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A Duke Energy Company

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

SUBJECT: Duke Energy Corporation
Docket Nos. 50-269, -270, -287
Oconee Nuclear Station Units 1, 2, and 3
Supplemental Information - Proposed Amendment to the Facility Operating License Regarding Methodology for Determining Steam Generator Tube Loads Following a Main Steam Line Break and Runout Protection for the Turbine-Driven Emergency Feedwater Pump (TSC-99-01)

REFERENCE: Letter, Framatome Technologies to USNRC, Report of Preliminary Safety Concern Related to Design Steam Generator Tube Tensile Loads (PSC 2-98), October 19, 1998

On April 26, 1999, Duke Energy Corporation (Duke) submitted a License Amendment Request (LAR) for Facility Operating License Nos. DPR-38, DPR-47, and DPR-55 for Oconee Nuclear Station Units 1, 2, and 3, respectively, pursuant to 10 CFR 50.90. The LAR provided a method for obtaining a Nuclear Regulatory Commission (NRC) review of: 1) the analytical details regarding a revised methodology for determining steam generator (SG) tube loads following a main steam line break (MSLB); and 2) the crediting of the MSLB detection and feedwater isolation instrumentation as a means for providing runout protection for the turbine-driven emergency feedwater (EFW) pump. As a result of subsequent enhanced analyses and evaluations to resolve a Preliminary Safety Concern documented in the referenced letter, it was established that the SG tube loads were greater than those Duke provided to the NRC in the LAR.

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The purpose of this letter is to supplement the LAR by providing the revised tube loads and supporting information by means of replacement pages to the April 26, 1999 LAR. Additionally, operator actions following a MSLB are updated to reflect recent emergency operating procedure changes.

Enclosure 1 to this letter provides a replacement UFSAR Section 5.2.3.4 page 5-25 and its associated insert found in LAR Attachment 3. Enclosure 2 to this letter provides replacement pages to the original LAR description and technical justification of the proposed changes (LAR Attachment 4). The enclosure cover sheets provide page change instructions.

This revision to the April 26, 1999, LAR has been reviewed and approved by the Oconee Plant Operations Committee.

This revision does not affect the No Significant Hazards Consideration Evaluation and Environmental Assessment/Impact Statement for the LAR.

Pursuant to 10CFR50.91, a copy of this revision is being provided to the State of South Carolina.

Questions concerning this submittal should be directed to Robert Douglas at (864) 885-3073.

Very truly yours,

W. R. McCollum, Jr., Site Vice President
Oconee Nuclear Site

Enclosures

xc w/attachments:

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AFFIDAVIT

W. R. McCollum, Jr., being duly sworn, states that he is Site Vice President of Duke Energy Corporation; that he is authorized on the part of said corporation to sign and file with the Nuclear Regulatory Commission this revision to the Oconee Nuclear Station License Nos. DPR-38, DPR-47, and DPR-55; and that all statements and matters set forth therein are true and correct to the best of his knowledge.

W. R. McCollum, Jr., Site Vice President

Subscribed and sworn to me: 5/15/00
Date

Notary Public: Connie M Breazeale

My Commission Expires: 2/12/2003
Date

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U. S. Nuclear Regulatory Commission
May 15, 2000

ENCLOSURE 1

Replacement Markup Pages of the UFSAR
For Attachment 3 to April 26, 1999
License Amendment Application

Replace marked-up UFSAR Page 5-25 and following Insert to UFSAR Section 5.2.3.4

Note: The replacement UFSAR page 5-25 is provided solely due to repagination that occurred with the UFSAR update of December 31, 1998. The associated insert has been revised.

SECTION 5.2.3.4, p 5-25

the mean temperature difference between the tubes and the shell. During normal operation of the steam generator, the tube mean temperature should not be more than 32°F higher than the shell mean temperature. The maximum calculated mean tube to shell ΔT at normal operating conditions poses no problems to the structural integrity of the reactor coolant boundary. The effect of loss of reactor coolant would impose tensile stresses on the tubes and cause slight yielding across the tubes. Such a condition would introduce a small permanent deformation in the tubes but would in no way violate the boundary integrity. ~~The rupture of a secondary pipe would cause the tubes to become warmer than the shell and may cause tube deformation. Blowdown tests simulating secondary side blowdown on a 37-tube model boiler, show that although a slight buckling in the tubes occurred, there was no loss of reactor coolant.~~ INSERT

Calculations confirm that the steam generator tube sheet will withstand the loading resulting from a loss-of-coolant accident. The basis for this analysis is a hypothetical rupture of a reactor coolant pipe resulting in a maximum design pressure differential from the secondary side of 1050 psi. Under these conditions there is no rupture of the primary to secondary boundary (tubes and tube sheet).

