3.1.6 Leakage

Specification

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

3.1.6.2 If unidentified reactor coolant leakage (excluding normal evaporative losses) exceeds 1 gpm or if any reactor coolant leakage is evaluated as unsafe, the reactor shall be shutdown within 24 hours of detection.

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

3.1.6.4 If ~~at any time, the leakage through the Unit 1~~ The Total Leakage Through the tubes of Both steam generators ~~tubes~~ equals or exceeds ~~0.3~~ 1.0 gpm, a reactor shutdown shall be initiated within 4 hours and the reactor shall be in a cold condition within the next 36 hours. If the leakage is less than ~~0.3~~ gpm, an assessment shall be made whether operations may be continued safely or the plant should be shutdown. ~~In either case, the NRC shall be notified in accordance with Section 6.6.2.1.~~

3.1.6.5 If reactor shutdown is required by Specification 3.1.6.1, 3.1.6.2 or 3.1.6.3, the rate of shutdown and the conditions of shutdown shall be determined by the safety evaluation for each case and justified in writing as soon thereafter as practicable.

3.1.6.6 Action to evaluate the safety implication of reactor coolant leakage shall be initiated within 4 hours of detection. The nature, as well as the magnitude, of the leak shall be considered in this evaluation. The safety evaluation shall assure that the exposure of offsite personnel to radiation is within the guidelines of 10 CFR 20.

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

3.1.6.8 When the reactor is critical and above 2% power, two reactor coolant leak detection systems of different operating principles shall be operable, with one of the two systems sensitive to radioactivity. The systems sensitive to radioactivity may be out-of-service for 48 hours provided two other means to detect leakage are operable.

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

3.1.6.10

a. The maximum allowable leakage for valves CF-12, CF-14, LP-47 and LP-48 shall be as follows:

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

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

f. Repair Limit means the imperfection depth beyond which the tube shall be either removed from service by plugging or repaired by sleeving because it may become unserviceable prior to the next inspection; it is equal to 40% of the nominal tube or sleeve wall thickness.

The Babcock and Wilcox process (or method) equivalent to the method described in report, BAW-1823P, Revision 1 will be used.

g. Unserviceable describes the condition of a tube if it leaks or contains a defect large enough to affect its structural integrity in the event of an Operating Basis Earthquake, a loss-of coolant accident, or a steam line or feedwater line break as specified in Specification 4.17.4.

h. Tube Inspection means an inspection of the steam generator tube from the point of entry completely to the point of exit.

4.17.6 Reports

a. The number of tubes plugged or repaired in each steam generator shall be reported to the ~~Director,~~ Regional Administrator Office of Inspection and Enforcement, Region II, within 30 days following the completion of the plugging or repair procedure.

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

1. Number and extent of tubes inspected.

2. Location and percent of wall-thickness penetration for each indication of a degraded tube.

3. Identification of tubes plugged or repaired.

~~c. Results of steam generator tube inspections which fall into Category C-3 and require prompt notification of the NRC shall be reported pursuant to Specification 6.6.2.1.a prior to resumption of plant operation. The written followup of this report shall provide a description of investigations conducted to determine cause of the tube degradation and corrective measures taken to prevent recurrence.~~

Bases

The program of periodic inservice inspection of steam generators provides the means to monitor the integrity of the tubing and to maintain surveillance in the event there is evidence of mechanical damage or progressive deterioration due to design, manufacturing errors, or operating conditions. Inservice inspection of the steam generator tubing also provides a means of characterizing the nature and cause of any tube degradation so that corrective measures may be taken.

TABLE 4.17-1

STEAM GENERATOR TUBE INSPECTION

1st SAMPLE INSPECTION			2nd SAMPLE INSPECTION		3rd SAMPLE INSPECTION	
SAMPLE SIZE	RESULT	ACTION REQUIRED	RESULT	ACTION	RESULT	ACTION
A minimum of S Tubes per S.G. (1)	C-1	None	N/A	N/A	N/A	N/A
	C-2	Plug or repair defective tubes and inspect additional 2S tubes in	C-1	None	N/A	
			C-2	Plug or repair defective tubes and inspect additional 4S tubes in this S.G.	C-1	N/A
					C-2	Plug or repair defective tubes
					C-3	Plug or repair defective tubes and perform action for C-3 result of 1st Sample
			C-3	Plug or repair defective tubes and perform actions for C-3 results of 1st Sample	N/A	N/A
	C-3	Inspect 6S tubes in this S.G., plug or repair defective tubes and inspect 2S tubes in the other S.G. Perform follow-on inspections in the other S.G. in accordance with results of the above inspection as applied to Table 4.17.1 ~~Prompt notification to NRC pursuant to specification 6.6.2.1.a~~	C-1	N/A	N/A	N/A
			C-2	N/A	N/A	N/A
			C-3 (2)	(a) if defects can be localized to an affected area, inspect all tubes in affected area and plug or repair defective tubes. (b) If defects cannot be localized to an affected area, inspect all tubes in this S.G. and plug or repair defectives tubes.	C-1	N/A
					C-2	N/A
					C-3	N/A

