March 31, 2006

Mr. Dhiaa Jamil
Vice President
Catawba Nuclear Station
Duke Energy Corporation
4800 Concord Road
York, SC 29745

SUBJECT: CATAWBA NUCLEAR STATION, UNIT 2, ISSUANCE OF AMENDMENTS REGARDING THE STEAM GENERATOR PROGRAM (TAC NO. MC9430)

Dear Mr. Jamil:

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 224 to Renewed Facility Operating License NPF-52 for the Catawba Nuclear Station, Unit 2 (Catawba Unit 2). The amendments consist of changes to the Renewed Operating License and the Renewed Technical Specifications in response to your application dated December 19, 2005, as supplemented February 2, 2006, February 28, 2006, and March 30, 2006.

The amendment involves a one-time change to the Technical Specifications, regarding the required steam generator (SG) tube repair criteria for Catawba Unit 2, during refueling outage 14 and operating cycle 15. In addition, the proposed amendment adds a license condition requiring a reduction in the allowable normal operating primary-to-secondary leakage rate from 150 gallons-per-day to 75 gallons-per-day through any one SG and from 600 gallons-per-day to 300 gallons-per-day through all SGs. The proposed license condition will be applicable only for the duration of Catawba Unit 2 cycle 15 operation.

A copy of the related Safety Evaluation is also enclosed. A Notice of Issuance will be included in the Commission's biweekly Federal Register notice.

Sincerely,

/RA/
John Stang, Senior Project Manager
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-414

Enclosures:
1. Amendment No. 224 to NPF-52
2. Safety Evaluation

cc w/encls: See next page

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DUKE ENERGY CORPORATION

NORTH CAROLINA MUNICIPAL POWER AGENCY NO. 1

PIEDMONT MUNICIPAL POWER AGENCY

DOCKET NO. 50-414

CATAWBA NUCLEAR STATION, UNIT 2

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 224
Renewed License No. NPF-52

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment to the Catawba Nuclear Station, Unit 2 (the facility) Renewed Facility Operating License No. NPF-52 filed by the Duke Energy Corporation, acting for itself, North Carolina Municipal Power Agency No. 1 and Piedmont Municipal Power Agency (licensees), dated December 19, 2005, as supplemented February 2, 2006, February 28, 2006, and March 30, 2006, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations as set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (I) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations set forth in 10 CFR Chapter I;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is hereby amended by page changes to the Technical Specifications as indicated in the attachment to this license amendment, and Paragraph 2.C.(2) of Renewed Facility Operating License No. NPF-52 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 224, which are attached hereto, are hereby incorporated into this license. Duke Energy Corporation shall operate the facility in accordance with the Technical Specifications.

3. Further, Renewed Facility Operating License No. NPF-52 is hereby amended to add a license condition to Appendix B of the license to read as follows:

   This amendment requires the licensee to use administrative controls, as described in the licensee's letter of February 2, 2006, and evaluated in the Staff's Safety Evaluation dated March 31, 2006, to restrict the primary-to-secondary leakage through any one steam generator to 75 gallons-per-day and through all steam generators to 300 gallons-per-day (in lieu of the limits in TS Sections 3.4.13d. and 5.5.9b.3.), for Cycle 15 operation.

4. This license amendment is effective as of its date of issuance and shall be implemented within 30 days of issuance. The above license condition will be applicable only for the duration of Catawba Unit 2 Cycle 15 operation.

FOR THE NUCLEAR REGULATORY COMMISSION

Evangelos C. Marinos, Chief
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to Technical Specification
and Facility Operating License

Date of Issuance: March 31, 2006

ATTACHMENT TO LICENSE AMENDMENT NO. 224

RENEWED FACILITY OPERATING LICENSE NO. NPF-52

DOCKET NO. 50-414

Replace the following pages of the Renewed Facility Operating License with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Unit 2

Remove	Insert
Page 5	Page 5
Appendix B, Page 2	Appendix B, Page 2

Replace the following page of the Appendix A Technical Specifications with the attached revised page. The revised page is identified by amendment number and contains marginal lines indicating the areas of change.

