April 28, 2005

Mr. Rick A. Muench
President and Chief Executive Officer
Wolf Creek Nuclear Operating Corporation
Post Office Box 411
Burlington, KS 66839

SUBJECT: WOLF CREEK GENERATING STATION - ISSUANCE OF EXIGENT AMENDMENT RE: STEAM GENERATOR (SG) TUBE SURVEILLANCE PROGRAM (TAC NO. MC6757)

Dear Mr. Muench:

The Commission has issued the enclosed Amendment No. **162** to Facility Operating License No. NPF-42 for the Wolf Creek Generating Station (WCGS). The amendment consists of changes to the Technical Specifications (TSs) in response to your application dated April 18, 2005 (ET 05-0001), as supplemented by the letter dated April 19, 2005 (ET 05-0002).

The amendment revises TS 5.5.9, "Steam Generator (SG) Tube Surveillance Program," to add changes to the SG inspection scope for WCGS for only the current refueling outage 14 and the subsequent operating cycle. Specifically, the amendment modifies the inspection requirements for portions of the SG tubes within the hot leg tubesheet region of the SGs.

This amendment is being issued under exigent circumstances in accordance with 50.91(a)(6) of Title 10 of the *Code of Federal Regulations*. The exigent circumstances and the final no significant hazards considerations are addressed in Section 7.0 and 8.0 of the enclosed Safety Evaluation (SE).

A copy of our related SE is enclosed. The Notice of Issuance will be included in the Commission's next biweekly *Federal Register* notice.

Sincerely,

/RA/

Jack N. Donohew, Senior Project Manager, Section 2
Project Directorate IV
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-482

Enclosures: 1. Amendment No. 162 to NPF-42
2. Safety Evaluation

cc w/encls: See next page

cc:
Jay Silberg, Esq.
Shaw Pittman, LLP
2300 N Street, NW
Washington, D.C. 20037

Regional Administrator, Region IV
U.S. Nuclear Regulatory Commission
611 Ryan Plaza Drive, Suite 400
Arlington, TX 76011-7005

Senior Resident Inspector
U.S. Nuclear Regulatory Commission
P.O. Box 311
Burlington, KS 66839

Chief Engineer, Utilities Division
Kansas Corporation Commission
1500 SW Arrowhead Road
Topeka, KS 66604-4027

Office of the Governor
State of Kansas
Topeka, KS 66612

Attorney General
120 S.W. 10th Avenue, 2nd Floor
Topeka, KS 66612-1597

County Clerk
Coffey County Courthouse
110 South 6th Street
Burlington, KS 66839

Vick L. Cooper, Chief
Air Operating Permit and Compliance
Section
Kansas Department of Health
and Environment
Bureau of Air and Radiation
1000 SW Jackson, Suite 310
Topeka, KS 66612-1366

Vice President Operations/Plant Manager
Wolf Creek Nuclear Operating Corporation
P.O. Box 411
Burlington, KS 66839

Supervisor Licensing
Wolf Creek Nuclear Operating Corporation
P.O. Box 411
Burlington, KS 66839

U.S. Nuclear Regulatory Commission
Resident Inspectors Office/Callaway Plant
8201 NRC Road
Steedman, MO 65077-1032

April 28, 2005

Mr. Rick A. Muench
President and Chief Executive Officer
Wolf Creek Nuclear Operating Corporation
Post Office Box 411
Burlington, KS 66839

SUBJECT: WOLF CREEK GENERATING STATION - ISSUANCE OF EXIGENT AMENDMENT RE: STEAM GENERATOR (SG) TUBE SURVEILLANCE PROGRAM (TAC NO. MC6757)

Dear Mr. Muench:

The Commission has issued the enclosed Amendment No. 162 to Facility Operating License No. NPF-42 for the Wolf Creek Generating Station (WCGS). The amendment consists of changes to the Technical Specifications (TSs) in response to your application dated April 18, 2005 (ET 05-0001), as supplemented by the letter dated April 19, 2005 (ET 05-0002).

The amendment revises TS 5.5.9, "Steam Generator (SG) Tube Surveillance Program," to add changes to the SG inspection scope for WCGS for only the current refueling outage 14 and the subsequent operating cycle. Specifically, the amendment modifies the inspection requirements for portions of the SG tubes within the hot leg tubesheet region of the SGs.