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

(2) Affected and unaffected areas shall be determined in the manner described in the Bases of this specification. The definition of these areas will be reported to the NRC when they are determined.

Attachment 2

Duke Power Company
Oconee Nuclear Station

Technical Justification for the Proposed Changes to
Technical Specification 3.1.6.4

Introduction:

In a letter dated October 2, 1987 the NRC requested that a technical specification limit of 1.0 gpm total primary to secondary leakage through both steam generators be established on Oconee Nuclear Station Units 2 and 3. At present, Oconee Unit 1 has a technical specification leakage limit of 0.3 gpm with no specific primary to secondary leakage rate for Units 2 and 3. The following discussion provides technical justification for the proposed revision to the Technical Specification 3.1.6.4 (Attachment 1) as requested by the NRC. The proposed revised Technical Specification 3.1.6.4 would require a leakage limit of 1.0 gpm total primary to secondary leakage through both steam generators per unit for all three Oconee units.

Proposed Technical Specification Revisions

The proposed technical specification revision (Attachment 1) includes two changes to the current Technical Specification 3.1.6.4. These changes are:

(a) The limit of 1.0 gpm total primary to secondary leakage through both steam generators per unit for all Oconee units compared to a present limit of 0.3 gpm for Unit 1 and no specific limit for Units 2 and 3.

(b) Deletion of the last sentence in the current Technical Specification 3.1.6.4 which require the NRC notification of steam generator tube leaks "in accordance with Section 6.6.2.1."

Other changes include:

(c) Deletion of the current Technical Specification 4.17.6.c on page 4.17-4 which require the NRC notification of the results of steam generator tube inspections which fall into Category C-3 "pursuant to Specification 6.6.2.1.a prior to resumption of plant operation."

(d) Deletion of the current requirement for a "prompt notification to NRC pursuant to Specification 6.6.2.1.a" in Table 4.17-1, Item C-3, on page 4.17-6.

(e) Changing of the term "Director" to "Regional Administrator" in present Specification 4.17.6.a on page 4.17-4.

The following paragraphs provide technical justification for each of these changes.

Justification for the Proposed Changes

As noted in the NRC letter of October 2, 1987, Duke Power Company (Duke) in a letter dated June 21, 1977 submitted a proposed change to incorporate a primary

to secondary leakage limit of 1.0 gpm into the Technical Specifications for Oconee Units 1, 2 and 3. The NRC took no action to address Duke's proposed 1.0 gpm limit for Oconee units. However, in an October 4, 1977 letter from the NRC a primary to secondary leakage limit of 0.3 gpm was imposed on Oconee Unit 1 due to a "unique state of tube degradation" in the Unit 1 once through steam generators (OTSGs). The proposed leakage limit in this submittal provides a consistent leakage limit of 1.0 gpm for all three units in that it adds a new limit for Units 2 and 3 and increases the limit for Unit 1 from 0.3 gpm to 1.0 gpm. The limit of 1.0 gpm is consistent with established limits for a number of other B&W plants and Standard Technical Specification (STS).

The increase in the leakage limit for Unit 1 is justified since corrective actions taken have substantially decreased and mitigated the concerns regarding the "unique state of tube degradation in the Oconee Unit 1 OTSGs". Since commercial operation on Oconee Unit 1, two regions in the OTSGs have been identified as contributing to this unique degradation. The regions are: (1) the lane and wedge region and (2) the periphery region.

To better understand the cause of these defect mechanisms and subsequent tube leaks, a remedial program was initiated in the late 1970's. The program included special instrumentation and operational testing, expanded nondestructive evaluation, and tube sample removal and examination. The lane and wedge region as presently defined was identified in the mid 1980's. Sixty-four (64) percent of the leaks as a result of tube degradation in the lane and wedge region at Oconee Unit 1 were located in this region. Nine of the pulled tubes examined had identified corrosion assisted fatigue as the primary failure mechanism. This failure mechanism occurs in this region because of the unique mechanical and thermal hydraulics characteristics of the Oconee OTSGs.

