ATTACHMENT 2 TO AEP:NRC:1166L

EXISTING TECHNICAL SPECIFICATION
PAGES MARKED TO REFLECT PROPOSED CHANGES

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.

c. 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.

d. Implementation of the steam generator tube/tube support plate interim plugging criteria for one fuel cycle (Cycle ~~13~~ 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:

Category	Inspection Results
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.

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.

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 requirement 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.

    1. A tube can remain in service if the signal amplitude of a crack indication is less than or equal to ~~1.0~~ 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 ~~1~~ 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 ~~1.0~~ 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 ~~1.0~~ 2.0 volt but less than or equal to ~~4.0~~ 3.6 volts if a rotating pancake probe inspection does not detect degradation. Indications of degradation with a bobbin coil signal amplitude greater than ~~4.0~~ 3.6 volts will be plugged or repaired.

JUL 2 9 1992

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.

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 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).

e. The results of steam line break leakage analysis performed under T/S 4.4.5.4.a.10 will be reported to the Commission prior to restart for Cycle ~~13~~. 14.

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 ~~13~~ 14,

d. 10 GPM IDENTIFIED LEAKAGE from the Reactor Coolant System,

e. Seal line resistance greater than or equal to 2.27E-1 ft/gpm^2 and,

f. 1 GPM leakage from any reactor coolant system pressure isolation valve specified in Table 3.4-0.

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, except when:

1. The leakage is less than or equal to 5.0 gpm, and

2. The most recent measured leakage does not exceed the previous measured leakage* by an amount that reduces the

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* To satisfy ALARA requirements, measured leakage may be measured indirectly (as from the performance of pressure indicators) if accomplished in accordance with approved procedures and supported by computations showing that the method is capable of demonstrating valve compliance with the leakage criteria.

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

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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 ~~13~~ 14). 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 opera ng 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

which will limit the

Maintaining an operating leakage limit of 150 gpd per steam generator (600 gpd total) for Fuel Cycle ~~13~~ 14 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 ~~1 gpm~~ 12.6 gpm. ~~This is less than the 120 gpm used to~~, calculate ~~the~~ 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 ~~1~~ 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 ~~1~~ 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.

The Surveillance Requirements for RCS 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 Pressure Isolation Valves is IDENTIFIED LEAKAGE and will be considered as a portion of the allowed limit.

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.

ATTACHMENT 3 TO AEP:NRC:1166L

PROPOSED REVISED
TECHNICAL SPECIFICATION PAGES

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.

c. 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.

d. 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:

Category	Inspection Results
C-1	Less than 5% of the total tubes inspected are degraded tubes and none of the inspected tubes are defective.
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C-3	More than 10% of the total tubes inspected are degraded tubes or more than 1% of the inspected tubes are defective.

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.

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 requirement 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.

    1. A tube can remain in service if the signal amplitude of a crack indication is less than or equal to 2.0 volts, 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 volts 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 volts 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.

SURVEILLANCE REQUIREMENTS (Continued)

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.

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 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).

e. The results of steam line break leakage analysis performed under T/S 4.4.5.4.a.10 will be reported to the Commission prior to restart for Cycle 14.

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 14,

d. 10 GPM IDENTIFIED LEAKAGE from the Reactor Coolant System,

e. Seal line resistance greater than or equal to 2.27E-1 ft/gpm^2 and,

f. 1 GPM leakage from any reactor coolant system pressure isolation valve specified in Table 3.4-0.

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, except when:

1. The leakage is less than or equal to 5.0 gpm, and

2. The most recent measured leakage does not exceed the previous measured leakage* by an amount that reduces the

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* To satisfy ALARA requirements, measured leakage may be measured indirectly (as from the performance of pressure indicators) if accomplished in accordance with approved procedures and supported by computations showing that the method is capable of demonstrating valve compliance with the leakage criteria.

** 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 allowabe primary-to-secondary leak rate is 150 gallons per day per steam generator for one fuel cycle (Cycle 14). 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.

Maintaining an operating leakage limit of 150 gpd per steam generator (600 gpd total) for Fuel Cycle 14 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.

The Surveillance Requirements for RCS 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 Pressure Isolation Valves is IDENTIFIED LEAKAGE and will be considered as a portion of the allowed limit.

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.

