February 28, 2008

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 - WITHDRAWAL OF LICENSE AMENDMENT REQUEST ON STEAM GENERATOR TUBE INSPECTIONS (TAC NO. MD0197)

Dear Mr. Muench:

By letter to the Nuclear Regulatory Commission (NRC) dated February 21, 2006 (ET 06-0004), as supplemented by letters dated May 3 and September 27, 2007 (WO 07-0012 and ET 07-0043), and January 25, 2008 (ET 08-0006), you submitted a license amendment request (LAR) to revise Technical Specification (TS) 5.5.9, "Steam Generator (SG) Tube Surveillance Program," to exclude portions of the SG tube below the top of the tubesheet in the SGs from periodic tube inspections based on the application of structural analysis and leak rate evaluation results to re-define the primary-to-secondary pressure boundary. In addition, there were also proposed changes to add new reporting requirements to TS 5.6.10, "Steam Generator Tube Inspection Report."

By letter dated February 14, 2008 (ET 08-0010), you have withdrawn the above request for a license amendment. This letter is to grant your request for withdrawal and provide you with a copy of the Notice of Withdrawal of Application for Amendment to Facility Operating License, which is enclosed and which will be forwarded to the Office of the Federal Register for publication. In your letter, you also requested that we provide you with the specific issues necessary to be resolved on the amendment request. The list of specific issues is provided in the second enclosure.

Sincerely,

/RA/

Balwant K. Singal, Senior Project Manager
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-482

Enclosures: 1. Notice of Withdrawal
2. Specific Issues with Amendment That Need to be Resolved

cc w/encls: See next page

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ADAMS Accession Nos.: ML080450185

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OFFICIAL RECORD COPY

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ENCLOSURE 1

NOTICE OF WITHDRAWAL OF APPLICATION FOR

AMENDMENT TO FACILITY OPERATING LICENSE

UNITED STATES NUCLEAR REGULATORY COMMISSION

WOLF CREEK NUCLEAR OPERATING CORPORATION

DOCKET NO. 50-482

NOTICE OF WITHDRAWAL OF APPLICATION FOR

AMENDMENT TO FACILITY OPERATING LICENSE

The U.S. Nuclear Regulatory Commission (the Commission) has granted the request of Wolf Creek Nuclear Operating Corporation (the licensee) to withdraw its application dated February 21, 2006, with supplemental letters dated May 3 and September 27, 2007, and January 25, 2008, for proposed amendment to Facility Operating License No. NPF-42 for the Wolf Creek Generating Station, located in Coffey County, Kansas.

The proposed amendment would have revised Technical Specification (TS) 5.5.9, "Steam Generator (SG) Tube Surveillance Program," to exclude portions of the steam generator tube below the top of the tubesheet from periodic tube inspections based on the application of structural analysis and leak rate evaluation results to re-define the primary-to-secondary pressure boundary. In addition, there were also proposed changes to add new reporting requirements to TS 5.6.10, “Steam Generator Tube Inspection Report."

The Commission had previously issued a Notice of Consideration of Issuance of Amendment, on the above proposed amendment application, that was published in the FEDERAL REGISTER on April 11, 2006 (71 FR 18377). However, by letter dated February 14, 2008, the licensee withdrew the proposed amendment.

For further details with respect to this action, see the application for amendment dated February 21, 2006, with supplemental letters dated May 3 and September 27, 2007, and

January 25, 2008, and the licensee's letter dated February 14, 2008, which withdrew the application for license amendment. Documents may be examined, and/or copied for a fee, at the NRC's Public Document Room (PDR), located at One White Flint North, Public File Area O1 F21, 11555 Rockville Pike (first floor), Rockville, Maryland. Publicly available records will be accessible electronically from the Agencywide Documents Access and Management Systems (ADAMS) Public Electronic Reading Room on the internet at the NRC Web site, http://www.nrc.gov/reading-rm.html. 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, or 301-415-4737 or by email to pdr@nrc.gov.

