MAY 5 1977

Dockets Nos. 50-269
50-270
and 50-287

Duke Power Company
ATTN: Mr. William O. Parker, Jr.
Vice President - Steam Production
Post Office Box 2178
422 South Church Street
Charlotte, North Carolina 28242

Gentlemen:

By letter dated November 30, 1976, you requested an amendment to the license for the Oconee Nuclear Station to incorporate requirements concerning the operability and inservice inspections of steam generators. Your request was in response to our letter dated September 21, 1976, which provided model Technical Specifications to be adapted to the Oconee Technical Specifications. We find that your response does not address all of the requirements covered in our model Technical Specifications. For those requirements you have addressed, we request additional information.

In our letter of September 21, 1976, we provided model Technical Specifications which included a reactor coolant leakage limit of 1 GPM through the steam generator tubes. Your letter of November 30, 1976, stated without justification, that a 1 GPM leakage rate is overly restrictive and that the 10 GPM now allowed by the Oconee Technical Specifications is adequate. It is our position that this leakage rate should be limited to 1 GPM, particularly in view of the steam generator tube leaks which have been occuring at Oconee and other PWR facilities. It is requested that you submit a request for change to the Oconee Technical Specifications that limits the leakage rate to 1 GPM or provide detailed justification for not doing so.

An analysis performed by us shows that with a 1 GPM reactor coolant-to-secondary leakage rate, dose rates from postulated accidents would be well below the limits of 10 CFR Part 100. This analysis assumed that the reactor coolant activity was 1.0 $\mu ci/_g$ and the secondary coolant activity was 0.1 $\mu ci/_g$.

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| OFFICE ➤ | | | | | | |
| SURNAME ➤ | | | | | | |
| DATE ➤ | | | | | | |

We have reviewed the Oconee Technical Specifications and find that they do not include iodine activity limit for the reactor coolant and the iodine limit for the secondary coolant is well above that assumed in our analysis. It is requested that you submit a request for a change to the Oconee Technical Specifications that would limit the iodine activity to 1.0 $\mu ci/_{gm}$ and 0.1 $\mu ci/_{gm}$ in the reactor and secondary coolants, respectively. Enclosure 1 is a copy of the B&W Standard Technical Specifications which you should use for guidance.

Enclosure 2 contains comments we have on your proposed Steam Generator Inservice Inspection Technical Specifications. It is requested that you respond to these comments by modifying your proposed Technical Specifications to conform with the model Technical Specifications provided in our letter of September 21, 1976, or by providing justification for any deviations.

It is requested that you respond to the requests herein within 45 days of receipt of this letter.

Sincerely,

**Original signed by**

A. Schwencer, Chief
Operating Reactors Branch #1
Division of Operating Reactors

Enclosures:
1. B&W Standard Technical Specifications
2. NRC Comments on Steam Generator Inservice Inspection

cc w/encl:
See next page

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

May 5, 1977

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and 50-287

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Vice President - Steam Production
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It is requested that you respond to the requests herein within 45 days of receipt of this letter.

Sincerely,

A. Schwencer

A. Schwencer, Chief
Operating Reactors Branch #1
Division of Operating Reactors

Enclosures:
1. B&W Standard Technical Specifications
2. NRC Comments on Steam Generator Inservice Inspection

cc w/encl:
See next page

cc: Mr. William L. Porter
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Charlotte, North Carolina 28242

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700 Shoreham Building
806-15th Street, NW.,
Washington, D.C. 20005

Oconee Public Library
201 South Spring Street
Walhalla, South Carolina 29691

DEFINITIONS

c. Reactor coolant system leakage through a steam generator to the secondary system.

UNIDENTIFIED LEAKAGE

1.15 UNIDENTIFIED LEAKAGE shall be all leakage which is not IDENTIFIED LEAKAGE or CONTROLLED LEAKAGE.

PRESSURE BOUNDARY LEAKAGE

1.16 PRESSURE BOUNDARY LEAKAGE shall be leakage (except steam generator tube leakage) through a non-isolable fault in a Reactor Coolant System component body, pipe wall or vessel wall.

CONTROLLED LEAKAGE

1.17 CONTROLLED LEAKAGE shall be that seal water flow supplied to the reactor coolant pump seals.

QUADRANT POWER TILT

1.18 QUADRANT POWER TILT is defined by the following equation and is expressed in percent.

QUADRANT POWER TILT =

$$100\left(\frac{\text{Power in any core quadrant}}{\text{Average power of all quadrants}} - 1\right)$$

DOSE EQUIVALENT I-131

1.19 DOSE EQUIVALENT I-131 shall be that concentration of I-131 (μCi/gram) which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134 and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Table III of TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites."