Remove	Insert
-	5.5-7a

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 224 TO RENEWED FACILITY

OPERATING LICENSE NPF-52

DUKE ENERGY CORPORATION, ET AL.

CATAWBA NUCLEAR STATION, UNIT 2

DOCKET NO. 50-414

1.0 INTRODUCTION

By letter dated December 19, 2005, as supplemented February 2 and 28, 2006, Duke Energy Corporation, et al. (DEC, the licensee), submitted a request for changes to the Catawba Nuclear Station, Unit 2 (Catawba Unit 2), Renewed Operating License and the Renewed Technical Specifications (TSs). The February 28, 2006, and March 30, 2006, letters provided clarifying information and did not enlarge the scope of the original application.

The requested changes would change the TSs, regarding the required steam generator (SG) tube repair criteria for Catawba Unit 2, during refueling outage 14 and operating cycle 15. In addition, the proposed amendment adds a license condition to require a reduction in the allowable normal operating primary-to-secondary leakage rate from 150 gallons-per-day to 75 gallons-per-day through any one SG and from 600 gallons-per-day to 300 gallons-per-day through all SGs. The proposed license condition will be applicable only for the duration of Catawba Unit 2 cycle 15 operation.

Catawba Unit 2 has four Model D5 recirculating, pre-heater-type SGs designed and fabricated by Westinghouse. The thermally treated Alloy 600 SG U-tubes have an outside diameter of 0.75 inches and a nominal wall thickness of 0.043 inches. The tube support plates are 1.125 inches thick stainless steel and have quatrefoil broached holes. The tubes are hydraulically expanded for the full depth of the tubesheet.

The licensee has been using eddy current bobbin coil probes for inspecting the length of tubing within the tubesheet. However, the bobbin probe is not capable of reliably detecting stress corrosion cracks (SCC) in the tubesheet region should such cracks be present. For this reason, the licensee has been supplementing the bobbin probe inspections with rotating pancake coil probes in a region extending from 2 inches above the top of the tubesheet (TTS) to the end of the tube at the bottom of the tubesheet. This zone includes the tube expansion transition zone located at the TTS. The expansion transition contains significant residual stress and was considered a likely location for SCC should it ever develop. Until the fall of 2004, there

had not been any reported instances of SCC affecting the tubesheet region of thermally treated Alloy 600 tubing, either at Catawba Unit 2, or elsewhere in the U.S.
In the fall of 2004, crack-like indications were found in tubes in the tubesheet region of Catawba Unit 2. These crack-like indications were found in bulges (or over-expansions) in the tubesheet region, in the tack roll region, and in the tube-to-tubesheet weld. (The tack expansion is an initial 0.7-inch long expansion at each tube end and formed prior to the hydraulic expansion over the full tubesheet depth. Its purpose was to facilitate performing the tube to tubesheet weld.) Crack-like indications were found in a bulge in one tube and in the tack expansion in nine tubes. Approximately 6 of the 196 tube-to-tubesheet weld indications extended into the parent tube.

The licensee believes that any flaws located below 17 inches below the TTS (i.e., in the bottom 4 inches of the tubesheet region, including the tack expansion region and the tubing in the vicinity of the welds) have no potential to impair tube integrity, and, thus, do not pose a safety concern. To avoid the unnecessary plugging or repair of tubes, the licensee is proposing on a one-time basis to revise the TS such that tubes found to contain flaws in the lower 4 inches of the tubesheet region need not be plugged or repaired as required by the TS should inspection reveal cracks in this region.

1.1 TS 5.5.9.c

A new paragraph has been added to state:

The following alternate tube repair criteria may be applied as an alternative to the 40% depth based criteria:

> For the Unit 2 End of Cycle 14 Refueling Outage and Cycle 15 operation only, the 40% depth based criterion does not apply to degradation identified in the portion of the tube below 17 inches from the top of the tubesheet. If degradation is identified in the portion of the tube from the top of the tubesheet to 17 inches below the top of the tubesheet, the tube shall be removed from service. If degradation is found in the portion of the tube below 17 inches from the top of the tubesheet, the tube does not require plugging.