Dated at Rockville, Maryland, this 28th day of February 2008.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Balwant K. Singal, Senior Project Manager
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

ENCLOSURE 2

REMAINING TECHNICAL ISSUES RELATED TO

TECHNICAL SPECIFICATION AMENDMENT REQUEST REGARDING
STEAM GENERATOR TUBE INSPECTIONS BASED ON H*/B* METHODOLOGY
TAC NO. MD0197

WOLF CREEK GENERATING STATION
DOCKET NO. 50-482

REMAINING TECHNICAL ISSUES RELATED TO

TECHNICAL SPECIFICATION AMENDMENT REQUEST REGARDING
STEAM GENERATOR TUBE INSPECTIONS BASED ON H*/B* METHODOLOGY
TAC NO. MD0197

WOLF CREEK GENERATING STATION
DOCKET NO. 50-482

References:

1. Westinghouse report WCAP-16932-P (Proprietary), Revision 1 (Agencywide Access and Management System (ADAMS) Accession No. ML031910141), Appendix B, "Tube-to-Tubesheet Joint Strength Analysis," May 2003.

2. Wolf Creek Nuclear Operating Corporation (WCNOC) letter dated February 21, 2006 (ADAMS Accession No. ML060600454), which requested an amendment to the Technical Specifications (TS) for H*/B* and which included a technical support document.

3. WCNOC letter dated May 3, 2007 (ADAMS Accession No. ML071290101), which provided a response to an NRC Request for Additional Information (RAI) dated June 27, 2006, and which also provided a revised amendment request for H*/B*.

4. WCNOC letter dated September 27, 2007 (ADAMS Accession No. ML072750610), which provided a response to an NRC Request for Additional Information (RAI) provided at a July 11, 2007, meeting with WCNOC (see NRC ADAMS Accession No. ML071980203 for meeting summary) and which provided a revised amendment request for H*/B*.

5. Summary of meeting with WCNOC representatives on November 28, 2007 regarding steam generator tube surveillance program (NRC ADAMS Accession No. ML073370045).

Information That Needs to be Provided to the Staff:

The U. S. Nuclear Regulatory Commission (NRC) staff's review of the proposed amendment was intended to ensure that all of the tubes in the Steam Generator (SG) have adequate structural integrity and that the SG has adequate leakage integrity. In reviewing the licensee's analysis supporting its amendment request, the NRC staff determined that the licensee had not provided the information below which is necessary to support a conclusion that the tubes would have adequate integrity if flaws were to occur in the region of the tube not inspected.

- The licensee must adequately demonstrate that the proposed H*/B* distances make adequate allowance for uncertainties such as to ensure with a high probability (e.g., 0.99) that all 22,500 tubes at Wolf Creek Generating Station (WCGS) will exhibit pullout resistances consistent with the TS performance criteria. The licensee's stacked uncertainty model does not conservatively bound the uncertainties for an individual tube. The licensee's discussion of extreme value considerations lacks completeness, not considering certain key variables and combinations of variables.

- The licensee must provide sufficient information to adequately characterize the potential range in values of residual contact pressure between the tube and tubesheet (due to the hydraulic expansion process) which may be encountered among the 22,500 tubes at WCGS. Only limited pullout data exists upon which the residual contact pressures are estimated. The data exhibits significant scatter. Residual contact pressure (and thus residual pullout load capacity) is highly sensitive to several parameters, including hydraulic expansion pressure, tube yield strength, tube material strain hardening properties, and initial radial gap (i.e., pre-expansion) between the tube and tubesheet. The licensee has not provided information to establish whether the pullout test specimens adequately envelop the range of values of these parameters which may be encountered in the WCGS SGs.

- The licensee must provide sufficient information (e.g., literature search for relevant data, expert opinion and experience) to adequately characterize the range in potential values of the thermal expansion coefficient (TEC) for the tubesheet material and tube material which may be encountered in the field. In addition, the licensee has not provided sufficient information concerning potential changes to the tube material TEC due to the hydraulic expansion process (i.e., due to cold working).

- The licensee’s analysis must sufficiently demonstrate that incremental slippage of the tubing under cyclic operational loads will not occur, or compensatory monitoring for slippage should be proposed.

- The method for accounting for the pressure in the crevice between the tube and tubesheet (i.e., limiting median crevice pressure model) must be demonstrated to be conservative for all conditions (e.g., normal operating conditions and steam line break) and for all tubes within the tubesheet.

- At the meeting on December 13, 2007, the licensee identified the use of a “beta factor” which is embedded in the crevice pressure model for at least some of the H*/B* analyses and which may have a significant influence on the H*/B* distances. This beta factor was only briefly discussed during the meeting, but apparently may significantly influence some of the H*/B* analyses. The use of this beta factor, its purpose, and its technical basis must be submitted for NRC review.