This amendment is being issued under exigent circumstances in accordance with 50.91(a)(6) of Title 10 of the *Code of Federal Regulations*. The exigent circumstances and the final no significant hazards considerations are addressed in Section 7.0 and 8.0 of the enclosed Safety Evaluation (SE).

A copy of our related SE is enclosed. The Notice of Issuance will be included in the Commission's next biweekly *Federal Register* notice.

Sincerely,

/RA/

Jack N. Donohew, Senior Project Manager, Section 2
Project Directorate IV
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-482

Enclosures: 1. Amendment No. 162 to NPF-42
2. Safety Evaluation

cc w/encls: See next page

DISTRIBUTION
PUBLIC
PDIV-2 Reading
RidsNrrDlpmLpdiv(HBerkow)
RidsNrrDlpmLpdiv2(RGramm)
RidsNrrPMJDonohew
RidsNrrLALFeizollahi
RidsNrrDlpmDpr (RKuntz)
RidsAcrsAcnwMailCenter
RidsRgn4MailCenter (DGraves)
GHill (2)
TBoyce
LLund
EMurphy
RidsOgcRp

TS: ML051230471 NRR-100
PKG.: ML051230044
ACCESSION NO.: ML051160100

NRR-058 w/comments

OFFICE	PDIV-2/PM	PDIV-2/LA	EMEB-C/SC	OGC Nlo	PDIV-2/SC
NAME	JDonohew:sp	DBaxley for LFeizollahi	LLund	APHoefling	RGramm
DATE	4/27/05	4/28/05	04/26/2005	4/28/05	4/28/05

DOCUMENT NAME: E:\Filenet\ML051160100.wpd

OFFICIAL RECORD COPY

WOLF CREEK NUCLEAR OPERATING CORPORATION

WOLF CREEK GENERATING STATION

DOCKET NO. 50-482

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 162
License No. NPF-42

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

   A. The application for amendment to the Wolf Creek Generating Station (the facility) Facility Operating License No. NPF-42 filed by the Wolf Creek Nuclear Operating Corporation (the Corporation), dated April 18, 2005, as supplemented by the letter dated April 19, 2005, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, as amended, 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;

   D. The issuance of this license 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 amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and Paragraph 2.C.(2) of Facility Operating License No. NPF-42 is hereby amended to read as follows:

   2. Technical Specifications

      The Technical Specifications contained in Appendix A, as revised through Amendment No. 162, and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated in the license. The Corporation shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. The license amendment is effective as of its date of issuance and shall be implemented before entry into Mode 4 in the restart from the current Refueling Outage 14.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/
Robert A. Gramm, Chief, Section 2
Project Directorate IV
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: April 28, 2005

ATTACHMENT TO LICENSE AMENDMENT NO. 162

FACILITY OPERATING LICENSE NO. NPF-42

DOCKET NO. 50-482

Replace the following pages of the Appendix A Technical Specifications with the attached pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change. The corresponding overleaf pages are provided to maintain document completeness.

REMOVE	INSERT
5.0-12	5.0-12
5.0-13	5.0-13
5.0-14	5.0-14
5.0-15	5.0-15

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 162 TO FACILITY OPERATING LICENSE NO. NPF-42

WOLF CREEK NUCLEAR OPERATING CORPORATION

WOLF CREEK GENERATING STATION

DOCKET NO. 50-482

1.0 INTRODUCTION

By application dated April 18, 2005, as supplemented by letter dated April 19, 2005, Wolf Creek Nuclear Operating Corporation (the licensee) requested changes to the Technical Specifications (TSs, Appendix A to Facility Operating License No. NPF-42) for the Wolf Creek Generating Station (WCGS) regarding the steam generator (SG) tube inspection and plugging requirements. This amendment request involves a one time change to TS 5.5.9, "Steam Generator (SG) Tube Surveillance Program," regarding the required scope of SG tube inspections and plugging for WCGS during Refueling Outage (RFO) 14 and the subsequent operating cycle. The proposed changes modify the inspection and plugging requirements for portions of the SG tubes within the hot leg tubesheet region of the SGs.

The supplemental letter dated April 19, 2005, provided additional information that clarified the application, did not expand the scope of the application as originally noticed and did not change the NRC staff's original proposed no significant hazards consideration determination. Because there was a short time interval between identification of the need for a TS change and the actual performance of the TS requirement, the licensee requested that the amendment request be considered under exigent circumstances and the NRC staff published a public notice providing the no significant hazards consideration in two local newspapers, the *Coffey County Republican* (on April 22 and 26, 2005) and the *Emporia Gazette* (on April 25 and 26, 2005).

