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October 24, 2006

U.S. Nuclear Regulatory Commission
Document Control Desk
Washington, D.C. 20555-0001

SUBJECT: Duke Power Company LLC d/b/a Duke Energy Carolinas, LLC

Oconee Nuclear Station, Units 1, 2, and 3
Docket Nos. 50-269, 50-270, and 50-287
Application for Technical Specification Improvement Regarding
Steam Generator Tube Integrity and Other Administrative Changes - Response to NRC Request for Additional Information

In a letter to the NRC dated April 11, 2006, (ML061080500), Duke Power Company LLC d/b/a Duke Energy Carolinas, LLC (Duke), submitted a license amendment request (LAR) to revise the steam generator tube integrity requirements in the Technical Specifications (TS) for Oconee Units 1, 2, and 3. The changes are consistent with NRC-approved Revision 4 to Technical Specification Task Force (TSTF) Standard Technical Specification Change Traveler, TSTF-449, "Steam Generator Tube Integrity." The availability of this TS improvement was announced in the Federal Register on May 6, 2005, (70FR24126), as part of the Consolidated Line Item Improvement Process (CLIIP). This letter provides Duke's answers to questions asked by the NRC staff on this LAR. Attachment 1 contains a statement of each NRC question followed by the Duke response. Attachment 2 provides revised TS and Bases page mark-ups for Oconee Units 1, 2, and 3. As discussed in Attachment 1, these revised pages replace the corresponding pages contained in the original April 11, 2006 LAR.

In accordance with 10 CFR 50.91, a copy of this application, with enclosures, is being provided to the designated official of the State of South Carolina.

If you should have any questions regarding this submittal, contact J. S. Warren at 704-875-5171.

Very truly yours,

Bruce Hamilton

B. H. Hamilton

Attachments

A001

Attachments:

1. Response to NRC Questions

2. Revised Technical Specifications and Bases Changes (Marked Pages) for Oconee Units 1, 2, and 3

xc (with attachments):

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AFFIDAVIT

B. H. Hamilton, being duly sworn, states that he is Vice President, Oconee Nuclear Site, Duke Power Company LLC d/b/a Duke Energy Carolinas, LLC, that he is authorized on the part of said Company to sign and file with the U. S. Nuclear Regulatory Commission this revision to the Renewed Facility Operating License Nos. DPR-38, DPR-47, and DPR-55; and that all statements and matters set forth herein are true and correct to the best of his knowledge.

Bruce Hamilton

B. H. Hamilton, Vice President
Oconee Nuclear Site

Subscribed and sworn to before me this 24 day of October, 2006

Sheila A Smith

Notary Public

My commission expires: 6/12/2013
Date

SEAL

bxc w/Attachments:

B. G. Davenport – ON03RC
R. V. Gambrell – ON03RC
J. E. Smith - ON03RC
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M. R. Hatley – MG05SE
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T. A. Saville
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NRIA File/ELL – EC08O
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Attachment 1

Response to NRC Questions

By letter dated April 11, 2006, (ML061080500), Duke Energy, submitted a request to revise the steam generator tube integrity requirements in the technical specifications for Oconee Units 1, 2, and 3. In order for the staff to complete its review, responses to the following questions are needed.

Question1

In proposed TS 5.5.10.a, you indicated that a condition monitoring assessment shall be conducted during an outage in which the steam generator tubes are "inspected and plugged". The intent of TSTF-449 was to require condition monitoring when the steam generator tubes are "inspected or plugged". Please discuss your plans to modify your submittal to make it consistent with TSTF-449.

Response

The submittal has been modified to be consistent with TSTF-449. A re-marked TS page (labeled INSERT 5.5.10) is included in Attachment 2 and the changes are indicated in the right margin.

Question 2

In proposed TS 5.5.10.b.2, you indicate, in part, that accident induced leakage is not to exceed 150 "gpd" per steam generator, "except for specific types of degradation at specific locations as described in paragraph c of the Steam Generator Program." Since "gpd" is not defined in this technical specification, please discuss your plans to replace "gpd" with "gallons per day". In addition, since the "exception" does not apply to Oconee, discuss your plans to remove the last part of the second sentence in TS 5.5.10.b.2.

Response

The submittal has been modified as requested. A re-marked TS page (labeled INSERT 5.5.10) is included in Attachment 2 and the changes are indicated in the right margin.

