC) indications. The determination of tube support plate intersections having ODSCC indications shall be based on the performance of at least a 20 percent random sampling of tubes inspected over their full length.

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

<u>Category</u>	<u>Inspection Results</u>
C-1	Less than 5% of the total tubes inspected 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 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 are degraded tubes or more than 1% of the inspected tubes are defective.

3/4 LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS
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SURVEILLANCE REQUIREMENTS (continued)

Note: In all inspections, previously degraded tubes must exhibit significant (greater than or equal to 10%) further wall penetrations to be included in the above percentage calculations.

4.4.5.3 Inspection Frequencies - The above required inservice inspections of steam generator tubes shall be performed at the following frequencies:

- a. The first inservice inspection shall be performed after 6 Effective Full Power Months but within 24 calendar months of initial criticality. Subsequent inservice inspections shall be performed at intervals of not less than 12 nor more than 24 calendar months after the previous inspection. If two consecutive inspections following service under AVT conditions, not including the preservice inspection, result in all inspection results falling into the C-1 category or if two consecutive inspections demonstrate that previously observed degradation has not continued and no additional degradation has occurred, the inspection interval may be extended to a maximum of once per 40 months.
- b. If the results of inservice inspection of a steam generator conducted in accordance with Table 4.4-2 at 40 month intervals fall in Category C-3, the inspection frequency 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 4.4.5.3.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 4.4-2 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 3.4.6.2.
 2. A seismic occurrence greater than the Operating Basis Earthquake.
 3. A loss-of-coolant accident requiring actuation of the engineered safeguards.
 4. A main steam line or feedwater line break.



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3/4 LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS
3/4.4 REACTOR COOLANT SYSTEM

SURVEILLANCE REQUIREMENTS (continued)

4.4.5.4 Acceptance Criteria

a. As used in this Specification:

1. Imperfection means an exception to the dimensions, finish or contour of a tube or sleeve from that required by fabrication drawings 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 or sleeve.
3. Degraded Tube or Sleeve means an imperfection greater than or equal to 20% of the nominal wall thickness caused by degradation.
4. Percent Degradation means the amount of the tube wall thickness affected or removed by degradation.
5. Defect means an imperfection of such severity that it exceeds the repair limit.
6. Repair/Plugging Limit means the imperfection depth at or beyond which the tube or sleeved tube shall be repaired or removed from service. Any tube which, upon inspection, exhibits tube wall degradation of 40 percent or more of the nominal tube wall thickness shall be plugged or repaired prior to returning the steam generator to service. This definition does not apply to the portion of the tube in the tubesheet below the F* distance for F* tubes. Any sleeve which, upon inspection, exhibits wall degradation of 29 percent or more of the nominal wall thickness shall be plugged prior to returning the steam generator to service. In addition, any sleeve exhibiting any measurable wall loss in sleeve expansion transition or weld zones shall be plugged. This definition does not apply to tube support plate intersections for which the voltage-based plugging criteria are being applied. Refer to 4.4.5.4.a.10 for the plugging limit applicable to these intersections.
7. Unserviceable describes the condition of a tube or sleeve 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 4.4.5.3.c, above.
8. Inspection determines the condition of the steam generator tube or sleeve from the point of entry (hot leg side) completely around the U-bend to the top support of the cold leg. For a tube in which the tube support plate elevation interim plugging limit has been applied, the inspection will include all the hot leg intersections and all cold leg intersections down to, at least, the level of the last crack indication.
9. Sleeving a tube is permitted only in areas where the sleeve spans the tubesheet area and whose lower joint is at the primary fluid tubesheet face.