Various alternatives were evaluated to mitigate the consequences of leaks in this region and sleeving was selected as the preferred method. By letter dated September 1, 1987 the NRC approved technical specifications (Amendment Nos. 161, 161 and 158 to Facility Operating Licenses Nos. DPR-38, DPR-47 and DPR-55) and accepted a sleeving qualification program to allow sleeving at Oconee Units 1, 2 and 3. Sleeves are considered to be structural members that meet all normal, upset, emergency, and faulted conditions resulting from normal operation and accident transients. Therefore, sleeving will provide a secondary pressure boundary. In 1987, sleeves were installed in a bounding lane and wedge region in both Oconee Unit 1 OTSGs as a preventive measure.

The second region identified as having a "unique state of tube degradation" is the periphery region. This region is defined by the tubes outside of the outer most stayrods and consists of approximately 7000 tubes. Fourteen (14) percent of the leaks as a result of tube degradation at Oconee Unit 1 have been located in this region. The primary failure mechanism as identified by 4 pulled tubes is erosion-corrosion at the 14th tube support plate. This failure mechanism is believed to be caused by impingment of micron sized particles accelerated by local flow perturbations around debris. This degradation is confined to the defined region and has a growth rate typically requiring several cycles to propagate to failure. Therefore, additional inspections are effective in controlling leakage. Since November 1979, a periphery region inspection has been performed on "1B" OTSG every refueling outage and no leaks have occurred as a result of erosion-corrosion.

Steam generator programs are continually evolving to improve the reliability of the steam generators. In addition to the improvements previously mentioned, the following enhancements have been implemented:

(1) Improved eddy current techniques.

(2) Enhanced steam generator data trending.

(3) Enhanced leak testing techniques.

(4) Improved steam generator operating procedures.

(5) Improved Chemistry.

(6) Improved preventive maintenance (i.e., Sleeving).

These improvements have allowed Duke Power Company to effectively manage leakage from tube degradation.

Duke Power Company has continually reviewed the steam generator degradation within Unit 2 Steam Generator. There has not been a tube leak within the lane and wedge region in the last eleven years, therefore, sleeving is not considered necessary as a remedial action to mitigate leakage within that region. Also, the periphery region in Unit 2 OTSGs has been monitored for degradation like that found on Unit 1. To date, neither the results of this review nor previous history of leaking tubes warrants the inspection of this region. However, the steam generator review process will continue, but at this time, the periphery region degradation is not a problem.

The lane and wedge region in the Unit 3 OTSGs will also be sleeved during an upcoming refueling outage in August, 1988 as a preventive measure. As in Unit 1, sleeving will significantly reduce the concern for tube leaks in this region of the Unit 3 steam generators. The Unit 3 periphery region has been monitored for degradation similar to that identified in Unit 1. An initial periphery region inspection is planned for August 1988. Therefore, the concern for periphery region degradation is addressed.

The improvements noted for Unit 1 also apply to Units 2 and 3. In addition, Duke's practice via administrative procedures is to remove the unit from service when the tube leakage is great enough to be located by the nitrogen bubble test. This philosophy provides adequate margin to prevent tube ruptures while allowing cost effective and reliable plant operation. Furthermore, remedial programs such as sleeving and expanded inspections have been effective in controlling steam generator tube leakage and therefore warrant increasing the restrictive leakage limit of 0.3 gpm to 1.0 gpm for Unit 1 and establishing a new leakage limit of 1.0 gpm for Units 2 and 3.

<u>Safety Analysis</u>

Duke has also analyzed the impact of the proposed primary to secondary leak rate of 1.0 gpm on current safety analyses. There would be no significant impact on any of the current safety analyses found in Chapter 15 of the Oconee Final Safety Analysis Report (FSAR) as a result of the proposed Technical Specification 3.1.6.4 revision to require a limit of 1.0 gpm total primary to secondary leakage per unit. This applies to all three Oconee units. This is due to the fact that those Chapter 15 events analyzed which address primary to secondary leak rate, the loss of load (Section 15.8.2) and the main steam line break with a concurrent steam generator tube rupture (Section 15.13.5), assume a 1.0 gpm tube leak for the environmental consequences calculations. Therefore, a Technical Specification requiring a limit of 1.0 gpm total leakage would not be outside the

current design basis of the plant. Since there is no significant impact on the Chapter 15 safety analysis, there would also be no significant increased probability of public exposure and reduction in the margin of safety.

