April 22, 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 - PUBLIC NOTICE OF EXIGENT REQUEST FOR REVISION TO TECHNICAL SPECIFICATION 5.5.9, "STEAM GENERATOR (SG) TUBE SURVEILLANCE PROGRAM" (TAC NO. MC6757)

Dear Mr. Muench:

The enclosed announcement of an exigent license amendment request was forwarded to the *Coffey County Republican* and *Emporia Gazette* for publication. This announcement relates to your application dated April 18, 2005, as supplemented by letter dated April 19, 2005, for amendment to Facility Operating License No. NPF-42 for the Wolf Creek Generating Station (WCGS). The proposed amendment request would revise Technical Specification (TS) 5.5.9, "Steam Generator (SG) Tube Surveillance Program," to incorporate changes in the inspection of the SG tubes for refueling outage 14 and the subsequent operating cycle. WCGS is currently in refueling outage 14.

Sincerely,

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

Docket No. 50-482

Enclosure: Public Notice

cc w/encl: See next page

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PUBLIC NOTICE

NRC STAFF PROPOSES TO AMEND OPERATING LICENSE AT THE

WOLF CREEK GENERATING STATION, UNIT 1

The U.S. Nuclear Regulatory Commission (NRC or the Commission) staff has received an application dated April 18, 2005, and a supplemental letter dated April 19, 2005, from the Wolf Creek Nuclear Operating Corporation (the licensee or WCNOC) for an exigent amendment to the operating license for the Wolf Creek Generating Station, Unit 1, (WCGS) located in Coffey County, Kansas.

The proposed amendment submitted in the licensee's application dated April 18, 2005, as supplemented by its letter dated April 19, 2005, would revise Technical Specification (TS) 5.5.9, "Steam Generator (SG) Tube Surveillance Program," to incorporate changes in the SG inspection scope for refueling outage 14 (the refueling outage that the unit is currently in) and the subsequent operating cycle. The amendment would change the inspection requirements for portions of the SG tubes within the hot leg tubesheet. The licensee requested that the amendment be approved by April 27, 2005, to support the completion of the SG tube inspections in refueling outage 14. The date requested by the licensee supports the completion of SG inspection activities.

During a telephone conference call on April 14, 2005, the licensee and NRC personnel discussed the proposed limited tubesheet inspection for the WCGS SGs based on Westinghouse letter LTR-CDME-05-82-P. It was concluded that the licensee's limited tubesheet inspection in areas where degradation could occur warranted a change to TS 5.5.9. In the absence of such a change, the license might have been in violation of Part 50, Appendix B "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," of Title 10 of the *Code of Federal Regulations* (10 CFR), by virtue of the choice of inspection techniques used to inspect the SG tubes. Limiting the inspection scope would

obviate this potential violation. Additional guidance is contained in NRC Information Notice 2005-09, "Indications in Thermally Treated Alloy 600 Steam Generator Tubes and Tube-to-Tubesheet Welds," issued on April 7, 2005. Information Notice 2005-09 provided further details of the findings at Catawba Nuclear Station, Unit 2.

Due to the short time interval between identification of the need for a TS change to allow a limited SG inspection scope, the actual performance of the WCGS inspection in the refueling outage, and the schedule for restart of the reactor and entry into Mode 4, time does not permit the Commission to publish a *Federal Register* notice allowing 30 days for prior public comment. Therefore, the licensee requested that this proposed TS change be considered under exigent circumstances as described in 10 CFR 50.91(a)(6). The licensee's statement of exigent circumstances is in Section 5.0 of Attachment I to its application dated April 18, 2005.

The licensee and the NRC staff have evaluated this proposed change with regard to the determination of whether or not a significant hazards consideration is involved. As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration in its application, which is presented below.

(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.

Following an initial review of this application, the requested amendment has been evaluated against the standards in 10 CFR 50.92 and the NRC staff has made a proposed determination that the requested amendment involves no significant hazards considerations. The changes do not significantly increase the probability or consequences of any accident previously considered, nor create the possibility of an accident of a different kind, nor significantly decrease any margin of safety.

If the proposed determination that the requested license amendment involves no significant hazards consideration becomes final, the NRC staff will issue the amendment without first offering an opportunity for a public hearing. An opportunity for a hearing will be published in the *Federal Register* at a later date and any hearing request will not delay the effective date of the amendment.

If the NRC staff decides in its final determination that the amendment does involve a significant hazards consideration, a notice of opportunity for a prior hearing will be published in the *Federal Register* and, if a hearing is granted, it will be held before the amendment is issued.

Comments on the proposed determination of no significant hazards consideration may be (1) telephoned to Robert A. Gramm, Chief, Section 2, Project Directorate IV, by collect call to 301-415-1395 or 301-415-1372, or by facsimile to 301-415-3313, (2) e-mailed to rag@nrc.gov, or (3) submitted in writing to the Chief, Rules and Directives Branch, Division of Administrative Services, Office of Administration, U.S. Nuclear Regulatory Commission, Washington, DC 20555. All comments received by close of business on April 27, 2005, from 7:30 a.m. to 4:15 p.m. EDT Federal workdays will be considered in reaching a final determination. A copy of the April 18, 2005, application and April 19, 2005, supplement may be examined electronically through the Agencywide Documents Access and Management System

(ADAMS) Public Electronic Reading Room link at the NRC Web site http://www.nrc.gov/reading-rm/adams.html and at the Commission's Public Document Room (PDR), located at One White Flint North, Public File Area O1 F21, 11555 Rockville Pike (first floor), Rockville, Maryland.

Persons who do not have access to ADAMS or who encounter problems in accessing the documents located in ADAMS, should contact the NRC PDR Reference staff by telephone at 1-800-397-4209, 301-415-4737, or by e-mail to pdr@nrc.gov.

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