3/4 LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS
3/4.4 REACTOR COOLANT SYSTEM

SURVEILLANCE REQUIREMENTS (continued)

10. Tube Support Plate Repair Limit is used for the disposition of a steam generator tube for continued service that is experiencing outside diameter stress corrosion cracking (ODSCC) confined within the thickness of the tube support plates. At tube support plate intersections, the repair limit is based on maintaining steam generator tube serviceability as described below:
 - a. Degradation attributed to ODSCC within the bounds of the tube support plate with bobbin voltage less than or equal to 2.0 volts will be allowed to remain in service.
 - b. Degradation attributed to ODSCC within the bounds of the tube support plate with a bobbin voltage greater than 2.0 volts will be repaired or plugged except as noted in 4.4.5.4.a.10.c below.
 - c. Indications of potential degradation attributed to ODSCC within the bounds of the tube support plate with a bobbin voltage greater than 2.0 volts but less than or equal to 5.6 volts may remain in service if a rotating pancake coil inspection does not detect degradation. Indications of ODSCC degradation with a bobbin voltage greater than 5.6 volts will be plugged or repaired.
11. F* Distance is the distance from the bottom of the hardroll transition toward the bottom of the tubesheet that has been conservatively determined to be 1.11 inches (not including eddy current uncertainty).
12. F* Tube is a tube with degradation, below the F* distance, equal to or greater than 40%, and not degraded (i.e., no indications of cracking) within the F* distance.
 - b. The steam generator shall be determined OPERABLE after completing the corresponding actions (plugging or sleeving all tubes exceeding the repair limit and all tubes containing through-wall cracks) required by Table 4.4-2.
 - c. Steam generator tube repairs may be made in accordance with the methods described in either WCAP-12623 or CEN-313-P.

3/4 LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS
3/4.4 REACTOR COOLANT SYSTEM

SURVEILLANCE REQUIREMENTS (continued)

4.4.5.5 Reports

- a. Following each inservice inspection of steam generator tubes, if there are any tubes requiring plugging or sleeving, the number of tubes plugged or sleeved in each steam generator shall be reported to the Commission within 15 days.
- b. The complete results of the steam generator tube inservice inspection shall be included in the Annual Operating Report for the period in which this inspection was completed. 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 or sleeved.
- c. Results of steam generator tube inspections which fall into Category C-3 and require prompt notification of the Commission shall be reported pursuant to Specification 6.9.1 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.
- d. For implementation of the voltage-based repair criteria to tube support plate intersections, notify the staff prior to returning the steam generators to service should any of the following conditions arise:
 1. If estimated leakage based on the actual measured end-of-cycle voltage distribution would have exceeded the leak limit (for the postulated main steam line break utilizing "Standard Review Plan - NUREG-0800" assumptions) during the previous operating cycle.
 2. If circumferential crack-like indications are detected at the tube support plate intersections.
 3. If significant indications are identified that extend beyond the confines of the tube support plate.
 4. If the calculated conditional burst probability, as calculated per WCAP-14277, exceeds 1×10^{-2} , notify the NRC and provide an assessment of the safety significance of the occurrence.

3/4 LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS
3.4.4 REACTOR COOLANT SYSTEM

OPERATIONAL LEAKAGE

LIMITING CONDITION FOR OPERATION

3.4.6.2 Reactor Coolant System leakage shall be limited to:

- a. No PRESSURE BOUNDARY LEAKAGE,
- b. 1 GPM UNIDENTIFIED LEAKAGE,
- c. 600 gallons per day total primary-to-secondary leakage through all steam generators and 150 gallons per day through any one steam generator for Fuel Cycle 15,
- d. 10 GPM IDENTIFIED LEAKAGE from the Reactor Coolant System,
- e. Seal line resistance greater than or equal to $2.27E-1$ ft/gpm² and,
- f. The leakage from each Reactor Coolant System Pressure Isolation Valves specified in Table 3.4-0 shall be limited to 0.5 gpm per nominal inch of valve size up to a maximum of 5 gpm, at a Reactor Coolant System average pressure within 20 psi of the nominal full pressure value.