In addition, the last sentence in the current Technical Specification 3.1.6.4 which reads "In either case, the NRC shall be notified in accordance with Section 6.6.2.1" is deleted in the proposed revision of the Technical Specification 3.1.6.4 as an administrative change. Also, the current Technical Specification 4.17.6.c on page 4.17-4 and the requirement for a "Prompt notification to NRC pursuant to Specification 6.6.2.1.a" in Table 4.17-1, under Item C-3, on page 4.17-6 are also deleted as administrative changes. These deleted requirements for reporting steam generator tube leaks were based on a previous version of Technical Specification 6.6.2 for conditions leading to operation in a "degraded mode" permitted by a limiting condition for operation or shutdown required by a limiting condition for operation. Following the issuance of the new Licensee Event Report Rule, 10 CFR 50.73, which does not require the reporting of degraded modes, Technical Specification 6.6.2 was also revised. The current version of Technical Specification 6.6.2 does not require the reporting of a degraded mode permitted by a limiting condition for operation, however, it references 10 CFR 50.73 requirements for reportable events.

Furthermore, the term "Director" in Specification 4.17.6 is changed to "Regional Administrator" to reflect organizational changes within the NRC. This change is also an administrative change and has no significant safety impact.

Finally, the proposed new paragraph, under item (b) in Attachment 1, for addition to the "Bases" section beginning on page 3.1-14a of the Oconee Technical Specifications has been submitted pursuant to 10 CFR 50, 50.36(a) and is not to be considered as part of the technical specifications. Specifically, 10 CFR 50, 50.36(a) states that "a summary statement of the bases or reasons for such specifications, other than those covering administrative controls, shall also be included in the application, but not become part of technical specifications". As such the added paragraph to the "Bases" section is provided to advise the NRC of the need to update the bases. The requirements of 10 CFR 50.92 for determination of a no significant hazards consideration are not applicable for this change and the NRC approval is not required.

Conclusion

In summary, Duke concludes that the proposed Technical Specification 3.1.6.4 revision which requires a limit of 1.0 gpm total primary to secondary leakage per unit for all Oconee units is justified for the following reasons:

(1) The consistent 1.0 gpm total primary to secondary leakage per unit for all Oconee units is in line with the STS limit of 1.0 gpm and the NRC recommendations detailed in letter dated October 2, 1987. This limit corresponds to the 1.0 gpm leakage assumed in the Oconee FSAR for purposes of demonstrating that the consequences of design basis accidents do not exceed 10 CFR Part 100 guidelines.

(2) Significant improvements in preventive maintenance and surveillance of the Oconee OTSGs have largely mitigated earlier concerns regarding OTSG tube degradation. Corrective actions such as sleeving, improved eddy current techniques, enhanced steam generator data trending, enhanced leak testing techniques, and improved steam generator operating procedures have allowed Duke Power Company to effectively manage leakage from tube degradation.

(3) Duke's philosophy and practice via administrative procedures have been and is to reduce power and remove the unit from service with tube leaks far below the proposed 1.0 gpm limit. This philosophy provides adequate assurance to prevent tube rupture, contamination of the secondary system for ALARA considerations, minimize personnel exposures, and compliance with the requirements of 10 CFR 100, while allowing cost effective, safe and reliable plant operation.

(4) The deletion of the reporting requirements in Technical Specifications 3.1.6.4 and 4.17.6.c and Table 4.17-1 for degraded modes is an administrative change. These requirements are based on a previous version of Technical Specification 6.6.2 which required reporting of degraded modes. As mentioned above, this requirement no long exist and has been superseded by 10 CFR 50.73 requirements. However, the reporting requirements of 50.73 are referenced in the current Technical Specification 6.6.2 for reportable events.

Attachment 3

Duke Power Company
Oconee Nuclear Station

No Significant Hazards Consideration Evaluation

Duke Power Company (Duke) has determined that the proposed amendment request poses no significant hazards as defined by NRC regulations in 10 CFR 50.92. This ensures that operation of the facility in accordance with the proposed amendment would not:

(1) involve a significant increase in the probability or consequences of an accident previously evaluated; or

(2) create the possibility of a new or different kind of accident from any accident previously evaluated; or

(3) involve a significant reduction in a margin of safety.