- During the December 13, 2007, meeting, the licensee also stated that the method used to scale the results of tubesheet finite element analyses from normal operating conditions to steam line break conditions was inappropriate, leading to over-conservative results for steam line break. The nature of the problem and the resolution must be submitted for NRC review.

The following items are the specific issues that the NRC staff had outstanding when the WCGS amendment was withdrawn:

1. This question does not need to be responded to if the "theory of elasticity model" in Reference 4, Enclosure I, page 7, is not used in the H*/B* analyses.

   Reference 4, Enclosure I, page 7, states that the F/L value in Reference 2, Enclosure I, Table 6-4, is correct based on the theory of elasticity model. However, the supporting discussion for this statement appears to contain inconsistencies and appears to be inconsistent with the original development of this model in Reference 1. Specifically, the theory of elasticity model presented in the middle of page 7 of Enclosure I to Reference 4 is described on page 8 (second to last paragraph) as being applicable to a linear distribution of initial contact pressure through the thickness of the tubesheet (over the length "L"). However, it is the staff's understanding, based on its review of Reference 1, that this model was actually developed assuming an initial contact pressure (P_0) from the hydraulic expansion which is constant through the thickness of the tubesheet (over the length "L"). Also, the NRC staff notes that the above F/L value was determined from Equation 3 on page 8 as a function of the term P_c whose value was assumed equal to P_0 as determined from the theory of elasticity. The NRC staff believes this approach leads to an incorrect value of F/L since the contact pressure will only equal P_0 at a distance "L" from the top of the tubesheet, decreasing to ever smaller values at decreasing distances from the top of the tubesheet. The NRC staff also notes that the stated F/L value suggests that the smallest length test specimen is capable of resisting total axial force, which is substantially higher than the test data indicates. The staff's concerns described above, including a complete description of why the stated F/L value is correct, need to be addressed.

2. Reference 4, Enclosure I, page 9 - The contact pressures and F/L values are based on the "first principle approach" such that they represent average values over the length of the test specimens. Based on the theory of elasticity equation discussed in item 1 above, pullout force is not expected to be linear with the length of the specimen. Thus, it appears to NRC staff that the listed contact pressures and F/L values for the smallest length specimens are not directly comparable to those for the longer specimens for purposes of calculating mean and mean minus 1 sigma values. The NRC staff has the same concern about the pullout data for the Model D5 and 44F specimens. The listed contact pressures and F/L values are only conservative for H*/B* distances less than the length the test specimens. For Model F, the listed contact pressures and F/L values for the smallest length specimens are not conservative if assumed to apply to H*/B* distances greater than that length. (This concern is much more urgent in the case of Model D5 and 44F since the test specimens for these models were substantially shorter) The staff's concerns discussed above need to be addressed including whether there is a need to adjust the pullout data such as to produce mean and mean minus 1 sigma estimates of contact pressure and F/L that are conservative for the range of H*/B* values that may ultimately be proposed.

3. Reference 4, Enclosure I, page 9 - The listed pull forces are described as corresponding to a 0.25 inch displacement. In Reference 2, Enclosure I, Table 6-3, these same pull forces are described as being the maximum value recorded. Table 6-4 implies that the pull force data are smaller for 0.25 inch displacements. Which data are correct? What are the pull force values for tests 1 though 8 corresponding to 0.25 inch displacements?

4. Reference 3, Enclosure 1, Table 1, provided test specimen dimensions for specimens 33625-09 to 33625-33625-15. Since the H*/B* analysis is now being based on pull test results for other specimens, a similar dimensional data for these other specimens (i.e., R12C08, R11C08, R12C07, R11C07, R10C08, R10C07) is needed. Also, the yield strength for each of the eight specimens in Table 1 is also needed.

5. Reference 4, Enclosure I, page 9 - The yield strength for each of the eight specimens in the table are needed.

6. Reference 4, Enclosure I, page 9 - Apart from the scaling issues discussed in Question 2 above, the adequacy of the data set in the Table to address the range of potential values of the key variables affecting the residual contact pressure and F/L values which may be found in the field among the population of 22500 tubes needs to be addressed. Based on the responses to questions 4 and 5 above, what adjustments to the mean and mean minus 1 sigma values of contact pressure and F/L are needed to reflect the full range of all key variables. The NRC staff notes that the spider chart (Figure 2 of Reference 3, Enclosure I) provides a potential tool for making such adjustments.