In its application, the licensee submitted the proprietary and non-proprietary versions of Westinghouse's topical report LTR-CDME-05-82, "Limited Inspection of the Steam Generator Tube Portion Within the Tubesheet at Wolf Creek Generating Station," dated April 2005. The NRC staff's determination, in accordance with Section 2.390 of Title 10 of the *Code of Federal Regulations* (10 CFR), to withhold the information designated as proprietary is given in the NRC letter to Westinghouse dated April 27, 2005.

2.0 BACKGROUND

WCGS has four model F SGs designed and fabricated by Westinghouse. There are 5626 tubes in each SG, each with an outside diameter of 0.688 inches and a nominal wall thickness of 0.040 inches. The tubes are hydraulically expanded for the full depth of the tubesheet at each end and are welded to the tubesheet at the bottom of each expansion. WCGS operates with a hot-leg temperature of 618 degrees Fahrenheit (618 EF) and has operated for approximately 16.5 effective full power years (EFPY) as of RFO 14. Prior to RFO 14, the only tube degradation identified in the WCGS SGs is related to tube wear (from loose parts or anti-vibration bars). No corrosion-related tube degradation had been identified.

The licensee has been using bobbin 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 coil probes in a region extending from 3 inches above the top of the tubesheet (TTS) to 3 inches below the TTS. 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 WCGS or elsewhere in the U.S.

In the fall of 2004, crack-like indications were found in tubes in the tubesheet region of Catawba Nuclear Station (Catawba), Unit 2's Westinghouse model D5 SGs. Like WCGS' SGs, the Catawba, Unit 2, SGs have thermally treated Alloy 600 tubing that is hydraulically expanded against the tubesheet. Catawba, Unit 2, had accumulated 14.7 EFPY service, slightly less than WCGS, with a comparable hot leg operating temperature. The crack-like indications at Catawba, Unit 2, 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.

As a result of the Catawba, Unit 2, findings, the WCGS licensee plans to expand the scope of previous rotating coil inspections to address the potential for cracks within the thickness of the tubesheet down to 17 inches below the TTS. However, 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 of tubes as would be required by the TS should inspection reveal cracks in this region, 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 and that the lower 4-inch region be excluded from current inspection requirements. In addition, the licensee proposed new requirements defining the minimum inspection scope with rotating coils for the upper region of the tubesheet region and requiring that all tubes found with degradation in this region be plugged.

3.0 REGULATORY EVALUATION

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

10 CFR Part 50 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 possible" (GDC 30), and shall be designed to permit "periodic inspection and testing...to assess... structural and leak tight integrity" (GDC 32). To this end, 10 CFR 50.55a specifies that components which 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) (i.e., the ASME Code). Section 50.55a further requires, in part, that throughout the service life of a pressurized water reactor (PWR) facility ASME Code Class 1 components meet the requirements, except design and access provisions and pre-service examination 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 TS.

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 an SG tube rupture (SGTR) and main steamline break (MSLB). These analyses consider the primary-to-secondary leakage through the tubing that may occur during these events and must show that the offsite radiological consequences do not exceed the GDC 19 criteria for control room operator doses or the applicable limits of the 10 CFR Part 100 guidelines for offsite doses or some fraction thereof as appropriate to the accident, or the NRC-approved licensing basis (e.g., a small fraction of these limits).

Under the plant TS SG surveillance 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, at the same time, allowing for potential flaw size measurement error and flaw growth between SG inspections. The required frequency and scope of tubing examinations and the tube plugging limits are specified in TS 5.5.9, “Steam Generator (SG) Tube Surveillance Requirements.”

The subject TS amendment request concerns the portions of the tubing that are subject to the TS SG tube surveillance requirements, including any necessary plugging or repairs, and the inspection methods to be employed. TS 5.5.9 defines a tube inspection as an inspection of the SG tube from the point of entry (hot leg side) completely around the U-bend to the top support of the cold leg. This includes the full length of tubing within the thickness of the tubesheet on the hot leg side.

The proposed license amendment would limit the required inspections, 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 4 to 6 inches below the TTS. The larger inspection zone for W* relative to F* is allowed because the explosively expanded joints do not exhibit as much residual interference fit as do hard rolled joints.