1.2 License Condition

As a conservative measure the licensee has proposed to add a condition to the license to limit normal operating primary-to-secondary identified leakage through one steam generator and the total leakage of all steam generators for the duration of operating cycle 15. The following proposed license condition will be added to Appendix B of the License:

> This amendment requires the licensee to use administrative controls, as described in the licensee's letter of February 2, 2006, and evaluated in the Staff's Safety Evaluation dated March 31, 2006, to restrict the primary to secondary leakage through any one steam generator to 75 gallons-per-day and through all steam generators to 300 gallons-per-day (in lieu of the limits in TS Sections 3.4.13d. and 5.5.9b.3.), for Cycle 15 operation.

2.0 REGULATORY EVALUATION

Steam Generator tubes function as an integral part of the reactor coolant pressure boundary (RCPB) and serve to isolate radiological fission products in the primary coolant from the secondary coolant and the environment. For the purposes of this safety evaluation, tube integrity means that the tubes are capable of performing these functions in accordance with the plant design and licensing basis.

Title 10 of the *Code of Federal Regulations* (10 CFR) establishes the fundamental regulatory requirements with respect to the integrity of the SG tubing. Specifically, the General Design Criteria (GDC) in Appendix A to 10 CFR Part 50 state that the RCPB shall have "an extremely low probability of abnormal leakage ... and gross rupture" (GDC 14), "shall be designed with sufficient margin" (GDC 15 and 31), shall be of "the highest quality standards practical" (GDC 30), and shall be designed to permit "periodic inspection and testing of important areas and features to assess their structural and leaktight integrity" (GDC 32). Section 50.55a(c)(1) specifies that components that are part of the RCPB must meet the requirements for Class 1 components in Section III of the American Society of Mechanical Engineers (ASME), *Boiler and Pressure Vessel Code* (Code). Section 50.55a(g)(3)(i) of 10 CFR requires that throughout the service life of a pressurized-water reactor (PWR) facility, ASME Code Class 1 components meet the requirements in Section XI, "Rules for Inservice Inspection [ISI] of Nuclear Power Plant Components," of the ASME Code, to the extent practical. This requirement includes the inspection and repair criteria of Section XI of the ASME Code. Section XI requirements pertaining to ISI of SG tubing are augmented by additional SG tube surveillance requirements in the TSs.

As part of the plant licensing basis, applicants for PWR licenses are required to analyze the consequences of postulated design-basis accidents (DBAs) such as a SG tube rupture (SGTR) and main steamline break (MSLB). These analyses consider the primary-to-secondary leakage through the tubing which may occur during these events and must show that the offsite radiological consequences do not exceed the applicable limits of the 10 CFR Part 100 guidelines for offsite doses, GDC 19 criteria for control room operator doses, or some fraction thereof as appropriate to the accident.

Under the plant TS SG program requirements, the licensee is required to monitor the condition of the SG tubing and to plug or repair tubes as necessary. Specifically, the licensee is required to perform periodic inspections of and to repair or remove from service by plugging all tubes found to contain flaws with sizes exceeding the acceptance limit, termed "plugging limit." The tube plugging limits were developed with the intent of ensuring that degraded tubes (1) maintain factors of safety against gross rupture consistent with the plant design basis (i.e., consistent with the stress limits of the ASME Code, Section III) and (2) maintain leakage integrity consistent with the plant licensing basis while allowing for potential flaw size measurement error and flaw growth between SG inspections. The requirements for SG tube plugging are specified in TS 5.5.9, "Steam Generator (SG) Program." The subject TS amendment request concerns the portions of the tubing that are subject to the TS SG program requirements related to plugging or repairs.

The proposed license amendment would limit plugging and repairs in the 21-inch thick tubesheet region to the upper 17 inches of the tubesheet region, and is conceptually similar to permanent amendments approved by the NRC staff for a number of plants. Examples include the F criteria approved for Westinghouse SGs where the tubes were hard roll expanded inside the tubesheet and the W criteria approved for plants where the tubes were explosively expanded against the tubesheet. In the case of the F criteria, the required inspection zone was limited to approximately the upper 1.5-inch zone below the TTS. The W criteria required an inspection zone extending approximately 8 inches below the TTS. The larger required inspection zone for W relative to F is that the explosively expanded joints do not exhibit as much residual interference fit as do hard rolled joints. The proposed license amendment for Catawba Unit 2 follows similar license amendment requests at other plants where the tubes are hydraulically expanded against the tubesheet (Braidwood, April 25, 2005, ML051110573 and Byron, September 19, 2005, ML052230016).