APPLICABILITY: MODES 1, 2, 3 and 4.*

ACTION:

- a. With any PRESSURE BOUNDARY LEAKAGE, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.
- b. With any Reactor Coolant System leakage greater than any one of the above limits, excluding PRESSURE BOUNDARY LEAKAGE, reduce the leakage rate to within limits within 4 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.
- c. With any reactor coolant system pressure isolation valve(s) leakage greater than the above limit, declare the leaking valve inoperable and isolate the high pressure portion of the affected system from the low pressure portion by the use of a combination of at least two closed valves, one of which may be the OPERABLE check valve and the other a closed de-energized motor operated valve. Verify the isolated condition of the closed de-energized motor operated valve at least once per 24 hours, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

* Specification 3.4.6.2.e is applicable with average pressure within 20 psi of the nominal full pressure value.

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3/4.4.5 STEAM GENERATORS TUBE INTEGRITY

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 that lead to corrosion. 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 plant 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 parameter limits, localized corrosion may likely result in stress corrosion cracking. The extent of cracking during plant operation would be limited by the limitation of steam generator tube leakage between the primary coolant system and the secondary coolant system. The allowable primary-to-secondary leak rate is 150 gallons per day per steam generator for one fuel cycle (Cycle 15). Axial or circumferentially oriented cracks having a primary-to-secondary leakage less than this limit during operation will have an adequate margin of safety to withstand the loads imposed during normal operation and by postulated accidents. Leakage in excess of this limit will require plant shutdown and an inspection, during which the leaking tubes will be located and plugged or repaired. A steam generator while undergoing crevice flushing in Mode 4 is available for decay heat removal and is operable/ operating upon reinstatement of auxiliary or main feed flow control and steam control.

Wastage-type defects are unlikely with the all volatile treatment (AVT) of secondary coolant. However, even if a defect of similar type should develop in service, it will be found during scheduled inservice steam generator tube examinations. Plugging or sleeving will be required for all tubes with imperfections exceeding the repair limit which is defined in Specification 4.4.5.4.a. 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.

Tubes experiencing outer diameter stress corrosion cracking within the thickness of the tube support plates are plugged or repaired by the criteria of 4.4.5.4.a.10.

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3/4.4.5 STEAM GENERATORS TUBE INTEGRITY (Continued)

Whenever the results of any steam generator tubing inservice inspection fall into Category C-3, these results will be promptly reported to the Commission pursuant to Specification 6.9.1 prior to resumption of plant operation. Such cases will be considered by the Commission 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.

Degraded steam generator tubes may be repaired by the installation of sleeves which span the section of degraded steam generator tubing. A steam generator tube with a sleeve installed meets the structural requirements of tubes which are not degraded.

To determine the basis for the sleeve plugging limit, the minimum sleeve wall thickness was calculated in accordance with Draft Regulatory Guide 1.121 (August 1976). In addition, a combined allowance of 20 percent of wall thickness is assumed for eddy current testing inaccuracies and continued operational degradation per Draft Regulatory Guide 1.121 (August 1976).

The following sleeve designs have been found acceptable by the NRC staff:

1. Westinghouse Mechanical Sleeves (WCAP-12623)
2. Combustion Engineering Leak Tight Sleeves (CEN-313-P)

Descriptions of other future sleeve designs shall be submitted to the NRC for review and approval in accordance with 10 CFR 50.90 prior to their use in the repair of degraded steam generator tubes. The submittals related to other sleeve designs shall be made at least 90 days prior to use.

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3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE

3/4.4.6.1 LEAKAGE DETECTION SYSTEMS

The RCS leakage detection systems required by this specification are provided to monitor and detect leakage from the Reactor Coolant Pressure Boundary. These detection systems are consistent with the recommendations of Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems," May 1973.

3/4.4.6.2 OPERATIONAL LEAKAGE

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 early detection of additional leakage.

The 10 GPM IDENTIFIED LEAKAGE limitations 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.