This proposed one-time license amendment for WCGS for RFO 14 and the subsequent operating cycle is similar to the one time license amendment request recently submitted for Braidwood Station (Braidwood), Unit 2, by letter dated April 14, 2005. The requested amendments for WCGS and Braidwood, Unit 2, are the first to exclude a portion of tubing in the tubesheet from TS SG inspection and plugging requirements for plants where the tubes are hydraulically expanded against the tubesheet. A previously submitted license amendment request dated October 3, 2002, for Callaway Nuclear Station, which has hydraulically expanded tubing, was withdrawn by the licensee.

4.0 LICENSEE'S PROPOSED TS CHANGES

- TS 5.5.9b, "Steam Generator Tube Sample Selection and Inspection"

  The following new requirement would be added as TS 5.5.9b.4):

  "During Refueling Outage 14, a sample of the SG B and C inservice tubes from the top of the hot leg tubesheet to 10 inches below the top of the tubesheet shall be inspected by rotating probe. This sample shall include a 20% minimum sample of the total population of bulges and overexpansions within the SG from the top of the hot leg tubesheet to 17 inches below the top of the tubesheet."

- TS 5.5.9d.1.f) "Plugging Limit"

  TS 5.5.9d.1.f) currently states:

  "Plugging Limit means the imperfection depth at or beyond which the tube shall be removed from service and is equal to 40% of the nominal tube wall thickness;"

  This criterion would be revised as follows:

  "Plugging Limit means the imperfection depth at or beyond which the tube shall be removed from service and is equal to 40% of the nominal tube wall thickness**. During Refueling Outage 14 and the subsequent operating cycle, this criterion does not apply to degradation identified in the portion of the tube below 17 inches from the top of the hot leg tubesheet. Degradation found in the portion of the tube below 17 inches from the top of the hot leg tubesheet does not require plugging. During Refueling Outage 14 and the subsequent operating cycle, all tubes with degradation identified in the portion of the tube within the region from the top of the hot leg tubesheet to 17 inches below the top of the tubesheet shall be removed from service**;"

  The above text in **bold** is being added to TS 5.5.9d.1.f).

- TS 5.5.9d.1.h) "Tube Inspection"

  TS 5.5.9d.1.h) currently states:

  "Tube Inspection means an inspection of the steam generator tube from the tube end (hot leg side) completely around the U-bend to the top support of the cold leg; and"

  This criterion would be revised as follows:

  "Tube Inspection means an inspection of the steam generator tube from the tube end (hot leg side) completely around the U-bend to the top support of the cold leg**. During Refueling Outage 14 and the subsequent operating cycle, the portion of the tube below 17 inches from the top of the hot leg tubesheet is excluded**;"

  The above text in **bold** is being added to TS 5.5.9d.1.h) and the word "and' is being moved to the end of TS 5.5.9d.1.i).

- TS 5.5.9d, "Acceptance Criteria"

  A new TS 5.5.9d.j) would be added defining "bulge" and "overexpansion:"

  "j) During Refueling Outage 14 and the subsequent operating cycle:

  Bulge refers to a tube diameter deviation within the tubesheet of 18 volts or greater as measured by bobbin coil probe; and

  Overexpansion refers to a tube diameter deviation within the tubesheet of 1.5 mils or greater as measured by bobbin probe."

  Also, the word "and" would be added to the end of TS 5.5.9d.1.i) because the new TS 5.5.9d.1.j) becomes the last item in the list in place of TS 5.5.9d.1.i).

5.0 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 the ASME Code, Section III, not as a friction or expansion joint. The weld itself was designed as a pressure boundary element in accordance the 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 the lower 4 inches of the 21-inch deep tubesheet region from a tube inspection (see proposed change to TS 5.5.9d.1.h, "Tube Inspection") and to exempt tubes with flaw indications in the lower 4-inch zone from the need to plug or repair (see proposed revision to TS 5.5.9d.1.f, "Plugging Limit"). The latter part of this proposal (i.e., to exempt tubes from plugging or repair) is needed as a practical matter since, although rotating coil probe inspections will not be performed in this region, the bobbin probe will necessarily be recording any signals produced in this zone. 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 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 the ASME Code, Section III as is discussed in Section 5.1 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 TS are in compliance with all applicable regulations. The NRC staff's evaluation of joint structural integrity and leakage integrity is discussed in Sections 5.1 and 5.2 of this SE, respectively.