The maximum primary membrane plus primary bending stress in the tube sheet under these conditions is 15,900 psi across the center ligaments which is well below the ASME Section III allowable limit of 40,000 psi at 650°F. Under the condition postulated, the stresses in the primary head show only the effect of its role as a structural restraint on the tube sheet. The stress intensity at the juncture of the spherical head with the tube sheet is 14,970 psi which is well below the allowable stress limit. It can therefore be concluded that no damage will occur to the tube sheet or the primary head as a result of this postulated accident.

In regard to tube integrity under loss of reactor coolant, actual pressure tests of 5/8 in. o.d./0.034 inch wall Inconel Tubing show collapse under an external pressure of 4,950 psig. This is a factor of safety of 4.7 against collapse under the 1,050 psig accidental application of external pressure to the tubes.

The rupture of a secondary pipe has been assumed to impose a maximum design pressure differential of 2,500 psi across the tubes and tube sheet from the primary side. The criterion for this accident permits no violation of the reactor coolant boundary (primary head, tube sheet, and tubes).

To meet this criterion, the stress limits delineated in the ASME Pressure Vessel Code, Section III, Paragraph N-714.2 for hydrotest limitations are applicable for the aforementioned abnormal operating circumstance. The referenced section states that the primary membrane stresses in the tube sheet ligaments, averaged across the ligament and through the tube sheet thickness, do not exceed 90 percent of the material yield stress at the operating temperature; in addition, the primary membrane plus primary bending stress in the tube sheet ligaments, averaged across the ligament width at the tube sheet surface location giving a maximum stress, does not exceed 135 percent of the material yield stress at the operating temperature.

An examination of stresses under these conditions show that for the case of a 2,500 psi design pressure differential, the stresses are within acceptable limits. These stresses together with the corresponding stress limits are given in Table 5-8.

The basic design criterion for the tubes assumes a pressure differential of 2,500 psi in accordance with Section III. Therefore, the secondary pressure loss accident condition imposes no extraordinary stress on the tubes beyond that normally expected and considered in Section III requirements.

The superimposed effect of secondary side pressure loss and maximum hypothetical earthquake has been considered. For this condition, the criterion is that there be no violation of the primary to secondary boundary (tube and tube sheet). For the case of the tube sheet, the maximum hypothetical earthquake

UFSAR Section 5.2.3.4 Insert

The rupture of a main steam line would result in an overcooling transient in which the steam generator tubes cool down faster than the steam generator shell. The tubes are then subjected to a tensile load that may cause tube deformation. An analysis of the MSLB accident is performed to determine the input for the steam generator tube stress analysis. The MSLB accident is analyzed with the RETRAN-02 code (Reference 27). The maximum break size of 6.305 ft^2 is analyzed from a full power initial condition to maximize the cooldown rate and the resulting stresses on the steam generator tubes. Main feedwater is isolated on low steam line pressure by the MSLB detection and feedwater isolation instrumentation. This circuit also inhibits the auto-start of or auto-stops the turbine-driven emergency feedwater pump. The motor-driven emergency feedwater pumps supply both steam generators until flow to the affected steam generator is manually isolated at 10 minutes. Additionally, the reactor coolant pumps are assumed to be manually tripped by the operator two minutes after the loss of subcooled margin occurs. The results of the RETRAN analysis, including the primary and secondary system pressures and the tube-to-shell temperature difference were used as input for the steam generator structural analysis. This analysis determined a tube axial load of 2870 lbf for the MSLB. The applicable tube stress acceptance criteria are based on the ASME code and industry practice. Specifically, the steam generator tubes shall retain a margin of safety against burst or gross failure of three times normal operating differential pressure, or 1.43 times the limiting accident differential pressure. In addition, ASME Section III has established a limit of the lesser of 2.4 x S_m, or 0.7 x S_u for design loads. The steam generator tubes have been evaluated for the 2870 lbf MSLB accident load and have been shown to meet these acceptance criteria.