3.0 TECHNICAL EVALUATION

The tube-to-tubesheet joint consists of the tube, which is hydraulically expanded against the bore of the tubesheet, the tube-to-tubesheet weld located at the tube end, and the tubesheet. The joint was designed as a welded joint in accordance with ASME Code, Section III, and not as a friction or expansion joint. The weld itself was designed as a pressure boundary element in accordance with ASME Code, Section III. It was designed to transmit the entire end cap pressure load during normal and DBA conditions from the tube to the tubesheet with no credit taken for the friction developed between the hydraulically expanded tube and the tubesheet. In addition, the weld serves to make the joint leak-tight.

The licensee, in effect, is proposing on a one-time basis to exempt tubes with flaw indications in the lower 4-inch zone from the need to plug or repair. This proposal, in effect, redefines the pressure boundary at the tube-to-tubesheet joint as consisting of a friction or expansion joint with the tube assumed to be hydraulically expanded against the tubesheet over the top 17 inches of the tubesheet region. Under this proposal, no credit is taken for the lower 4 inches of the tube or the tube-to-tubesheet weld in contributing to the structural or leakage integrity of the joint. The lower 4 inches of the tube and weld are assumed not to exist.

The regulatory standard by which the NRC staff has evaluated the subject license amendment is that the amended TSs should continue to ensure that tube integrity will be maintained. This includes maintaining structural safety margins consistent with the plant design basis as embodied in the stress limit criteria of ASME Code, Section III as is discussed in Section 3.1 below. In addition, this includes limiting the potential for accident-induced primary-to-secondary leakage to values not exceeding those assumed in the licensing basis accident analyses. Maintaining tube integrity in this manner ensures that the amended TSs are in compliance with all applicable regulations. The NRC staff's evaluation of joint structural integrity and leakage integrity is discussed in Sections 3.1 and 3.2, respectively, of this safety evaluation.

The licensee is also proposing, on a one-time basis, to plug or repair on detection any flaw indication found in the upper 17-inch region of the tubesheet region of the tubes, irrespective of whether the flaw exceeds the TS 40-percent plugging limit (see proposed second new paragraph for TS 5.5.9c, "Plugging or Repair Limit"). The NRC staff finds this acceptable since it is more conservative than the current TS 40-percent plugging limit and will provide added assurance that the length of tubing along the entire proposed 17-inch inspection zone will be effective in resisting tube pull-out under tube end cap pressure loads and in resisting primary-to-secondary leakage between the tube and tubesheet.

3.1 Joint Structural Integrity

Westinghouse has conducted analyses and testing to establish the engagement (embedment) length of hydraulically expanded tubing inside the tubesheet that is necessary to resist pull-out under normal operating and DBA conditions. Pull-out is the structural failure mode of interest since the tubes are radially constrained against axial fishmouth rupture by the presence of the tubesheet. The axial force that could produce pull-out derives from the pressure end cap loads due to the primary-to-secondary pressure differentials associated with normal operating and DBA conditions. The licensee's contractor, Westinghouse, determined the required engagement distance on the basis of maintaining a factor of three against pull-out under normal operating conditions and a factor of 1.4 against pull-out under accident conditions. Pull-out was conservatively treated as tube slippage relative to the tubesheet of 0.25 inches. The Nuclear Regulatory Commission (NRC) staff concurs that these are the appropriate safety factors to apply to demonstrate structural integrity. As documented in detail in a safety evaluation accompanying the NRC staff's approval of new performance-based SG TSs for the Joseph M, Farley Nuclear Plant, Units 1 and 2 (Reference: Letter, Sean Peters, NRC, to L. M. Stinson, Vice President, Southern Nuclear Operating Company, "Joseph M. Farley Nuclear Plant, Units 1 and 2, re: Issuance of Amendments to Facilitate Implementation of Industry Initiative NEI 97-06, Steam Generator Program Guidelines," dated September 10, 2004, ADAMS Accession No. (ML042570427)), the NRC staff has concluded that these safety-factor criteria are consistent with the stress-limit criteria in ASME Code, Section III.