$\bar{E}$ - AVERAGE DISINTEGRATION ENERGY

1.20 $\bar{E}$-AVERAGE DISINTEGRATION ENERGY shall be the average (weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the sum of the average beta and gamma energies

per disintegration (in MeV) for isotopes, other than iodines, with half lives greater than 15 minutes, making up at least 95% of the total non-iodine activity in the coolant.

STAGGERED TEST BASIS

1.21 A STAGGERED TEST BASIS shall consist of:

a. A test schedule for n systems, subsystems, trains or designated components obtained by dividing the specified test interval into n equal subintervals.

b. The testing of one system, subsystem, train or designated components at the beginning of each subinterval.

FREQUENCY NOTATION

1.22 The FREQUENCY NOTATION specified for the performance of Surveillance Requirements shall correspond to the intervals defined in Table 1.2.

AXIAL POWER IMBALANCE

1.23 AXIAL POWER IMBALANCE shall be the THERMAL POWER in the top half of the core expressed as a percentage of RATED THERMAL POWER minus the THERMAL POWER in the bottom half of the core expressed as a percentage of RATED THERMAL POWER.

SHIELD BUILDING INTEGRITY

1.24 SHIELD BUILDING INTEGRITY shall exist when:

a. Each door in each access opening is closed except when the access opening is being used for normal transit entry and exit, then at least one door shall be closed.

b. The shield building filtration system is OPERABLE.

c. The sealing mechanism associated with each penetration (e.g. welds, bellows or O-rings) is OPERABLE.

REACTOR PROTECTION SYSTEM RESPONSE TIME

1.25 The REACTOR PROTECTION SYSTEM RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its trip setpoint at the channel sensor until power interruption at the control rod drive breakers.

## LIMITING CONDITION FOR OPERATION

3.4.8 The specific activity of the primary coolant shall be limited to:

a. $\leq$ 1.0 μCi/gram DOSE EQUIVALENT I-131.

b. $\leq$ 100/$\bar{E}$ μCi/gram.

APPLICABILITY: MODES 1, 2, 3, 4 and 5.

ACTION:

MODES 1, 2 and 3*.

a. With the specific activity of the primary coolant > 1.0 μCi/gram DOSE EQUIVALENT I-131 but within the allowable limit (below and to the left of the line) shown on Figure 3.4-1, operation may continue for up to 48 hours provided that operation under these circumstances shall not exceed 10% of the unit's total yearly operating time. The provisions of Specification 3.0.4 are not applicable.

b. With the specific activity of the primary coolant > 1.0 μCi/gram DOSE EQUIVALENT I-131 for more than 48 hours during one continuous time interval or exceeding the limit line shown on Figure 3.4-1, be in at least HOT STANDBY with $T_{avg}$ < (500)°F within 6 hours.

c. With the specific activity of the primary coolant > 100/$\bar{E}$ μCi/gram, be in at least HOT STANDBY with $T_{avg}$ < (500)°F within 6 hours.

MODES 1, 2, 3, 4 and 5:

a. With the specific activity of the primary coolant > 1.0 μCi/gram DOSE EQUIVALENT I-131 or > 100/$\bar{E}$ μCi/gram, perform the sampling and analysis requirements of item 4 a) of Table 4.4-4 until the specific activity of the primary coolant is restored to within its limits. A REPORTABLE OCCURRENCE shall be prepared and submitted to the Commission pursuant to Specification 6.9.1. This report shall contain the results of the specific activity analyses together with the following information:

*With $T_{avg} \geq$ (500)°F.

ACTION: (Continued)

1. Reactor power history starting 48 hours prior to the first sample in which the limit was exceeded.

2. Fuel burnup by core region.

3. Clean-up flow history starting 48 hours prior to the first sample in which the limit was exceeded.

4. History of de-gassing operations, if any, starting 48 hours prior to the first sample in which the limit was exceeded.

5. The time duration when the specific activity of the primary coolant exceeded 1.0 μCi/gram DOSE EQUIVALENT I-131.

## SURVEILLANCE REQUIREMENTS

4.4.8 The specific activity of the primary coolant shall be determined to be within the limits by performance of the sampling and analysis program of Table 4.4-4.