U. S. Nuclear Regulatory Commission
May 15, 2000

ENCLOSURE 2

Replacement Pages
For Attachment 4 to April 26, 1999
License Amendment Request

Replacement Instructions and Information:

Replace pages 10 through 17 dated April 26, 1999, with pages 10 through 18 dated May 15, 2000.

Replacement pages 10 through 12, and pages 16 and 17 contain revisions. All other replacement pages are provided due to repagination.

All changes to the above pages are indicated with a change bar in the right margin.

> cause tube deformation. Analyses have shown that the resulting stresses would not result in tube rupture (Refer to Section 5.2.3.4)."

Section 5.2.3.4

Currently, UFSAR Section 5.2.3.4 states:

> "The rupture of a secondary pipe would cause the tubes to become warmer than the shell and may cause tube deformation. Blowdown tests simulating secondary side blowdown on a 37-tube model boiler, show that although a slight buckling in the tubes occurred, there was no loss of reactor coolant."

This LAR proposes to modify UFSAR Section 5.2.3.4 by replacing the information stated above with the following information:

> "The rupture of a main steam line would result in an overcooling transient in which the steam generator tubes cool down faster than the steam generator shell. The tubes are then subjected to a tensile load that may cause tube deformation. An analysis of the MSLB accident is performed to determine the input for the steam generator tube stress analysis. The MSLB accident is analyzed with the RETRAN-02 code (Reference 27). The maximum break size of 6.305 ft^2 is analyzed from a full power initial condition to maximize the cooldown rate and the resulting stresses on the steam generator tubes. Main feedwater is isolated on low steam line pressure by the MSLB detection and feedwater isolation instrumentation. This circuit also inhibits the auto-start of or auto-stops the turbine-driven emergency feedwater pump. The motor-driven emergency feedwater pumps supply both steam generators until flow to the affected steam generator is manually isolated at 10 minutes. Additionally, the reactor coolant pumps are assumed to be manually tripped by the operator two minutes after the loss of subcooled margin occurs. The results of the RETRAN analysis, including the primary and secondary system pressures and the tube-to-shell temperature difference were used as input for the steam

generator structural analysis. This analysis determined a tube axial load of 2870 lbf for the MSLB. The applicable tube stress acceptance criteria are based on the ASME code and industry practice. Specifically, the steam generator tubes shall retain a margin of safety against burst or gross failure of three times normal operating differential pressure, or 1.43 times the limiting accident differential pressure. In addition, ASME Section III has established a limit of the lesser of 2.4 x S_m, or 0.7 x S_u for design loads. The steam generator tubes have been evaluated for the 2870 lbf MSLB accident load and have been shown to meet these acceptance criteria."

Section 5.2.4

This LAR proposes to modify UFSAR Section 5.2.4 by adding a reference to RETRAN-02 - A Program for Transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems, EPRI NP-1850-CCM, Revision 4, EPRI, November 1988.

Chapter 15 Table of Contents

This LAR proposes to modify the Chapter 15 Table of Contents by deleting the reference to Section 15.13.4 and renumbering the references to Sections 15.13.5 and 15.13.6.

Section 15.13.4

This LAR proposes to delete UFSAR Section 15.13.4. The revised MSLB steam generator tube analysis will be provided in UFSAR Section 5.2.3.4

Section 15.13.5

This LAR proposes to modify UFSAR Section 15.13.5 by:

a) renumbering the Section as UFSAR Section 15.13.4; and

b) adding a reference to UFSAR Section 5.2.3.4.

Section 15.13.6

This LAR proposes to renumber UFSAR Section 15.13.6 as UFSAR Section 15.13.5.

Justification

The revised MSLB thermal-hydraulic (T-H) analysis assumes a full power initial condition and a double-ended rupture of the largest main steam line. The analysis assumptions were selected to maximize the tube-to-shell temperature difference, which results in conservative steam generator tube loads. The steam generator MSLB T-H time history data from the T-H analysis[1] were used as input to the structural analysis and adjusted for features not specifically modeled. The structural analysis was performed with an ANSYS finite element model of the steam generator to calculate the axial tube loads and tube/tube sheet dilations. The steam generator finite element model is an axisymmetric model of the entire steam generator. The structural analysis resulted in a maximum tube-to-shell temperature difference of 256°F and a tube loading of 2870 lbf. Although these results exceed the loads described in the current UFSAR Section 15.13.4, the revised loads meet the applicable acceptance criteria as described in the revised UFSAR Section 5.2.3.4 provided in Attachment 3.