The resistance to pull-out is the axial friction force developed between the expanded tube and the tubesheet over the engagement distance. The friction force is a function of the radial contact pressure between the expanded tube and the tubesheet. The radial contact pressure derives from several contributors: (1) the contact pressure associated directly with the hydraulic expansion process itself, (2) additional contact pressure due to differential radial thermal expansion between the tube and tubesheet under hot operating conditions, (3) additional contact pressure caused by the primary pressure inside the tube, and (4) additional or reduced contact pressure associated with tubesheet bore dilation (distortion) caused by tubesheet bow (deflection) as a result of the primary-to-secondary pressure load acting on the tubesheet. Westinghouse employed a combination of pull-out tests and analyses, including finite element analyses, to evaluate these contributors. Based on these analyses and tests, Westinghouse concluded that the required engagement distances to ensure the safety factor criteria against pull-out are achieved vary from 3 to 8.6 inches depending on the radial location of the tube within the tube bundle, with the largest engagement distances needed toward the center of the bundle.

The NRC staff has not reviewed the Westinghouse analyses in detail, and, thus, has not reached a conclusion with respect to whether 3 to 8.6 inches of engagement (termed H criterion by Westinghouse) is adequate to ensure that the necessary safety margins against pull-out are maintained. The licensee, therefore, is proposing on a one-time basis to inspect the tubes in the tubesheet region such as to ensure a minimum of 17 inches of effective engagement, well in excess of the 3 to 8.6 inches that the Westinghouse analyses indicate are needed. Based on the following considerations, the NRC staff concludes the proposed 17 inch engagement length is clearly acceptable to ensure the structural integrity of the tubesheet joint.

- Pull-out tests demonstrate that the radial contact pressure produced by the hydraulic expansion alone is such as to require an engagement distance of 6 inches to ensure the appropriate safety margins against pull-out. This estimate is a mean minus one standard deviation estimate based on nine pull-out tests. This estimate ignores the effect on needed engagement distance from differential thermal expansion, internal primary pressure in the tube, and tubesheet bore dilations associated with tubesheet bow.

- Radial differential thermal expansion between the tube and tubesheet under hot operating and accident conditions will act to further tighten the joint (i.e., increase radial contact pressure) and to reduce the necessary engagement distance relative to room temperature conditions. The radial differential thermal expansion arises from the fact that the Alloy 600 tubing has a slightly higher (by 6 percent) coefficient of thermal expansion than does the SA-508 Class 2a tubesheet material and that the tubes are a little hotter than the tubesheet.

- The internal primary pressure inside the tube under normal operating and accident conditions also acts to tighten the joint relative to unpressurized conditions, thus reducing the necessary engagement distance.

- Tubesheet bore dilations caused by tubesheet bow under primary-to-secondary pressure can increase or decrease contact pressure depending on the tube location within the bundle and on the location along the length of the tube in the tubesheet region. Basically, the tubesheet acts as a flat, circular plate under an upward acting net pressure load. The tubesheet is supported axially around its periphery with a partial restraint against tubesheet rotation provided by the SG shell and channel head. The SG divider plate provides a spring support against upward displacement along a diametral mid-line. Over most of the tubesheet away from the periphery, the bending moment resulting from the applied primary-to-secondary pressure load can be expected to put the tubesheet into tension at the top and compression at the bottom. Thus, the resulting distortion of the tubesheet bore (tubesheet bore dilation) tends to be such as to loosen the tube to tubesheet joint at the top of the tubesheet and to tighten the joint at the bottom of the tubesheet. The amount of dilation and resulting change in joint contact pressure would be expected to vary in a linear fashion from top to bottom of the tubesheet. Given the neutral axis to be at approximately the axial mid-point of the tubesheet thickness (i.e., 10.5 inches below the TTS), tubesheet bore dilation effects would be expected to further tighten the joint from 10 inches below the TTS to 17 inches below the TTS, which would be the lower limit of the proposed tubesheet region inspection zone. Combined with the effects of the joint tightening associated with the radial differential thermal expansion and primary pressure inside the tube, contact pressure over at least a 6.5-inch distance should be considerably higher than the contact pressure simulated in the above mentioned pull-out tests. A similar logic applied to the periphery of the tubesheet leads the staff to conclude that at the top 10.5 inches of the tubesheet region, the contact pressure should be considerably higher than the contact pressure simulated in the above mentioned pull-out tests. Thus, the staff concludes that the proposed 17-inch engagement distance (or inspection zone) is acceptable to ensure the structural integrity of the tubesheet joint.