## TABLE 4.4-4

## PRIMARY COOLANT SPECIFIC ACTIVITY SAMPLE AND ANALYSIS PROGRAM

| TYPE OF MEASUREMENT AND ANALYSIS | SAMPLE AND ANALYSIS FREQUENCY | MODES IN WHICH SAMPLE AND ANALYSIS REQUIRED |
|---|---|---|
| 1. Gross Activity Determination | At least once each 72 hours | 1, 2, 3, 4 |
| 2. Isotopic Analysis for DOSE EQUIVALENT I-131 Concentration | 1 per 14 days | 1 |
| 3. Radiochemical for $\bar{E}$ Determination | 1 per 6 months* | 1 |
| 4. Isotopic Analysis for Iodine Including I-131, I-133, and I-135 | a) Once per 4 hours, whenever the specific activity exceeds 1.0 µCi/gram DOSE EQUIVALENT I-131 or 100/$\bar{E}$ µCi/gram. | 1#, 2#, 3#, 4#, 5# |
| | b) One sample between 2 and 6 hours following a THERMAL POWER change exceeding 15 percent of the RATED THERMAL POWER within a one hour period. | 1, 2, 3 |

#Until the specific activity of the primary coolant system is restored within its limits.

*Sample to be taken after a minimum of 2 EFPD and 20 days of POWER OPERATION have elapsed since the reactor was last subcritical for 48 hours or longer.