The key assumptions of the current MSLB steam generator tube analysis are: 1) the plant is operating at full power; 2) the ICS is utilized to control steam generator levels to the post-trip minimum level; 3) the MFW system remains in operation; thus, the EFW system is not actuated; and 4) no operator actions are required to mitigate the event.

The revised MSLB analysis utilizes a different methodology. The key assumptions in this analysis are:

1. The plant is operating at full power with the ICS in manual.

[1] RETRAN-02, A Program for Transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems, EPRI NP-1850-CCM, Revision 4, EPRI, November 1988.

2. The MSLB detection and feedwater isolation instrumentation trips both MFW pumps, isolates the flow of MFW to both steam generators, and inhibits auto-start of or auto-stops the turbine-driven EFW pump.

3. The motor-driven EFW pumps start and supply both steam generators.

4. The EFW control valve on the affected steam generator fails open.

5. Operator action is taken to trip the reactor coolant pumps two minutes after a loss of subcooled margin.

6. Operator action is taken to isolate EFW flow to the affected steam generator ten minutes after event initiation.

The emergency feedwater is directed onto the peripheral steam generator tubes near the top of the steam generator, where it boils on contact with the tubes and support plates as it flows downward and inward. The resulting steam flows upward and out the break, and interacts with the downward flowing liquid. This local three-dimensional thermal-hydraulic process is not specifically modeled in the analysis, which models the average steam generator tube. The steam and feedwater interaction is assumed to be sufficient to justify the approximate average tube modeling approach. A conservative analysis result is assured given the large margin in the operator response time credited for isolating emergency feedwater. Main and emergency feedwater isolation are the dominant factors in the analysis.

<u>MSLB Detection and Feedwater Isolation Instrumentation</u>

Currently, Duke credits the MSLB detection and feedwater isolation instrumentation in the MSLB containment response analysis. The credited functions of the MSLB detection and feedwater isolation instrumentation are: 1) tripping both MFW pumps; 2) isolating MFW flow to both steam generators; and 3) inhibiting auto-start of the turbine-driven EFW pump or auto-stopping the turbine-driven EFW pump if it is running. The use of this circuitry for this purpose was approved by the NRC in a Safety Evaluation dated December 7, 1998, issued in support of Amendments 235, 234, and 233 to Facility Operating Licenses DPR-38, -47, and -55, for Oconee Nuclear Station Units 1, 2, and 3. This License Amendment incorporated Technical Specification requirements regarding the MSLB detection and feedwater isolation instrumentation. The requirements regarding the MSLB detection and feedwater isolation instrumentation are provided in ITS 3.3.11, 3.3.12, and 3.3.13.

The MSLB detection and feedwater isolation instrumentation is qualified as QA-1, whereas the ICS is non-safety. The circuitry is divided into two parts consisting of the MSLB detection circuitry and the feedwater isolation circuitry. The MSLB detection circuitry consists of three MSLB detection analog channels per main steam header (total of six). The feedwater isolation circuitry is divided into two redundant digital channels, with each digital channel comprised of two parallel 2 out of 3 logic combinations. The three analog detection channels on each main steam header provide input to the two parallel 2 out of 3 logic combinations in each digital channel. Actuation of either logic combination in a digital channel will actuate that digital channel. Feedwater isolation will occur if either digital channel is actuated. Thus, low steam generator pressure in either steam generator fully actuates the system.

A Loss of Offsite Power (LOOP) does not impact the ability of the MSLB detection and feedwater isolation circuitry to perform its intended safety function. Specifically, if a LOOP occurs during a MSLB event, a sufficient inventory of air will continue to be available, via the Instrument Air System, to drive the main and startup feedwater control valves closed. The feedwater control valves are supplied by two redundant subsystems of the Oconee Instrument Air System. These two systems are the Instrument Air System and the Service Air System. In the event that all of the air sources (compressors) to these two air systems are lost due to a LOOP, a sufficient air inventory exists in the reservoirs of these two systems to provide air to close the main feedwater control valves within 25 seconds of the initiation of the MSLB detection and feedwater isolation instrumentation.