3.2 Joint Leakage Integrity

If no credit is to be taken for the presence of the tube-to-tubesheet weld, a potential leak path between the primary-to-secondary is introduced between the hydraulically expanded tubing and the tubesheet. In addition, leaving tubes in service with active degradation in the lower 4 inches of the tubesheet region may lead to an increased potential for 100 percent throughwall flaws in this zone and the potential for leakage of primary coolant through the crack and up between the hydraulically expanded tubes and tubesheet to the secondary system. Operational leakage integrity is assured by monitoring primary-to-secondary leakage relative to the applicable TS limiting condition for operation (LCO) limits. However, it must also be demonstrated that the proposed TS changes do not create the potential for leakage during DBAs that may exceed values assumed in the licensing basis accident analyses. The licensee states that this is ensured by limiting primary-to-secondary leakage to 150 gallons-per-day (gpd) in the faulted SG during an MSLB.

To support its H criterion (discussed above), Westinghouse has developed a detailed leakage prediction model, which considers the resistance to leakage from cracks located within the thickness of the tubesheet. The NRC staff has not reviewed or accepted this model. For the proposed one-time 17-inch inspection zone, Westinghouse cited a number of qualitative arguments supporting a conclusion that a minimum 17-inch engagement length ensures that leakage during an MSLB will not exceed two times the observed leakage during normal operation. Westinghouse refers to this as the "bellwether approach." Catawba Unit 2 is adopting an operational leakage limit of 75 gpd as a license condition associated with the TS change. Thus, for an SG leaking at the operational limit (i.e., 75 gpd) under normal operating conditions, Westinghouse estimates that leakage would not be expected to exceed the 150 gpd assumed in the licensing basis accident analyses for an MSLB.

The factor of 2 upper bound is based on the Darcy equation for flow through a porous media where leakage rate would be proportional to differential pressure. Westinghouse considered normal operating pressure differentials between 1200 and 1400 psi and accident differential pressures on the order of 2560 to 2650 psi, essentially a factor of 2 difference. The factor of 2 as an upper bound is based on a premise that the flow resistance between the tube and tubesheet remains unchanged. Westinghouse states that the flow resistance varies as a log normal linear function of joint contact pressure. The NRC staff concurs that the factor of 2 upper bound is reasonable, given the stated premise. The NRC staff notes that the assumed linear relationship between leak rate and differential pressure is conservative relative to alternative models such as Bernoulli or orifice models, which assume leak rate to be proportional to the square root of differential pressure.

The NRC staff reviewed the qualitative arguments developed by Westinghouse regarding the conservatism of the aforementioned premise, namely the conservatism of assuming that flow resistance between the expanded tubing and the tubesheet does not decrease under the most limiting accident relative to normal operating conditions. Most of the Westinghouse observations are based on insights derived from the finite element analyses performed to assess joint contact pressures and from test data relating leak flow resistance to joint contact pressure, neither of which has been reviewed by the NRC staff in detail. Among the Westinghouse observations is that for all tubes there is at least an 8-inch zone in the upper 17 inches of the tubesheet where there is an increase in joint contact pressure due to higher primary pressure inside the tube and changes in tubesheet bore dilation along the length of the tubes. In Section 3.1 above, the NRC staff observed that there is at least a 6.5-inch zone over which changes in tubesheet bore dilations, when going from unpressurized to pressurized conditions, should result in an increase in joint contact pressure. The contact pressure due to changes in tubesheet bore dilation should increase further over this 6.5-inch zone under the increased pressure loading on the tubesheet during accident conditions. Considering the higher pressure loading in the tube when going from normal operating to accident conditions, the Westinghouse estimate that contact pressures, and, thus, leak flow resistance, always increases over at least an 8-inch distance appears reasonable to the NRC staff.

