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50-270	Oconee Nuclear Station, Unit 2, Duke Power Co.	05000270
50-287	Oconee Nuclear Station, Unit 3, Duke Power Co.	05000287
50-369	William B. McGuire Nuclear Station, Unit 1, Duke Powe	05000369
50-370	William B. McGuire Nuclear Station, Unit 2, Duke Powe	05000370
50-413	Catawba Nuclear Station, Unit 1, Duke Power Co.	05000413
50-414	Catawba Nuclear Station, Unit 2, Duke Power Co.	05000414

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Document Control Branch (Document Control Desk)

SUBJECT: Forwards response to GL 97-05, "SG Tube Insp Techniques."

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February 24, 1998

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

ATTENTION: Document Control Desk

SUBJECT: Duke Energy Corporation

Oconee Nuclear Station - Units 1, 2, and 3
Docket Nos. 50-269, 50-270, and 50-287

McGuire Nuclear Station - Units 1 and 2
Docket Nos. 50-369 and 50-370

Catawba Nuclear Station - Units 1 and 2
Docket Nos. 50-413 and 50-414

Response to Generic Letter 97-05

NRC Generic Letter 97-05, dated December 17, 1997, required holders of operating licenses for pressurized-water reactors to submit information to the NRC regarding steam generator tubes sizing. The Duke Energy Corporation response to NRC Generic Letter 97-05 is provided in Attachment 1 to this letter.

As discussed in Attachment 1, the nuclear power industry recently voted to adopt an initiative requiring each utility to meet the intent of guidance provided in NEI 97-06, *Steam Generator Program Guidelines*, no later than the first refueling outage starting after January 1, 1999. Duke Energy Corporation will implement programs that meet NEI 97-06 at Oconee, McGuire, and Catawba Nuclear Stations.

Please direct questions on this matter to J. S. Warren at (704) 382-4986.

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I declare, under penalty of perjury, that the statements set forth herein are true and correct to the best of my knowledge.

Very truly yours,

M. S. Tuckman

M. S. Tuckman

MST/JSW

Attachment

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Attachment 1
Duke Energy Corporation
Response to NRC Generic Letter 97-05

For each Duke nuclear station, the response to the requested information in NRC Generic Letter 97-05 "Steam Generator Tube Inspection Techniques" is provided in the subsequent paragraphs.

McGuire Nuclear Station Units 1 and 2

Generic Letter 97-05 Required Information, Response Item 1:

Whether it is their practice to leave steam generator tubes with indications in service based on sizing,

Response to Item 1 for McGuire Units 1 and 2:

McGuire Nuclear Station Units 1 and 2 have Babcock and Wilcox Canada Model CFR-80 steam generators installed. These steam generators were installed in March 1997 and December 1997 respectively. It is currently not the practice at these units to leave steam generator tubes with indications in service based on sizing.

Generic Letter 97-05 Required Information, Response Item 2:

If the response to Item (1) is affirmative, those licensees should submit a written report that includes, for each type of indication, a description of the associated nondestructive examination method being used and the technical basis for the acceptability of the technique used.

Response to Item 2 for McGuire Units 1 and 2:

No response is required for McGuire since it is currently not the practice at either McGuire unit to leave steam generator tubes with indications in service based on sizing.

Attachment 1
Duke Energy Corporation
Response to NRC Generic Letter 97-05

Catawba Nuclear Station Units 1 and 2

Generic Letter 97-05 Required Information, Response Item 1:

Whether it is their practice to leave steam generator tubes with indications in service based on sizing,

Response to Item 1 for Catawba Unit 1:

Catawba Nuclear Station Unit 1 has Babcock and Wilcox Canada Model CFR-80 steam generators installed. These steam generators were installed in September 1996. It is currently not the practice at this unit to leave steam generator tubes with indications in service based on sizing.

Response to Item 1 for Catawba Unit 2:

Catawba Nuclear Station Unit 2 has Westinghouse Model D5 steam generators installed. It is the practice at Catawba Nuclear Station Unit 2 to leave steam generator tubes with wear in service based on sizing less than the Technical Specification value of 40%.

Generic Letter 97-05 Required Information, Response Item 2:

If the response to Item (1) is affirmative, those licensees should submit a written report that includes, for each type of indication, a description of the associated nondestructive examination method being used and the technical basis for the acceptability of the technique used.

Response to Item 2 for Catawba Unit 1:

No response is required for Catawba Unit 1 since it is currently not the practice at Catawba Unit 1 to leave steam generator tubes with indications in service based on sizing.

Response to Item 2 for Catawba Unit 2:

The nuclear power industry recently voted to adopt an initiative requiring each utility to meet the intent of the guidance provided in NEI 97-06, *Steam Generator Program Guidelines*, no later than the first refueling outage starting after January 1, 1999. As required by NEI 97-06, each utility is required to follow the inspection guidelines contained in the latest revision of the EPRI *PWR Steam Generator Examination Guidelines*.