The licensee has also proposed on a one-time basis to add a specific requirement to perform a rotating coil examination of a 20 percent sample of the bulges and over-expansions within the upper 17-inch inspection zone (see proposed addition to TS 5.5.9.b, "Steam Generator Tube Sample Selection and Inspection"). The NRC staff has no objection to this new requirement since the 20 percent sample size (which is based on industry guidelines) exceeds current TS minimum sample size requirements (i.e., 3 percent) and, thus, is more conservative than the currently applicable requirement. However, bulges and over-expansions selected for this sample would be taken exclusively from the upper 10 inches of the tubesheet region with the intention of biasing the sample to focus on the most safety significant portion of the tubesheet region. This is acceptable to the NRC staff because the degradation conditions in the upper 10 inches are expected to be representative of those over the "17-inch inspection zone." Should these inspections identify flaw indications, additional inspection samples may be required as defined in the current TS (industry guidelines contain more conservative sample expansion criteria) which, depending on what is found during the initial 20 percent sample, could ultimately lead to an inspection of all bulges and over-expansions in the 17-inch inspection zone.

To clarify the above proposed requirement, the licensee has proposed adding definitions of "bulge" and "over-expansion" as discussed in Section 4.0 of this SE. The licensee states that the definition of "bulge" (i.e., tube diameter deviation producing an 18 volt bobbin response) is approximately equivalent to the voltage response of a 1 mil over-expansion and is just above the lowest voltage that can be reasonably differentiated from noise. The definition of "over-expansion" (i.e., tube diameter deviation of 1.5 mils or greater) is intended to ensure that tube diameter deviations of this magnitude are inspected, irrespective of voltage amplitude. The NRC staff notes that these definitions clarify how the proposed rotating coil sampling plan for bulges and over-expansions is to be implemented. Thus, the NRC staff finds the proposed definitions to be acceptable.

The licensee is also proposing on a one-time basis to plug all tubes found with degradation in the upper 17-inch region of the tubesheet (see proposed revision to TS 5.5.9d.1.f, "Plugging Limit"). "Degradation" is defined in the TSs to mean service-induced cracking, wastage, wear, or general corrosion occurring on either the inside or outside of a tube. This definition is not limited to flaws that exceed the current TS 40 percent plugging limit. The NRC staff finds this proposed requirement 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 pullout under tube end cap pressure loads and in resisting primary-to-secondary leakage between the tube and tubesheet.

5.1 Joint Structural Integrity

Westinghouse has conducted analysis and testing to establish the engagement (embedment) length of hydraulically expanded tubing inside the tubesheet that is necessary to resist pullout under normal operating and DBA conditions. Pullout 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 pullout derives from the pressure end cap loads due to the primary to secondary pressure differentials associated with normal operating and design basis accident conditions. The licensee’s contractor, Westinghouse, determined the required engagement distance on the basis of maintaining a factor of 3 against pullout under normal operating conditions and a factor of 1.4 against pullout under accident conditions. The NRC staff concurs that these are the appropriate safety factors to apply to demonstrate structural integrity. As documented in detail in an SE accompanying the NRC staff’s approval of new performance based SG TSs for Farley, 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 plant design basis (i.e., the stress limit criteria in ASME Code, Section III).

The resistance to pullout 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 including (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 pullout tests and analyses, including finite element analyses, to evaluate these contributors. Based on these analyses and tests, Westinghouse concludes that the required engagement distances to ensure the safety factor criteria against pullout are achieved vary from about 5.8 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 5.8 to 8.6 inches of engagement (termed H* criterion by Westinghouse) is adequate to ensure that the necessary safety margins against pullout 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 5.8 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.

- The NRC staff estimates based on the Westinghouse pullout tests indicate that the radial contact pressure produced by the hydraulic expansion and differential radial thermal expansion is such as to require an engagement distance of 8.6 inches to ensure the appropriate safety margins against pullout based on no-slip. This estimate is a mean minus one standard deviation estimate based on six pullout tests. This estimate ignores the effect on needed engagement distance from internal primary pressure in the tube and tubesheet bore dilations associated with tubesheet bow. The NRC staff notes that from a tube pullout standpoint, the use of a “no slip” criterion is conservative. Allowing slippage of about 0.2 to 0.3 inches decreases the necessary engagement distance to 5.1 inches, again ignoring the effect on needed engagement distance from internal primary pressure in the tube and tubesheet bore dilations associated with tubesheet bow.

- 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.

- The 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 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 tubesheet), 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 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 pullout tests. A similar logic applied to the periphery of the tubesheet leads the NRC staff to conclude that at the top 10.5 inches of the tubesheet region, contact pressure should be considerably higher than the contact pressure simulated in the above mentioned pullout tests. Thus, the NRC staff concludes that the proposed 17-inch engagement distance (or inspection zone) is acceptable to ensure the structural integrity of the tubesheet joint.