Although joint contact pressures and leak flow resistance decrease over other portions of the tube length, Westinghouse expects a net increase in total leak flow resistance on the basis of its insights from leakage test data that demonstrates that leak flow resistance is more sensitive to changes in joint contact pressure as contact pressure increases due to the linear log normal nature of the relationship. The NRC staff's depth of review did not permit it to credit this aspect of the Westinghouse assessment. However, it is clear from the above discussion that there would be no significant reduction in leakage flow resistance when going from normal operating to accident conditions.

4.0 SUMMARY

The NRC staff has considered that undetected cracks in the lower 4 inches are unlikely to produce leakage rates during normal operation that would approach the LCO leakage limits during normal operation, thus providing additional confidence that such cracks will not result in leakage in excess of the values assumed in the accident analyses. Any axial cracks will be tightly clamped by the tubesheet against opening of the crack faces. In addition, little of the end cap pressure load should remain in the tube below 17 inches, and, thus, any circumferential cracks would be expected to remain tight. Thus, irrespective of the flow resistance in the upper 17 inches of the tubesheet between the tube and tubesheet, the tightness of the cracks themselves should limit leakage to very small values. Based on the above, the NRC staff concludes that there is reasonable assurance that the proposed one-time exclusion of the lower 4 inches of the tubes in the tubesheet region from the tube inspection and plugging and repair requirements will not impair the leakage integrity of the tube-to-tubesheet joint, ensures that the structural and leakage integrity of the tube-to-tubesheet joint will be maintained with structural safety margins consistent with the design basis with leakage integrity within assumptions employed in the licensing basis accident analyses, and, thus, is in accordance with the applicable regulations without undue risk to public health and safety. Therefore, the NRC staff concludes that the proposed amendment is acceptable.

5.0 FINAL NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

The Commission's regulations in 10 CFR 50.92(c) state that the Commission may make a final determination that a license amendment involves no significant hazards consideration if operation of the facility in accordance with the 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 previously evaluated; or,

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

The following analysis was provided by the licensee in its February 2, 2006, letter.

First Standard

Does operation of the facility in accordance with the proposed amendment involve a significant increase in the probability or consequences of an accident previously evaluated? No.

The previously analyzed accidents are initiated by the failure of plant structures, systems, or components. The proposed change that alters the SG tube repair criteria does not have a detrimental impact on the integrity of any plant structure, system, or component that initiates an analyzed event. The proposed change will not alter the operation of, or otherwise increase the failure probability of any plant equipment that initiates an analyzed accident.

Of the applicable accidents previously evaluated, the limiting transients with consideration to the proposed change to the SG tube repair criteria, are the SG tube rupture event and the steam line break event.

During the SG tube rupture event, the required structural integrity margins of the SG tubes will be maintained by the presence of the SG tubesheet. SG tubes are hydraulically expanded in the tubesheet area. Tube rupture in tubes with cracks in the tubesheet region of the tube is precluded by the constraint provided by the tubesheet. This constraint results from the hydraulic expansion process, thermal expansion mismatch between the tube and tubesheet, and the differential pressure between the primary and secondary side. Based on this design, the structural margins against burst, discussed in the TS are maintained for both normal and postulated accident conditions.

The proposed change does not affect other systems, structures, components, or operational features. Therefore, the proposed changes result in no significant increase in the probability of the occurrence of a SG tube rupture event.