Appendix H, "Performance Demonstration for Eddy Current Examination," of the EPRI *PWR Steam Generator Examination Guidelines*, Revision 5, provides guidance on the qualification of steam generator tubing examination techniques and equipment used to detect and size flaws. Damage mechanisms are divided into the following categories: thinning, pitting, wear, outside diameter intergranular attack and stress corrosion cracking, primary-side stress corrosion cracking, and impingement damage.

For qualification purposes, test samples are used to evaluate detection and sizing capabilities. While pulled tube samples are preferred, fabricated samples may be used. If fabricated test samples are used, the samples are verified to produce signals similar to those being observed in the field in terms of signal characteristics, signal amplitude, and signal-to-noise ratio. Samples are examined to determine the actual through wall defect measurements as part of the Appendix H qualification process. Flaw dimensions for samples used to construct grading units included in the qualification data set have been measured.

The procedures developed in accordance with Appendix H specify the essential variables for each procedure. These essential variables are associated with an individual instrument, probe, cable, or particular on-site equipment configurations.

For wear at the tube support plates, anti-vibration bars, vertical and diagonal straps, sizing is accomplished using the 400/130 kHz absolute mix signal of the bobbin probe. A calibration curve for amplitude vertical/maximum is determined based on the applicable standards replicating the damage mechanism type and quantity. This sizing technique is based on 64 sample data points. The samples ranged in depth from 4% to 78% through wall depth.

Oconee Nuclear Station Units 1, 2, and 3

Generic Letter 97-05 Required Information, Response Item 1:

Whether it is their practice to leave steam generator tubes with indications in service based on sizing,

Response to Item 1 for Oconee Units 1, 2, and 3:

Oconee Nuclear Station Units 1, 2, and 3 have Babcock and Wilcox Once Through Steam Generators(OTSG) installed. It is the practice at Oconee to leave steam generator tubes with indications of wear and impingement in service based on sizing less than the Technical Specification value of 40%.

Generic Letter 97-05 Required Information, Response Item 2:

If the response to Item (1) is affirmative, those licensees should submit a written report that includes, for each type of indication, a description of the associated nondestructive examination method being used and the technical basis for the acceptability of the technique used.

Response to Item 2 for Oconee Units 1, 2, and 3:

The nuclear power industry recently voted to adopt an initiative requiring each utility to meet the intent of the guidance provided in NEI 97-06, *Steam Generator Program Guidelines*, no later than the first refueling outage starting after January 1, 1999. As required by NEI 97-06, each utility is required to follow the inspection guidelines contained in the latest revision of the EPRI *PWR Steam Generator Examination Guidelines*.

Appendix H, "Performance Demonstration for Eddy Current Examination," of the EPRI *PWR Steam Generator Examination Guidelines,* Revision 5, provides guidance on the qualification of steam generator tubing examination techniques and equipment used to detect and size flaws.

Attachment 1
Duke Energy Corporation
Response to NRC Generic Letter 97-05

Damage mechanisms are divided into the following categories: thinning, pitting, wear, outside diameter intergranular attack and stress corrosion cracking, primary-side stress corrosion cracking, and impingement damage.

For qualification purposes, test samples are used to evaluate detection and sizing capabilities. While pulled tube samples are preferred, fabricated samples may be used. If fabricated test samples are used, the samples are verified to produce signals similar to those being observed in the field in terms of signal characteristics, signal amplitude, and signal-to-noise ratio. Samples are examined to determine the actual through wall defect measurements as part of the Appendix H qualification process. Flaw dimensions for samples used to construct grading units included in the qualification data have been measured.

The procedures developed in accordance with Appendix H specify the essential variables for each procedure. These essential variables are associated with an individual instrument, probe, cable, or particular on-site equipment configurations.

For flaws that are the result of impingement at non-dented tube support plates (TSP) locations in OTSG tubing, the 400/200 kHz differential mix off the bobbin probe is used to size the extent of the wall thinning. A calibration curve is established using the 20%, 60% and 100% holes of the ASME calibration standard. The size of the flaw is called off the phase angle using the maximum rate of change. The sizing procedure is based on the analysis of 30 sample data points. The samples ranged in depth from 16% to 98%.

For wear at broached TSP's, the 300 kHz, 200 kHz, and 100 kHz signals from the mid-range 0.115" pancake coil are used to size the depth of the wear flaw. A calibration curve is established for this technique using the 0%, 40%, and 60% depth wear scars under land contacts to provide more accurate depth measurements at shallow depths. The size of the flaw is called off the largest amplitude signal from the

300/100 kHz mix. Filters are not used. The sizing procedure is based on the analysis of 26 sample data points. The samples ranged in depth from 22% to 100%.