5.2 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, not inspecting the tubing 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 limiting condition for operation (LCO) limits in the TSs. However, it must also be demonstrated that the proposed TS changes do not create the potential for leakage during design basis accidents 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 1 gallon per minute (gpm) in the faulted SG during a 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 MSLB will not exceed two times the observed leakage during normal operation. Westinghouse refers to this as the “bellwether approach.” Thus, for SG leaking at the TS LCO limit (i.e., 500 gallons per day (gpd) or 0.347 gpm) under normal operating conditions, Westinghouse estimates that leakage would not be expected to exceed 0.694 gpm, which is less than the 1 gpm assumed in the licensing basis accident analyses for 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 pound per square inch 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 finds 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 11-inch zone in the upper 17 inches of the tubesheet where there is an increase in joint contact pressure, and, thus, leak flow resistance, due to higher primary pressure inside the tube and changes in tubesheet bore dilation along the length of the tubes. In Section 5.1 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 pressured conditions should result in an increase in joint contact pressure. The contact pressure due to changes in tubesheet bore dilation should increase further over the 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 NRC staff estimates on a qualitative basis that the length over which contact pressure would be expected to increase should be at least 10 inches, which is reasonably consistent with the more detailed Westinghouse analysis.

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 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 should be no significant reduction in leakage flow resistance when going from normal operating to accident conditions.

Finally, 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 TS LCO operational 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, minimal 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.

5.3 Conclusions

The NRC staff finds that the proposed one time license amendment 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, in accordance with the applicable regulations without undue risk to public health and safety. Based on this, the NRC staff concludes that the proposed changes to TS 5.5.9 are acceptable.

6.0 REGULATORY COMMITMENTS

In its application, the licensee made the following two regulatory commitments concerning if cracking were found in the inspections of the SG tubes in RFO 14:

> If cracking is found in the sample population of bulges or overexpansions, the inspection scope will be increased to 100% of the bulges and overexpansions population for the region from the top of the hot leg tubesheet to 17 inches below the top of the tubesheet in the affected steam generator and 20% of the bulges

and overexpansions population in the unaffected steam generators from the top of the hot leg tubesheet to 17 inches below the top of the tubesheet.

If cracking is reported at one or more tube locations not designated as either a top of the tubesheet expansion transition, a bulge or an overexpansion, an engineering evaluation will be performed. This evaluation will determine the cause for the signal, e.g., some other tube anomaly, in order to identify a critical area for the expansion of the inspection. This expanded inspection will be limited to the identified critical area within 17 inches from the top of the hot leg tubesheet.

For the cases of cracking found in either (1) the sample population of bulges or overexpansions or (2) at one or more tube locations not designated as either a top of the tubesheet expansion transition, a bulge or an overexpansion, the licensee committed to increase the inspection scope or perform an evaluation prior to startup from RFO 14.

The NRC staff's acceptance of guidance issued by the Nuclear Energy Institute (NEI) on licensees managing regulatory commitments made to the NRC is described in Regulatory Information Summary (RIS) 2000-17, "Managing Regulatory Commitments Made by Power Reactor Licensees to the NRC Staff."

7.0 EXIGENT CIRCUMSTANCES

The regulations at 10 CFR 50.91 contain provisions for issuance of amendments when the usual 30-day public comment period cannot be met. One type of special exception is an exigency. An exigency is a case where the NRC staff and licensee need to act promptly. In this case, there is insufficient time to process the license amendment request within the normal time frame. Pursuant to 10 CFR 50.91(a)(6), the licensee requested the proposed amendment on an exigent basis.

Under such circumstances, the Commission notifies the public in one of two ways: (1) by issuing a *Federal Register* notice providing an opportunity for hearing and allowing at least two weeks for prior public comments, or (2) by issuing a press release discussing the proposed changes, using local media. In this case, the Commission used the second approach and published a public notice in two local newspapers, the *Coffey County Republican* (on April 22 and 26, 2005) and the *Emporia Gazette* (on April 25 and 26, 2005).