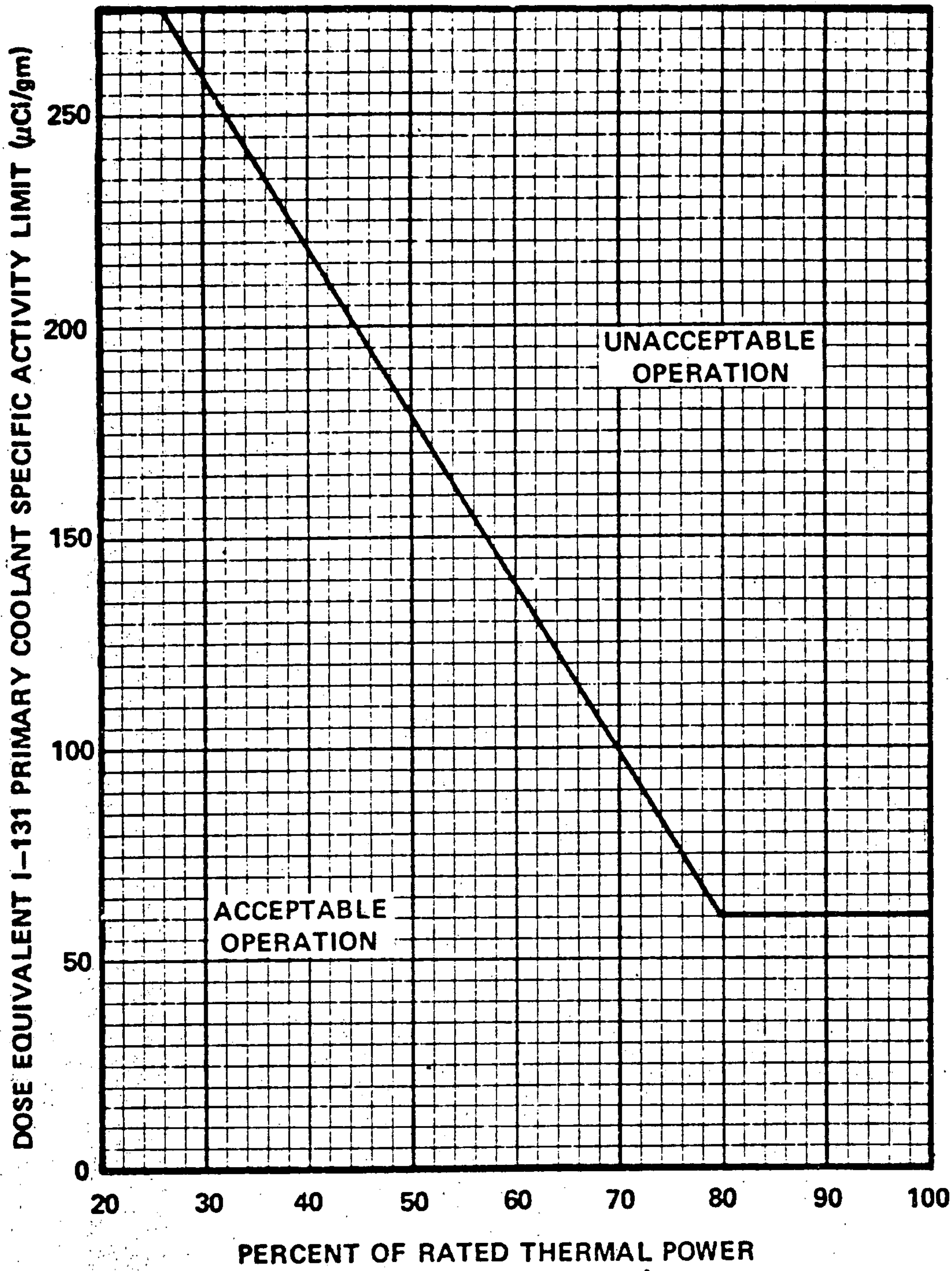


FIGURE 3.4-1

DOSE EQUIVALENT I–131 Primary Coolant Specific Activity Limit Versus Percent of RATED THERMAL POWER with the Primary Coolant Specific Activity > 1.0 $\mu$Ci/gram Dose Equivalent I-131

## ACTIVITY

### LIMITING CONDITION FOR OPERATION

3.7.1.4 The specific activity of the secondary coolant system shall be $\leq$ 0.10 μCi/gram DOSE EQUIVALENT I-131.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the specific activity of the secondary coolant system > 0.10 μCi/gram DOSE EQUIVALENT I-131, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

### SURVEILLANCE REQUIREMENTS

4.7.1.4 The specific activity of the secondary coolant system shall be determined to be within the limit by performance of the sampling and analysis program of Table 4.7-2.

## TABLE 4.7-2

## SECONDARY COOLANT SYSTEM SPECIFIC ACTIVITY SAMPLE AND ANALYSIS PROGRAM

| TYPE OF MEASUREMENT AND ANALYSIS | SAMPLE AND ANALYSIS FREQUENCY |
|---|---|
| 1. Gross Activity Determination | At least once per 72 hours |
| 2. Isotopic Analysis for DOSE EQUIVALENT I-131 Concentration | a) 1 per 31 days, whenever the gross activity determination indicates iodine concentrations greater than 10% of the allowable limit.<br><br>b) 1 per 6 months, whenever the gross activity determination indicates iodine concentrations below 10% of the allowable limit. |

BASES

### 3/4.4.7 CHEMISTRY

The limitations on Reactor Coolant System chemistry ensure that corrosion of the Reactor Coolant System is minimized and reduce the potential for Reactor Coolant System leakage or failure due to stress corrosion. Maintaining the chemistry within the Steady State Limits shown on Table 3.4-1 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.

### 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 the Part 100 limit 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 the specific site parameters of the 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.

The ACTION statement permitting POWER OPERATION to continue for limited time periods with the primary coolant's specific activity > 1.0 μCi/gram DOSE EQUIVALENT I-131, but within the allowable limit shown on Figure 3.4-1, accommodates possible iodine spiking phenomenon which may occur following changes in THERMAL POWER. Operation with specific activity levels exceeding 1.0 μCi/gram DOSE EQUIVALENT I-131 but within the limits shown on Figure 3.4-1 must be restricted to no more than 10 percent of the units yearly operating time since the activity levels allowed by Figure 3.4-1 increase the 2 hour thyroid dose at the site boundary by a factor of up to 20 following a postulated steam generator tube rupture.

Reducing $T_{avg}$ to < (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.

## 3/4.4.9 PRESSURE/TEMPERATURE LIMITS

All components in the Reactor Coolant System are designed to withstand the effects of cyclic loads due to system temperature and pressure changes. These cyclic loads are introduced by normal load transients, reactor trips, and startup and shutdown operations. The various categories of load cycles used for design purposes are provided in Section (        ) of the FSAR. During heatup and cooldown, the rates of temperature and pressure changes are limited so that the maximum specified heatup and cooldown rates are consistent with the design assumptions and satisfy the stress limits for cyclic operation.

During heatup, the thermal gradients in the reactor vessel wall produce thermal stresses which vary from compressive at the inner wall to tensile at the outer wall. These thermal induced compressive stresses tend to alleviate the tensile stresses induced by the internal pressure. Therefore, a pressure-temperature curve based on steady state conditions (i.e., no thermal stresses) represents a lower bound of all similar curves for finite heatup rates when the inner wall of the vessel is treated as the governing location.

## 3/4.7.1.2 AUXILIARY FEEDWATER SYSTEMS

The OPERABILITY of the auxiliary feedwater systems ensures that the Reactor Coolant System can be cooled down to less than (305)°F from normal operating conditions in the event of a total loss of offsite power.