The proposed technical Specification 3.1.6.4 revision in this submittal is in response to the NRC request dated October 2, 1987 and establishes a new total primary to secondary leakage rate limit of 1.0 gpm per unit for all Oconee units. The current Technical Specification 3.1.6.4 requires that:

"If at any time, the leakage through the Unit 1 steam generator tubes equals or exceeds 0.3 gpm, a reactor shutdown shall be initiated within 4 hours and the reactor shall be in a cold condition within the next 36 hours. If the leakage is less than 0.3 gpm, an assessment shall be made whether operations may be continued safely or the plant should be shutdown. In either case, the NRC shall be notified in accordance with Section 6.6.2.1."

The present Oconee Technical Specifications, however, do not include specific primary to secondary leakage rate limits for Units 2 and 3.

The proposed Technical Specification 3.1.6.4 (Attachment 1) has been prepared to require a 1.0 gpm leakage rate limit for all three Oconee Units. The 1.0 gpm limit is in line with the standard technical specifications (STSs) and as proposed in Enclosure 1 and 2 of the NRC letter dated October 2, 1987 which recommends a total primary to secondary leakage limit of 1.0 gpm per unit. Specifically, the proposed change would:

(a) Establish a 1.0 gpm primary to secondary total leakage per unit for Units 2 and 3. This is a new restriction not presently in the Oconee Technical Specifications.

(b) Increase the current limit of 0.3 gpm to 1.0 gpm for Unit 1 for consistency with the other units.

(c) Delete the last sentence in the present Technical Specification 3.1.6.4 which reads, "In either case, the NRC shall be notified in accordance with Section 6.6.2.1."

(d) Delete present Specification 4.17.6.c and the corresponding portion of Table 4.17-1.

(e) Provide an administrative title change in present Specification 4.17.6.a.

The technical justification for these changes is provided in Attachment 2 of this submittal. Furthermore, the Commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered not likely to involve significant hazards consideration. Example (ii) relates to a change that constitutes an additional limitation, restriction, or control not presently included in the Technical Specifications. Example (vii) relates to a change to make a license conform to changes in the regulations, where the license change results in very minor changes to facility operations clearly in keeping with the regulations.

In this case, the change in this request is similar to example (ii) in that an additional limitation and margin of safety is provided by establishing a new primary to secondary leakage limit of 1.0 gpm for Units 2 and 3 which is clearly an improvement in safety and reliability of plant operation. Additionally, the change proposed in this request is similar to example (vii) in that the last sentence in the current Technical Specification 3.1.6.4 is deleted to provide consistency with the new and revised Technical Specification 6.6.2, and with the new LER Rule, 10 CFR 50.73, which does not require the reporting of degraded modes. This also applies to the proposed deletion of Specification 4.17.6.c and affected portion of Table 4.17-1. Specification 4.17.6.a is revised to use the latest NRC administrative title "Regional Administrator" instead of "Director".

The subsequent paragraphs address each of the three standards that are promulgated in 10 CFR 50.92.

First Standard

(Amendment would not) involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed amendment will revise Technical Specification 3.1.6.4 to require a consistent 1.0 gpm total primary to secondary leakage limit per unit for all Oconee Units compared to the current version of the Technical Specification 3.1.6.4 which requires a limit of 0.3 gpm primary to secondary leakage for Unit 1 and no specific limits for Units 2 and 3. The consistent 1.0 gpm total primary to secondary leakage rate is in accordance with the NRC recommendations for Units 2 and 3 detailed in a letter dated October 2, 1987 and in line with the industry limits such as those in STSs. This limit corresponds to the 1.0 gpm leakage assumed in the Oconee FSAR to demonstrate that the consequences of Design Basis Accidents (DBAs) do not exceed 10 CFR 100 limits.

The limit of 0.3 gpm for Unit 1 was imposed by the NRC on October 4, 1977 due to a "unique state of tube degradation" in the Unit 1 OTSGs. As mentioned in the technical justifications (Attachment 2), corrective actions such as sleeving, improved eddy current techniques, enhanced steam generator data trending, enhanced leak testing techniques, and improved operating procedures have allowed Duke to effectively manage leakage from tube degradation. Also, Duke's practice via administrative procedures have been and is to reduce power and shutdown the unit with tube leaks equal or less than the proposed 1.0 gpm limit. Therefore, while the new 1.0 gpm limit would provide adequate assurance to prevent tube

rupture, contamination of the secondary system for ALARA considerations (minimizing personnel exposures) and compliance with the requirements of 10 CFR 100, it also allows cost effective, safe and reliable plant operation. It should be noted that the requirements to initiate shutdown within 4 hours and be in cold condition within 36 hours if the leakage equals or exceeds the limit are unchanged. Therefore, the consequences of a 1.0 gpm total primary to secondary leakage for any of the Oconee Units are bounded by the current FSAR analyses.