In its application, the licensee discussed the need for an exigent review of the proposed license amendment. The licensee stated that, on December 17, 2004, the industry was notified that tube degradation had been detected in the tubesheet region in Catawba, Unit 2, Westinghouse Model D5 SGs. Additional details of the findings at Catawba, Unit 2, are contained in NRC Information Notice (IN) 2005-09, “Indications in Thermally Treated Alloy 600 Steam Generator Tubes and Tube-to-Tubesheet Welds,” issued on April 7, 2005. The licensee explained that the Catawba, Unit 2, SGs are similar to the WCGS Model F SGs in that they both contain Alloy 600 thermally treated tubes and this was considered in the WCGS RFO 14 degradation assessment. The licensee stated that the degradation assessment concluded that tube degradation could occur in the WCGS SG tubesheet region and specific potential degradation locations are the top of tubesheet expansion transition region, tube bulge or overexpansion locations, tack expansion region and degradation propagating from the tube-end weld into the tube. In preparation for the inspection of these regions, the licensee stated that a method for dispositioning potential indications within the tube-end weld was initiated. An evaluation was performed to justify a limited tubesheet inspection in the region from the top of the hot leg tubesheet to 17 inches below the top of the tubesheet and it was concluded in the evaluation that degradation contained within the tube-end weld could not be properly addressed by ASME Code analysis methods. It was also concluded in the evaluation that the bottom portion of the tube is not a critical portion of the tube necessary to maintain structural and leakage integrity.

Also, in its application, the licensee explained that it had, in January 2005, requested a telephone conference call with the NRC staff to discuss the proposed SG tube inspections planned for RFO 14 and the NRC staff determined that a telephone conference call was not necessary. Based on that, the licensee stated that it proceeded with the planning of the tube inspections based in part on how the Alloy 600 mill-annealed licensees responded to Generic Letter (GL) 2004-01, “Requirements for Steam Generator Tube Inspections,” dated August 30, 2004, and indications from the NRC on the acceptability of the licensee's response to the GL. Subsequently, the NRC staff requested a telephone conference call on April 14, 2005 to discuss the SG tube inspections particularly in the tubesheet region. It was not until the call on April 14, 2005, that the NRC staff understood that the licensee was applying a limited tube inspection in the tubesheet region in areas where degradation potential could occur. Based on this telephone conference call, the licensee concluded that its limited tube inspection required a change to TS 5.5.9, “Steam Generator (SG) Tube Surveillance Program,” and submitted its application dated April 18, 2005.

The licensee concluded that, due to the short time interval between identification of the need for a TS change to allow a limited SG tube inspection scope based on the guidance provided in IN 2005-09, and the actual performance of the WCGS SG tube inspection in the current refueling outage, insufficient time remains for normal NRC processing and notification. Therefore, the licensee requested that this proposed TS change be considered under exigent circumstances as described in 10 CFR 50.91(a)(6) and that the amendment be issued on April 27, 2004, to support the completion of the SC tube inspections.

On the basis of minimal time available between the telephone conference call with the licensee, the additional guidance contained in IN 2005-09, and the completion of the WCGS current outage (i.e., RFO 14), the NRC staff concludes that there was insufficient time available in which to process a normal license amendment request. Therefore the NRC staff has determined that a valid need exists for issuance of the license amendment in accordance with the exigent provisions of 10 CFR 50.91(a)(6).

8.0 FINAL NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

The Commission’s regulations in 10 CFR 50.92 state that the Commission may make a final determination that a license amendment involves no significant hazards considerations 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 accident previously evaluated; or (3) involve a significant reduction in the margin of safety. In Section 5.1 of its application, the licensee provided the following no significant hazards consideration:

(1) Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

Response: No

The previously analyzed accidents are initiated by the failure of plant structures, systems, or components. The proposed change that alters the steam generator inspection criteria do 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 changes to the steam generator tube inspection criteria, are the steam generator tube rupture (SGTR) event and the steam line break (SLB) accident.

During the SGTR event, the required structural integrity margins of the steam generator tubes will be maintained by the presence of the steam generator tubesheet. Steam generator tubes are hydraulically expanded in the tubesheet area. Tube rupture in tubes with cracks in the tubesheet 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 from the differential pressure between the primary and secondary side. Based on this design, the structural margins against burst, discussed in Regulatory Guide (RG) 1.121, “Bases for Plugging Degraded PWR Steam Generator Tubes,” 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 SGTR accident.

At normal operating pressures, leakage from primary water stress corrosion cracking (PWSCC) below the proposed limited inspection 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 an SGTR 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 SLB is unaffected by the potential failure of a steam generator tube as this failure is not an initiator for a SLB.