Each electric driven auxiliary feedwater pump is capable of delivering a total feedwater flow of (350) gpm at a pressure of (1133) psig to the entrance of the steam generators. Each steam driven auxiliary feedwater pump is capable of delivering a total feedwater flow of (700) gpm at a pressure of (1133) psig to the entrance of the steam generators. This capacity is sufficient to ensure that adequate feedwater flow is available to remove decay heat and reduce the Reactor Coolant System temperature to less than (305)°F where the Decay Heat Removal System may be placed into operation.

## 3/4.7.1.3 CONDENSATE STORAGE TANK

The OPERABILITY of the condensate storage tank with the minimum water volume ensures that sufficient water is available for cooldown of the Reactor Coolant System to less than (305)°F in the event of a total loss of offsite power or of the main feedwater system. The minimum water volume is sufficient to maintain the RCS at HOT STANDBY conditions for (  ) hours with steam discharge to atmosphere concurrent with loss of offsite power. The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics.

## 3/4.7.1.4 ACTIVITY

The limitations on secondary system specific activity ensure that the resultant offsite radiation dose will be limited to a small fraction of 10 CFR Part 100 limits in the event of a steam line rupture. This dose includes the effects of a coincident 1.0 GPM primary to secondary tube leak in the steam generator of the affected steam line. These values are consistent with the assumptions used in the safety analyses.

## 3/4.7.1.5 MAIN STEAM LINE ISOLATION VALVES

The OPERABILITY of the main steam line isolation valves ensures that no more than one steam generator will blowdown in the event of a steam line rupture. This restriction is required to 1) minimize the

ENCLOSURE 2

COMMENTS ON OCONEE UNITS 1, 2, AND 2 PROPOSED
TECHNICAL SPECIFICATIONS FOR STEAM GENERATOR
INSERVICE INSPECTION

1. Table 4.17-1 which specifies steam generator tube sample size, inspection result classification, and the corresponding action required for each sample inspection has several discrepancies with the standard technical specifications:

   a. If the results of the first sample inspection fall in the C-2 category, the corrective action should be the plugging of the defective tubes and inspection of $6\frac{N}{n}\%$ additional tubes in that steam generator. Where N is the number of steam generators in the unit, and n is the number of steam generators inspected during an inspection.

   b. If the results of the first sample inspection of a steam generator fall in the C-3 category, the corrective action should include inspection of all tubes in the steam generator, plugging of defective tubes, and inspection of an additional $6\frac{N}{n}\%$ of the tubes in each other steam generator.

   c. Corrective actions corresponding to results of second and third sample inspections have not been specified in the table. These actions should be specified in accordance with Table 4.4-2 of the standard technical specifications. Guidance for subsequent (second and third) sample inspections as stated in paragraph 4.17.1,b of the proposed technical specifications is unacceptable.

The sample sizes required during each sample inspection are clearly specified in Table 4.4-2 of the standard technical specifications. Deviation from the specified sample sizes will require a statistical analysis justifying the proposed sample sizes.

2. Paragraphs 4.4.5.2,b, and c of the standard technical specifications should be included under paragraph 4.17.1,b of the proposed technical specifications.

3. Paragraph 4.17.2,b should read the same as paragraph 4.4.5.3,b in the standard technical specification.

4. Paragraph 4.17.2,c should also specify that unscheduled inservice inspections shall be performed on each steam generator in accordance with the first sample inspection during the shutdown subsequent to:

   a. A seismic occurrence greater than the Operating Basis Earthquake,

   b. A loss-of-coolant accident requiring actuation of the engineered safeguards, or

   c. A main steam line or feedwater line break.

5. Justification for the proposed 40% plugging limit must be provided in accordance with Regulatory Guide 1.121.

6. The term "unserviceable" used in the definition of plugging limit must be defined.

7. The definition of defect should read as follows:

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

8. The basis should state:

   a. Cracks having a primary-to-secondary leakage less than the specified 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 unscheduled inspection, during which the leaking tubes will be located and plugged.

   b. Cases when the results of any steam generator tubing inservice inspection fall into category C-3 will be considered by the Commission on a case-by-case basis and may result in a requirement for analyses, laboratory examination, tests, additional eddy current inspection, and revision of the Technical Specification, if necessary.

9. The subject of operability of steam generators as discussed in sections 3.4.5 and 4.4.5.6 of the standard technical specifications should be addressed.