At normal operating pressures, leakage from stress corrosion cracking below the proposed limited tube repair depth is limited by both the tube-to-tubesheet crevice and the limited crack opening permitted by the tubesheet constraint. Consequently, negligible normal operating leakage is expected from cracks within the tubesheet region. The consequences of a SG tube rupture event are affected by the primary-to-secondary leakage flow during the event. Primary-to-secondary leakage flow through a postulated broken tube is not affected by the proposed change since the tubesheet enhances the tube integrity in the region of the hydraulic expansion by precluding tube deformation beyond its initial hydraulically expanded outside diameter.

The probability of a steam line break event is unaffected by the potential failure of a SG tube, as this failure is not an initiator for a steam line break event.

The consequences of a steam line break event are also not significantly affected by the proposed change. During a steam line break event, the reduction in pressure above the tubesheet on the shell side of the SG creates an axially uniformly distributed load on the tubesheet due to the reactor coolant system pressure on the underside of the tubesheet. The resulting bending action constrains the tubes in the tubesheet, thereby restricting primary-to-secondary leakage below the midplane.

Primary-to-secondary leakage from tube degradation in the tubesheet area during the limiting accident (i.e., a steam line break event) is limited by flow restrictions resulting from the crack and tube-to-tubesheet contact pressures that provide a restricted leakage path above the indications and also limit the degree of potential crack face opening as compared to free span indications. The primary-to-secondary leak rate from tube degradation in the tubesheet region during postulated steam line break event conditions will be no more than twice that allowed during normal operating conditions when the pressure boundary is relocated to the 17 inch depth. Since normal operating leakage is limited to 75 gallons-per-day through any one SG per the proposed license condition, the associated accident condition leak rate, assuming all leakage to be from lower tubesheet indications, would be limited to 150 gallons-per-day per SG. This is the value that is assumed in the steam line break dose analysis.

Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

<u>Second Standard</u>

Does operation of the facility in accordance with the proposed amendment create the possibility of a new or different kind of accident from any accident previously evaluated? No.

The proposed change does not introduce any new equipment, create new failure modes for existing equipment, or create any new limiting single failures. Plant operation will not be altered, and all safety functions will continue to be performed as previously assumed in accident analyses. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

<u>Third Standard</u>

Does operation of the facility in accordance with the proposed amendment involve a significant reduction in the margin of safety? No.

The proposed change maintains the required structural margins of the SG tubes for both normal and accident conditions. NEI 97-06 and the Catawba TS are used as the bases in the development of the limited tubesheet tube repair depth methodology for determining that SG tube integrity considerations are maintained within acceptable limits. Regulatory Guide 1.121 describes a method acceptable to the NRC for meeting General Design Criterion (GDC) 14, “Reactor coolant pressure boundary,” GDC 15, “Reactor coolant system design,” GDC 31, “Fracture prevention of reactor coolant pressure boundary,” and GDC 32, “Inspection of reactor coolant pressure boundary,” by reducing the probability and consequences of a SG tube rupture event. By determining the limiting safe conditions for tube wall degradation, the probability and consequences of a SG tube rupture event are reduced. Safety factors are used for loads for tube burst that are consistent with the requirements of Section III of the American Society of Mechanical Engineers (ASME) Code.

For axially oriented cracking located within the tubesheet, tube burst is precluded due to the presence of the tubesheet. For circumferentially oriented cracking, the analysis provided in support of this proposed amendment defines a length of degradation free expanded tubing that provides the necessary resistance to tube pullout due to the pressure induced forces, with applicable safety factors applied. Application of the limited tubesheet tube repair depth criterion (17 inches) will preclude unacceptable primary-to-secondary leakage during all plant conditions.

Therefore, the proposed change does not involve a significant reduction in any margin of safety.

Based upon the preceding discussion, Duke Energy Corporation has concluded that the proposed amendment does not involve a significant hazards consideration.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff determines that the proposed amendment involves no significant hazards consideration.

6.0 STATE CONSULTATION

In accordance with the Commission's regulations, the South Carolina State official was notified of the proposed issuance of the amendments. The State official had no comments.

7.0 ENVIRONMENTAL CONSIDERATION

The amendment changes requirements with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20, and changes the surveillance requirements. The NRC staff has determined that the amendment involves no significant increase in the amounts and no significant change in the types of any effluents that may be released offsite and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (71 FR 9169). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

8.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

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