Each accident analysis addressed in the Oconee FSAR has been examined with respect to the proposed amendment. The probability of any DBA is not affected by this change, nor are the consequences of a DBA affected by this change. Also, the deletion of the last sentence of the present Technical Specification 3.1.6.4, the deletion of Technical Specification 4.17.6.c, revision to Table 4.17-1 and a title change in Specification 4.17.6.a as explained earlier, are all administrative changes. As such, these administrative changes are not considered to be an initiator or a contributor to any previously evaluated accident. Therefore, the requested amendment will not involve a significant increase in the probability or consequences of a previously evaluated accident.

Second Standard

(Amendment would not) create the possibility of a new or different kind of accident from any kind of accident previously evaluated.

The proposed Technical Specification 3.1.6.4 has been prepared to address the NRC request to provide a specific technical specification limit of 1.0 gpm for total primary to secondary leakage per unit for Units 2 and 3. In addition, the current limit of 0.3 gpm for Unit 1 has been increased to 1.0 gpm for consistency with Units 2 and 3. The proposed change would provide a consistent technical specification limit of 1.0 gpm for all Oconee Units. This proposed revision will not result in any plant modification or operating procedural changes.

The deletion of the last sentence in Technical Specification 3.1.6.4, Technical Specification 4.17.6.c and applicable portion of Table 4.17-1 which provide reporting requirements for primary to secondary leaks by referencing Technical Specification 6.6.2.1 requirements are administrative changes. The current version of Technical Specification 6.6.2 does not require the reporting of a degraded mode permitted by a limiting condition for operation, however, it references 10 CFR 50.73 requirements for reportable events. The proposed title change in Specification 4.17.6.a is also an administrative change. As such, plant modifications or operating procedures revisions are not required as a result of these administrative changes. Accordingly, no new or different kind of accident can occur.

As mentioned before, Duke's philosophy and practice via administrative procedures for management of the primary to secondary leakage provides adequate assurance that an affected unit is removed form service before the technical specification limit is reached. This assures safe and economical plant operation and compliance with the requirements of 10 CFR 100. Therefore, it is not expected that the proposed changes would create the possibility of a new or different kind of accident from any kind of accident previously analyzed.

Third Standard

(Amendment would not) involve a significant reduction in a margin of safety.

The proposed Technical Specification 3.1.6.4 establishes a new specific technical specification limit of 1.0 gpm for total primary to secondary leakage for Oconee Units 2 and 3. The proposed limit of 1.0 gpm is in response to the NRC request and is based on recommended proposed technical specifications in their letter of October 2, 1987. The proposed technical specification revision is a change that constitutes an additional limitation and restriction not presently included in the Oconee Technical Specifications. This is clearly an improvement in the margin of safety.

The proposed leakage limit of 1.0 gpm also applies to the Oconee Unit 1 in that the present limit of 0.3 gpm for Unit 1 is increased to 1.0 gpm for consistency. The 0.3 gpm leakage limit was imposed on Unit 1 in October, 1977 as a result of the NRC concerns for "unique state of tube degradation" in Unit 1 OTSGs. As discussed in the technical justifications for the proposed amendment in Attachment 2, corrective actions such as sleeving, improved surveillance and operating procedures, enhanced leak testing techniques and data trending have allowed Duke to effectively managed leakage from tube degradation. The consequences of a 1.0 gpm total primary to secondary leakage for Unit 1 are bounded by the present FSAR analyses which ensure compliance with the requirements of 10 CFR Part 100. Furthermore, administrative procedures already in place would assure the reduction of power and removal of the affected plant from service for OTSG leak rates equal or less than the proposed 1.0 gpm limit. Therefore, the proposed 1.0 gpm total primary to secondary for Unit 1 does not involve a significant reduction in a margin of safety.

As discussed in the proceeding paragraphs and technical justification (Attachment 2) the deletion of the last sentence in the proposed Technical Specification 3.1.6.4 is an administrative change and does not involve a safety issue. This also applies to the deletion of Specification 4.17.6.c and the affected portion of Table 4.17-1, as well as the title change in Specification 4.17.6.a.

Duke has determined, based on the above discussion that there is a No Significant Hazards Consideration involved in this amendment request.