Question 3

Proposed TS 3.4.16 requires the plant to be in Mode 3 within 6 hours when steam generator tube integrity is not maintained. The steam generator program indicates, in part, that tube integrity is maintained when the operational leakage criterion in Limiting Condition for Operation (LCO) 3.4.13 is met. As a result, when the primary-to-secondary leak rate exceeds 150 gpd, tube integrity is not maintained and the plant must be in Mode 3 within 6 hours. However, TS 3.4.13 (as proposed) requires the plant to be in

Mode 3 within 12 hours when primary-to-secondary leakage exceeds 150 gallons per day (gpd). Please discuss your plans to address this apparent inconsistency between these two specifications (i.e., LCO 3.4.13 and 3.4.16).

Response

The requirement to be in Mode 3 within 12 hours for RCS Operational Leakage is included in current Oconee TS 3.4.13. No guidance was included in TSTF-449 to change this requirement. Therefore, when the submittal was made this requirement was not changed. This does create an apparent inconsistency with the proposed new TS 3.4.16. After evaluating the inconsistency between the existing TS (3.4.13) and the proposed TS (3.4.16), Duke concluded that it would be more appropriate to deviate from the TSTF. Generally, throughout the Oconee TS, the time allowed to place a unit in MODE 3 is 12 hours. Therefore, Duke has revised proposed new TS 3.4.16, Condition B, to require shut down to Mode 3 within 12 hours for consistency with the shutdown requirement of 12 hours in TS 3.4.13, Condition B, and similar shutdown requirements throughout the Oconee TS.

Oconee deviated from the Standard Technical Specifications (STS) by retaining the 12-hour Completion Time for Mode 3 when it converted to the Improved Technical Specifications (ITS). Excerpting from Oconee's ITS submittal (Reference 1):

> "The current licensing basis (CLB) generally permits 12 hours to place a unit in Hot Shutdown when an LCO is not met. To maintain consistency with current procedures, training and staffing requirements, the 12 hours, which are generally permitted to place a unit in Hot Shutdown, is retained in the ITS. CTS Hot Shutdown is comparable to ITS MODE 3. When applicable, the subsequent Completion Time to MODE 4 is modified accordingly (i.e., to allow an additional 6 hours)."

During its review, the NRC staff relied on 10 CFR 50.36 and the STS as guidance for acceptance of the plant-specific changes contained in Oconee's ITS submittal. The NRC's December 16, 1998 letter and safety evaluation (Reference 2) approving the Oconee ITS conversion provided a summary basis for the its conclusion that Duke can develop ITS for Oconee based on the STS, as modified by plant-specific changes. Part 3.0 of this SE further explains the NRC's conclusion that the conversion of the Oconee TS to those based on STS, as modified by plant-specific changes, is consistent with the Oconee current licensing basis and the requirements of 10 CFR 50.36.

Revised TS page 3.4.16-1 and Bases page B 3.4.16-5 are included in Attachment 2 and the changes are indicated in the right margin.

Question 4

In the Applicable Safety Analyses section on page B.3.4.13-1, you indicated that you would be deleting reference to the "loss of load" safety analyses. Please discuss the basis for removing this reference. In addition, in the same section of your current Bases it indicates that the steam line break analyses assumes leakage "greater" than 300 gallons per day as the initial condition. This appears inconsistent with Attachment 1 to your submittal which indicates that the accident leakage is calculated at the TS limit (which currently is 150 gallons per day per steam generator). Please clarify. The staff notes that if your accident analyses assumption is 300 gallons per day per steam generator, there are several others areas of your submittal that will need modification (e.g., last paragraph on page B 3.4.16-3)

Response

The original Oconee FSAR Chapter 14, Safety Analysis, included Section 14.1.2.8, Loss of Electric Power, with the two analysis cases being a loss of load condition, and a station blackout condition. Section 14.1.2.8.2, Results of Loss-of-Load Conditions Analysis, presented the transient analysis results along with an offsite dose evaluation.