The consequences of a SLB are also not significantly affected by the proposed change. During a SLB accident, the reduction in pressure above the tubesheet on the shell side of the steam generator 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 SLB) 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 during postulated SLB accident conditions would be expected to be less than that during normal operation for indications near the bottom of the tubesheet (i.e., including indications in the tube-end welds). This conclusion is based on the observation that while the driving pressure causing leakage increases by approximately a factor of two, the flow resistance associated with an increase in the tube-to-tubesheet contact pressure, during a SLB, increases by approximately a factor of 6. While such a leakage decrease is logically expected, the postulated accident leak rate could be conservatively bounded by twice the normal operating leak rate if the increase in contact pressure is ignored. Since normal operating leakage is limited to less than 0.347 gpm (500 gpd) per TS 3.4.13, “RCS Operational LEAKAGE,” the associated accident condition leak rate, assuming all leakage to be from lower tubesheet indications, would be bounded by 0.694 gpm. This value is well within the assumed accident leakage rate of 1.0 gpm discussed in WCGS Updated Safety Analysis Report, Table 15.1-3, “Parameters Used in Evaluating the Radiological Consequences of a Main Steam Line Break.” Hence it is reasonable to omit any consideration of inspection of the tube, tube-end weld, bulges/overexpansions or other anomalies below 17 inches from the top of the hot leg tubesheet. Therefore, the consequences of a SLB accident remain unaffected.

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

(2) Does the proposed change create the possibility of a new or different accident from any accident previously evaluated?

Response: 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 perform as previously assumed in accident analyses. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any previously evaluated.

(3) Does the proposed change involve a significant reduction in a margin of safety?

Response: No

The proposed changes maintain the required structural margins of the steam generator tubes for both normal and accident conditions. Nuclear Energy Institute (NEI) 97-06, “Steam Generator Program Guidelines,” and RG 1.121, “Bases for Plugging Degraded PWR Steam Generator Tubes,” are used as the bases in the development of the limited hot leg tubesheet inspection depth methodology for determining that steam generator tube integrity considerations are maintained within acceptable limits. RG 1.121 describes a method acceptable to the NRC for meeting General Design Criteria (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 SGTR. RG 1.121 concludes that by determining the limiting safe conditions for tube wall degradation the probability and consequences of a SGTR are reduced. This RG uses safety factors on 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, Westinghouse letter LTR-CDME-05-82-P, “Limited Inspection of the Steam Generator Tube Portion Within the Tubesheet at Wolf Creek Generating Station,” 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 hot leg tubesheet inspection depth criteria will preclude unacceptable primary-to-secondary leakage during all plant conditions. The methodology for determining leakage provides for large margins between calculated and actual leakage values in the proposed limited hot leg tubesheet inspection depth criteria.

Therefore, the proposed changes do not involve a significant reduction in any margin to safety.

This no significant hazards consideration determination was included in the notices published in the two local newspapers, the *Coffey County Republican* (on April 22 and 26, 2005) and the *Emporia Gazette* (on April 25 and 26, 2005).

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

9.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Kansas State official was notified of the proposed issuance of the amendment. The State official had a comment on the licensee's proposed addition of the new requirement to TS 5.5.9b. The comment was on the statement that "This sample shall include a 20% minimum sample of the total population of bulges and overexpansions within the SG from the top of the hot leg tubesheet to 17 inches below the top of the tubesheet." The comment was that the 20 percent minimum sample of the total population of bulges and overexpansions may not provide a statistically significant result if the population is very small.

The response from the NRC staff are the following statements. The licensee proposed the 20 percent sample in the top 17 inches of the tubesheet because this sampling level is consistent with industry guidelines. Had the licensee not proposed this sample, the existing sampling requirements in the TSs would apply. The existing TS sampling requirements call for an initial 6 percent sample if two of the four SGs are being inspected, as is the case for WCGS during the current outage. The proposed license amendment, therefore, increases the likelihood of detecting indications in the 17-inch zone. The purpose of the license amendment is to delete existing inspection requirements in the 4-inch zone below the top 17 inches within the tubesheet and, for the reasons given in the SE, the staff has concluded that adequate tube integrity will exist without the lower 4 inches of any tubes being inspected. In proposing the amendment, the licensee also proposed a sampling level that is greater than that required in the TSs.

10.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. 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 made a final finding that the amendment involves no significant hazards consideration. 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 amendments.

11.0 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.

Principal Contributor: E. Murphy

Date: April 28, 2005