ATTACHMENT 4 TO AEP:NRC:1166L

STEAMLINE BREAK RADIOLOGICAL ANALYSIS

AEP-94-218

Westinghouse Electric Corporation

Energy Systems

Box 355
Pittsburgh Pennsylvania 15230-0355

Mr. John Jensen
Nuclear Engineering Department
American Electric Power Service Corporation
One Riverside Plaza
Columbus, OH 43216-6631

February 16, 1994

ET-NSL-OPL-II-94-074

Ref: 1) AEP-94-642

AMERICAN ELECTRIC POWER SERVICE CORPORATION
DONALD C. COOK NUCLEAR PLANT
Radiological Analysis

Dear Mr. Jensen:

Attached for your information and use is the revised description of the analysis performed to determine the allowable steam generator primary to secondary leak rate during a steamline break at the Donald C. Cook Nuclear Plant Unit 1. The attachment was revised to incorporate AEPSC comments. This description supersedes that transmitted by Reference 1.

Should you have any questions or comments, please call Ms. Robin Lapides (412/374-5683) or me.

Very truly yours,

K. F. Matthews
Senior Sales Engineer
Power Systems Field Sales

RSL/bbp

Attachments

cc: R. Lapides - (W)
T. Georgantis - AEPSC
M. Ackerman - AEPSC

An evaluation has been performed to determine the maximum permissible steam generator primary to secondary leak rate during a steam line break for the Donald C. Cook Nuclear Plant Unit 1. The evaluation considered both pre-accident and accident initiated iodine spikes. The results of the evaluation show that the accident initiated spike yields the limiting leak rate. This case was based on a 30 rem thyroid dose at the site boundary and initial primary and secondary coolant iodine activity levels of 1 μCi/gm and 0.1 μCi/gm DE I-131, respectively. A leak rate of 12.6 gpm was determined to be the upper limit for allowable primary to secondary leakage in the SG in the faulted loop. The SG in each of the three intact loops was assumed to leak at a rate of 150 gpd (approximately 0.1 gpm), the Technical Specification LCO.

Thirty rem was selected as the thyroid dose acceptance criteria for a steam line break with an assumed accident initiated iodine spike based on the guidance of the Standard Review Plan (NUREG-0800) Section 15.1.5, Appendix A. Only the release of iodine and the resulting thyroid dose was considered in the leak rate determination. Whole-body doses due to noble gas immersion have been determined, in other evaluations, to be less limiting than the corresponding thyroid doses.

The salient assumptions follow.

- Initial primary coolant iodine activity - 1 μCi/gm DE I-131

- Initial secondary coolant iodine activity - 0.1 μCi/gm DE I-131

- Steam released to the environment (0 to 2 hours)

  - from 3 SG's in the intact loops, 463,000 lb
  - from the affected SG, 99,300 lb (the entire initial SG water mass)

- Iodine partition coefficients for primary-secondary leakage

  - SG's in intact loops, 1.0 (leakage is assumed to be above the mixture level)
  - SG in faulted loop, 1.0 (SG is assumed to steam dry)

- Iodine partition coefficients for activity release due to steaming of SG water

  - SG's in intact loops, 0.1
  - SG in faulted loop, 1.0 (SG is assumed to steam dry)

- Atmospheric dispersion factor (SB 0 to 2 hours), 3.10E-4 sec/m^3

- Thyroid dose conversion factors, ICRP 2

The activity released to the environment due to a main steam line break can be separated into two distinct releases: the release of the iodine activity that has been established in the secondary coolant prior to the accident and the release of the primary coolant iodine activity that is transferred by tube leakage during the accident. Based on the assumptions stated previously, the release of the activity initially contained in the secondary coolant (4 SG's) results in a site boundary thyroid dose of approximately 1.0 rem. This dose is independent of the leak location(s). The dose contribution from 1 gpm of primary-to-

secondary leakage (4 SG's) is 2.25 rem. With the thyroid dose limit of 30 rem and with 1 rem from the initial activity contained in the secondary coolant, the total allowable primary-to-secondary leak rate is 30 rem - 1 rem/2.25 rem per gpm, or 12.9 gpm. Allowing 0.1 gpm per each of the 3 intact SGs leaves 12.9 - 0.3 or 12.6 gpm for the SG on the faulted loop.