In a letter to the NRC dated July 30, 1997 (Reference 3), Duke submitted Topical Report DPC-NE-3005-P, UFSAR Chapter 15 Transient Analysis Methodology," to the NRC for review and approval. This report describes the analysis methodologies to be used to revise UFSAR Chapter 15 (formerly Chapter 14) for the purpose of establishing a new licensing basis. As part of this report the loss of load event was replaced with the turbine trip event since the turbine trip is a more limiting event (see p. 1-2). The original NRC Safety Evaluation report for DPC-NE-3005-P (Reference 4) is dated October 1, 1998 (TAC Nos. M99349, M99350, and M99351). Page 8 of the SER states the method of analysis for the turbine trip event is acceptable. Subsequent to NRC approval Chapter 15 of the Oconee UFSAR was revised to delete the original loss of load analysis and replace it with the turbine trip analysis (Section 15.8). Therefore, the loss of load analysis is no longer part of the licensing basis, and any reference to it in the Technical Specification Bases is incorrect and should be revised.

The current Bases value of 300 gallon per day is for total leakage from both SGs. The proposed Bases is 150 gallons per day per SG, thus essentially the same. The 150 gallons per day value is the initial condition for the analysis. The Bases has been revised to indicate that the Steam Line Break analysis assumes total primary to secondary leakage of 150 gallons per day as shown in the revised markup of Bases page B 3.4.13-1 that is contained in Attachment 2.

Question 5

If your TS operational primary-to-secondary leakage limit is identical to what was assumed in your design bases accident analysis (see previous question), please address the following:

The NRC staff recognizes that plants have assumed that the leak rate during a design basis accident is the same as the leak rate during normal operation. However, it is important (required) to ensure that neither of these limits are exceeded. As a result, it may be necessary to ensure that the operational leak rate is kept well below the operational leak rate limit since the leak rate experienced during a design basis accident may be higher than that observed during normal operation. This increase in leak rate can be a result of either (1) the higher differential pressure associated with a design basis accident causing the leak rate from flaws leaking during normal operation to leak at higher rates or (2) the higher loadings associated with a design basis accident causing a flaw that was not leaking during normal operation to leak during the accident.

Given the above, discuss whether your procedures recognize this potential leakage issue or discuss your plans to modify your procedures to ensure that you will not exceed the accident induced leak rate limit as a result of the higher leak rates that may be observed during a design basis accident (as a result of inducing "new" leakage or as a result of the higher driving force for leakage). Alternatively, discuss your plans (and the technical basis) for modifying your normal operating and accident induced leakage limit to address these effects.

Response

Oconee procedures recognize the potential leakage issue discussed in the NRC question. To address this issue, the procedural limit for prompt shutdown is 100 gallons per day per SG. This will ensure that action is taken to maintain the leak rate well below the operational leakage limit.

Question 6

In the last paragraph on page B 3.4.16-3, you have the phrase "except for specific types of degradation at specific locations where the NRC has approved greater accident induced leakage." Since there are no alternate tube repair criteria approved for the Oconee units, please discuss your plans to remove this phrase.

Response

The submittal has been modified as requested. The phrase as described above has been removed. A revised Bases page B 3.4.16-3 is included in Attachment 2 and the change is indicated in the right margin.

References

1. Letter, W. R. McCollum, Jr., Duke Energy Corporation, to Document Control Desk (USNRC), SUBJECT: McGuire Nuclear Station, Generic Letter 88-20, Dated November 4, 1991.
2. Letter, David E. LaBarge, USNRC, to William R. McCollum, Jr., Duke Energy Corporation, SUBJECT: ISSUANCE OF AMENDMENTS – OCONEE NUCLEAR STATION, UNITS 1, 2, AND 3 (TAC NOS. M99912, 99913, AND 99914), Dated December 16, 1998.
3. Letter, M. S. Tuckman, Duke Energy Corporation, to Document Control Desk (USNRC), SUBJECT: Oconee Nuclear Station, Docket Numbers 50-269, -270, and -

287, UFSAR Chapter 15 Transient Analysis Methodology, DPC-NE-3005-P, Dated July 30, 1997.

4. Letter, David E. LaBarge, USNRC, to W. R. McCollum, Duke Energy Corporation, SUBJECT: REVIEW OF UPDATED FINAL SAFETY ANALYSIS REPORT, CHAPTER 15, TRANSIENT ANALYSIS METHODOLOGY SUBMITTAL – OCONEE NUCLEAR STATION, UNITS 1, 2, AND 3 (TAC NOS. M99349, M99350, AND M99351), Dated October 1, 1998.