Some of the equipment which is actuated by this circuitry is not fully QA-1 or single-failure proof. For example, the MFW control valve operators and the associated power supplies, as well as the startup MFW control valve operators and the associated power supplies, and the turbine-driven EFW pump steam admission valve are non QA-1, have no backup air supply path, and are subject to a single failure. In addition, the MFW pump trip circuitry and the MFW pumps are not QA-1 or single-failure proof. This information is summarized in a Duke submittal to the NRC dated June 14, 1995. While some of these components are not safety-related or single-failure proof, there is a high level of confidence that these components will perform their intended functions. The main feedwater control valves are under constant use above 20% full power. The main and startup feedwater control valves are in a preventive maintenance program. In addition, the main and startup feedwater control valves, the feedwater pump trip circuitry, the capability to inhibit auto-start of or auto-stop the turbine-driven EFW pump will be functionally tested in accordance with Chapter 16 of the UFSAR, "Selected Licensee Commitments."

Should a malfunction occur in the MSLB detection and feedwater isolation instrumentation, the non-safety ICS remains available to control steam generator water level. As demonstrated in the original analysis, the resulting steam generator tube stresses would decrease relative to the revised analysis.

<u>Operator Action to Trip Reactor Coolant Pumps</u>

Operator action is assumed in the revised MSLB steam generator tube load analysis at 2 minutes to trip the reactor coolant pumps on the loss of subcooled margin. This action is performed by one Control Room operator in the Control Room. Subcooling margins are monitored on the front control board. Regulatory Guide 1.97 qualified, QA-1 instrumentation is available for monitoring this parameter.

This operator action has been in place in the station Emergency Operating Procedure (currently, EP/1, 2, or 3/A/1800/001) for a number of years. Originally, it was added to address Item II.K.3.5 of the TMI Action Plan (i.e., NUREG-0737). The operator action to manually trip the reactor coolant pumps on the loss of subcooled margin was approved by the NRC in a Safety Evaluation issued to Oconee on March 15, 1988, and a Safety Evaluation issued to the B&W Owner's Group on May 29, 1986.

Operator Action to Isolate EFW Flow to Affected Steam Generator

The revised MSLB steam generator tube load analysis assumes that the two motor-driven EFW pumps start and provide flow to both steam generators. Additionally, the most limiting single failure is the EFW control valve to the affected steam generator, resulting in uncontrolled EFW flow. To mitigate this single failure, operator action is credited at 10 minutes to isolate EFW flow from the motor-driven EFW pumps to the affected steam generator. This action is performed in the Control Room by one Control Room Operator. The operator would determine the need for this action by monitoring steam generator level and pressure on the front control board. Regulatory Guide 1.97 qualified, QA-1 instrumentation is available for monitoring these parameters.

This operator action has been previously approved by the NRC for use in the MSLB containment response analysis in a Safety Evaluation dated December 7, 1998.

Training

Licensed Operators are extensively trained on emergency response actions. These actions include:

a) Tripping the reactor coolant pumps on a loss of subcooling margin. This action is expected to be completed within two minutes of a loss of subcooling margin.

b) Performing the following actions on identifying a MSLB:

- manually initiating the MSLB detection and feedwater isolation instrumentation;

- verifying tripped or tripping both MFW pumps;

- securing the motor-driven EFW pump on the affected steam generator;

- closing the EFW control valve(s) on the affected steam generator(s); and

- placing the switches for the startup and MFW blocks on both steam generators in the closed position.

The above actions are typically completed within three minutes of identifying a MSLB.

The above actions are taken per emergency operating procedures. Conditions for entry into the emergency operating procedures are committed to memory per an operations management procedure. The operators are very familiar with these procedures from frequent usage in requalification training.

These tasks are covered in the License Prep class including the bases for the task as well as how to perform it. Simulator training is also provided regarding these evolutions. Successful completion of the License Prep class ensures that all operators that would be expected to perform these tasks are adequately trained to do so. To maintain proficiency, these tasks are periodically covered in licensed requalification training. Some of the Active Simulator Exams used on the annual requalification exam also require these tasks to be completed. Only licensed operators would perform these evolutions in the Control Room and completing the requirements for licensed requalification ensures sufficient ability to do so correctly.

Conclusion

The revised MSLB steam generator tube load analysis results in a greater tube-to-shell temperature difference; thus, the steam generator tube load is increased. The revised MSLB steam generator tube load analysis concludes that steam generator tube integrity is maintained even with the increased steam generator tube load.