The limitation on seal line resistance ensures that the seal line resistance is greater than or equal to the resistance assumed in the minimum safeguards LOCA analysis. This analysis assumes that all of the flow that is diverted from the boron injection line to the seal injection line is unavailable for core cooling.

Maintaining an operating leakage limit of 150 gpd per steam generator (600 gpd total) for Fuel Cycle 15 will minimize the potential for a large leakage event during steam line break under LOCA conditions. Based on the NDE uncertainties, bobbin coil voltage distribution and crack growth rate from the previous inspection, the expected leak rate following a steam line rupture is limited to below 12.6 gpm which will limit the calculated offsite doses to within 10 percent of the 10 CFR 100 guidelines. Leakage in the intact loops is limited to 150 gpd. If the projected end of cycle distribution of crack indications results in primary-to-secondary leakage greater than 12.6 gpm in the faulted loop during a postulated steam line break event, additional tubes must be removed from service in order to reduce the postulated primary-to-secondary steam line break leakage to below 12.6 gpm.

PRESSURE BOUNDARY LEAKAGE of any magnitude is unacceptable since it may be indicative of an impending gross failure of the pressure boundary. Should PRESSURE BOUNDARY LEAKAGE occur through a component which can be isolated from the balance of the Reactor Coolant System, plant operation may continue provided the leaking component is promptly isolated from the Reactor Coolant System since isolation removes the source of potential failure.

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3/4.4.7 CHEMISTRY

The limitations on Reactor Coolant System chemistry ensure that corrosion of the Reactor Coolant System is minimized and reduces the potential for Reactor Coolant System leakage or failure due to stress corrosion. Maintaining the chemistry within the Steady State Limits provides adequate corrosion protection to ensure the structural integrity of the Reactor Coolant System over the life of the plant. The associated effects of exceeding the oxygen, chloride, and fluoride limits are time and temperature dependent. Corrosion studies show that operation may be continued with contaminant concentration levels in excess of the Steady State Limits, up to the Transient Limits, for the specified limited time intervals without having a significant effect on the structural integrity of the Reactor Coolant System. The time interval permitting continued operation within the restrictions of the Transient Limits provides time for taking corrective actions to restore the contaminant concentrations to within the Steady State Limits.

The surveillance requirements provide adequate assurance that concentrations in excess of the limits will be detected in sufficient time to take corrective action.

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3/4.4.8 SPECIFIC ACTIVITY

The limitations on the specific activity of the primary coolant ensure that the resulting 2 hour doses at the site boundary will not exceed an appropriately small fraction of Part 100 limits following a steam generator tube rupture accident in conjunction with an assumed steady state primary-to-secondary steam generator leakage rate of 1.0 GPM. The values for the limits on specific activity represent interim limits based upon a parametric evaluation by the NRC of typical site locations. These values are conservative in that specific site parameters of the Cook Nuclear Plant site, such as site boundary location and meteorological conditions, were not considered in this evaluation. The NRC is finalizing site specific criteria which will be used as the basis for the reevaluation of the specific activity limits of this site. This reevaluation may result in higher limits.

Offsite doses following a main steam line break are limited to 10 percent of the 10 CFR 100 guideline. The restriction is based on a Cook Nuclear Plant site-specific radiological evaluation that assumes a post-accident primary-to-secondary leak rate of 120 gpm in the faulted loop and a primary coolant specific activity concentration corresponding to 1% fuel defects (approximately 4.6 microCuries/gram dose equivalent I-131), rather than a specific activity of 1.0 microCuries dose equivalent I-131.

Reducing T_{avg} to less than 500°F prevents the release of activity should a steam generator tube rupture since the saturation pressure of the primary coolant is below the lift pressure of the atmospheric steam relief valves. The surveillance requirements provide adequate assurance that excessive specific activity levels in the primary coolant will be detected in sufficient time to take corrective action. Information obtained on iodine spiking will be used to assess the parameters associated with spiking phenomena. A reduction in frequency of isotopic analyses following power changes may be permissible if justified by the data obtained.

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