Attachment 2

Oconee Nuclear Station Units 1, 2, and 3

Revised Technical Specifications and Bases Changes (Marked Pages)

OCONEE INSERTS

INSERT 5.5.10

A Steam Generator Program shall be established and implemented to ensure that SG tube integrity is maintained. In addition, the Steam Generator Program shall include the following provisions:

a. Provisions for condition monitoring assessments. Condition monitoring assessment means an evaluation of the "as found" condition of the tubing with respect to the performance criteria for structural integrity and accident induced leakage. The "as found" condition refers to the condition of the tubing during an SG inspection outage, as determined from the inservice inspection results or by other means, prior to the plugging of tubes. Condition monitoring assessments shall be conducted during each outage during which the SG tubes are inspected or plugged to confirm that the performance criteria are being met.

b. Performance criteria for SG tube integrity. SG tube integrity shall be maintained by meeting the performance criteria for tube structural integrity, accident induced leakage, and operational LEAKAGE.

   1. Structural integrity performance criterion: All in-service steam generator tubes shall retain structural integrity over the full range of normal operating conditions (including startup, operation in the power range, hot standby, and cool down and all anticipated transients included in the design specification) and design basis accidents. This includes retaining a safety factor of 3.0 against burst under normal steady state full power operation primary-to-secondary pressure differential and a safety factor of 1.4 against burst applied to the design basis accident primary-to-secondary pressure differentials. Apart from the above requirements, additional loading conditions associated with the design basis accidents, or combination of accidents in accordance with the design and licensing basis, shall also be evaluated to determine if the associated loads contribute significantly to burst or collapse. In the assessment of tube integrity, those loads that do significantly affect burst or collapse shall be determined and assessed in combination with the loads due to pressure with a safety factor of 1.2 on the combined primary loads and 1.0 on axial secondary loads.

   2. Accident induced leakage performance criterion: The primary to secondary accident induced leakage rate for any design basis accident, other than a SG tube rupture, shall not exceed the leakage rate assumed in the accident analysis in terms of total leakage rate for all SGs and leakage rate for an individual SG. Leakage is not to exceed 150 gallons per day per SG.

   3. The operational LEAKAGE performance criterion is specified in LCO 3.4.13, "RCS Operational LEAKAGE."

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.16 Steam Generator (SG) Tube Integrity

LCO 3.4.16 SG tube integrity shall be maintained.

AND

All SG tubes satisfying the tube repair criteria shall be plugged in accordance with the Steam Generator Program.

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

ACTIONS

---NOTE---
Separate Condition entry is allowed for each SG tube.

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One or more SG tubes satisfying the tube repair criteria and not plugged in accordance with the Steam Generator Program.	A.1	Verify tube integrity of the affected tube(s) is maintained until the next refueling outage or SG tube inspection.	7 days
	AND		
	A.2	Plug the affected tube(s) in accordance with the Steam Generator Program.	Prior to entering MODE 4 following the next refueling outage or SG tube inspection
B. Required Action and associated Completion Time of Condition A not met. OR SG tube integrity not maintained.	B.1	Be in MODE 3.	12 hours
	AND		
	B.2	Be in MODE 5.	36 hours

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.13 RCS Operational LEAKAGE

BASES

BACKGROUND

Components that contain or transport the coolant to or from the reactor core make up the RCS. Component joints are made by welding, bolting, rolling, or pressure loading, and valves isolate connecting systems from the RCS.

During unit life, the joint and valve interfaces can produce varying amounts of reactor coolant LEAKAGE, through either normal operational wear or mechanical deterioration. The purpose of the RCS Operational LEAKAGE LCO is to limit system operation in the presence of LEAKAGE from these sources to amounts that do not compromise safety. This LCO specifies the types and amounts of LEAKAGE.

The safety significance of RCS LEAKAGE varies widely depending on its source, rate, and duration. Therefore, detecting and monitoring reactor coolant LEAKAGE into the containment area ~~are~~ is necessary. Separating the identified LEAKAGE from the unidentified LEAKAGE is necessary to provide quantitative information to the operators, allowing them to take corrective action should a leak occur detrimental to the safety of the facility and the public.

This LCO deals with protection of the reactor coolant pressure boundary (RCPB) from degradation and the core from inadequate cooling, in addition to preventing the accident analysis radiation release assumptions from being exceeded. The consequences of violating this LCO include the possibility of a loss of coolant accident (LOCA). However, the ability to monitor leakage provides advance warning to permit unit shutdown before a LOCA occurs. This advantage has been shown by "leak before break" studies.

APPLICABLE SAFETY ANALYSES

Except for primary to secondary LEAKAGE, the safety analyses do not address operational LEAKAGE. However, other operational LEAKAGE is related to the safety analyses for LOCA; the amount of leakage can affect the probability of such an event. The steam line break (SLB) ~~and Loss of Load Safety~~ ~~analyses~~ analysis ~~assume~~ assumes ~~total~~ primary to secondary LEAKAGE ~~greater than 300~~ of 150 gallons per day per SG as the initial condition.

BASES

LCO (continued)

There are three SG performance criteria: structural integrity, accident induced leakage, and operational LEAKAGE. Failure to meet any one of these criteria is considered failure to meet the LCO.

The structural integrity performance criterion provides a margin of safety against tube burst or collapse under normal and accident conditions, and ensures structural integrity of the SG tubes under all anticipated transients included in the design specification. Tube burst is defined as, "The gross structural failure of the tube wall. The condition typically corresponds to an unstable opening displacement (e.g., opening area increased in response to constant pressure) accompanied by ductile (plastic) tearing of the tube material at the ends of the degradation." Tube collapse is defined as, "For the load displacement curve for a given structure, collapse occurs at the top of the load versus displacement curve where the slope of the curve becomes zero." The structural integrity performance criterion provides guidance on assessing loads that have a significant effect on burst or collapse. In that context, the term "significant" is defined as "An accident loading condition other than differential pressure is considered significant when the addition of such loads in the assessment of the structural integrity performance criterion could cause a lower structural limit or limiting burst/collapse condition to be established." For tube integrity evaluations, except for circumferential degradation, axial thermal loads are classified as secondary loads. For circumferential degradation, the classification of axial thermal loads as primary or secondary loads will be evaluated on a case-by-case basis. The division between primary and secondary classifications will be based on detailed analysis and/or testing.

Structural integrity requires that the primary membrane stress intensity in a tube not exceed the yield strength for all ASME Code, Section III, Service Level A (normal operating conditions) and Service Level B (upset or abnormal conditions) transients included in the design specification. This includes safety factors and applicable design basis loads based on ASME Code, Section III, Subsection NB (Ref. 4) and Draft Regulatory Guide 1.121 (Ref. 5).

The accident induced leakage performance criterion ensures that the primary to secondary LEAKAGE caused by a design basis accident, other than a SGTR, is within the accident analysis assumptions. The accident analysis assumes that accident induced leakage does not exceed 150 gallons per day per SG. The accident induced leakage rate includes any primary to secondary LEAKAGE existing prior to the accident in addition to primary to secondary LEAKAGE induced during the accident.

BASES

Actions (continued)

A Completion Time of 7 days is sufficient to complete the evaluation while minimizing the risk of plant operation with a SG tube that may not have tube integrity.

If the evaluation determines that the affected tube(s) have tube integrity, Required Action A.2 allows plant operation to continue until the next refueling outage or SG inspection provided the inspection interval continues to be supported by an operational assessment that reflects the affected tubes. However, the affected tube(s) must be plugged prior to entering MODE 4 following the next refueling outage or SG inspection. This Completion Time is acceptable since operation until the next inspection is supported by the operational assessment.

B.1 and B.2

If the Required Actions and associated Completion Times of Condition A are not met or if SG tube integrity is not being maintained, the reactor must be brought to MODE 3 within 12 hours and MODE 5 within 36 hours.

The allowed Completion Times are reasonable, based on operating experience, to reach the desired plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.4.16.1

During shutdown periods the SGs are inspected as required by this SR and the Steam Generator Program. NEI 97-06, Steam Generator Program Guidelines (Ref. 1), and its referenced EPRI Guidelines, establish the content of the Steam Generator Program. Use of the Steam Generator Program ensures that the inspection is appropriate and consistent with accepted industry practices.

During SG inspections a condition monitoring assessment of the SG tubes is performed. The condition monitoring assessment determines the "as found" condition of the SG tubes. The purpose of the condition monitoring assessment is to ensure that the SG performance criteria have been met for the previous operating period.