7. Reference 4, Enclosure I, page 16 – The report states that certain parameters were not directly evaluated as part of the H*/B* sensitivity study. In Table 2-1 on page 19 of the report, the residual contact pressure uncertainty is assumed to be a function of only yield strength and Young's Modulus uncertainties. However, the spider chart in Figure 2 of Reference 3 shows that the residual contact pressure associated with the hydraulic expansion process is highly sensitive to certain variables. For this reason, the response to question 6 above should address hydraulic expansion pressure, strain hardening, and initial tube to tubesheet radial clearance as potential key variables affecting residual contact pressure.

8. Reference 4, Enclosure I, pages 10 and 11 - The pull test data for Model D5 and 44F exhibit considerably more scatter (even in terms of P_0 as determined from the theory of elasticity equation discussed in Question 1) than the Model F data on page 9. An explanation of the Model F data with that for Model D5 and 44F is needed. Whether the data values and scatter are consistent among the models when the data are adjusted to reflect the full range of all the key input parameters affecting the pullout resistance needs to be addressed.

9. Reference 4, Enclosure I, page 13, bottom of page - It's not clear to the NRC staff where the mean minus 1 sigma value of residual contact pressure from hydraulic expansion comes from. The cited reference (Reference 3, Enclosure I, Figure 2) is the "spider chart" which doesn't address this value. Also, this value is slightly inconsistent with the value shown in the Table in Reference 4, Enclosure I. Explain this apparent discrepancy.

10. Reference 4, Enclosure I, page 18 - The report considered two different nominal values of thermal expansion coefficient (TEC) for alloy 600 tubing and two different nominal values for the A508 tubesheets. The nominal TEC values are based on (1) nominal ASME Code values for both the tube and tubesheet and (2) data from ANTER Laboratories Inc. for both the tube and tubesheet. TEC variability relative to these nominal values was assumed to have a standard deviation of 1.5% based on reported measurement uncertainty associated with the ANTER data. However, the report provided no evidence that either TEC model (i.e., that based on Code values versus that based on the ANTER data) captures the range of TEC values that may be encountered in the field. In addition, recent TEC data from PMIC indicate smaller TEC differences between the tube and tubesheet materials than is indicated by either of the above TEC models, thus adding to the staff's concern as whether either TEC model captures the range of TEC values which may exist in the field. A more complete technical justification for the TEC model is needed in terms of its ability to capture the range of TEC differences between the tube and tubesheet that may be encountered in the field. This technical justification should address, but not necessarily be limited to the following:

    a. Literature search for relevant TEC data, and evaluation of that data. If the PMIC data is considered not relevant, what is the basis? Apart from the American Society of Mechanical Engineers (ASME) Code vetting process, is there any reason to believe that the pedigree of the PMIC data (or ANTER data) is not as good as the data upon which the Code values are based? What is the feasibility of subjecting the PMIC and ANTER data to a review similar to that performed as part of the Code vetting process?

    b. Expert opinion and experience on the variability of TEC that may be exhibited by materials such as alloy 600 and A508 steel that may have fabricated from different heats of material, at different times, and with different processing histories (e.g., mill annealed versus thermally treated alloy 600, temperatures experienced during post weld heat relief).

    c. Expert opinion and experience on potential changes to alloy 600 TEC due to hydraulic expansion process. Data concerning sensitivity of TEC to cold working for metals in general needs to be provided.

11. Reference 4, Enclosure I, page 19 - The report states that the probability that the stacked model H*/B* will not meet the performance criteria of NEI 97-06, Rev. 2 is 2.6E-06 which is orders of magnitude less than the acceptance criteria which is a probability of 0.01. This appears to the NRC staff to be a misstatement. The staff's review of the stacked model analysis in Table 2-1 indicates that the 2.6E-06 probability number applies to each individual tube, not to the population of 22,500 tubes at Wolf Creek. Given that the H*/B* distance of each individual tube has a certain probability (e.g., 2.6E-06) of not meeting the performance criteria, it must be verified that that the probability that one or more tubes out of a population of 22,500 tubes will fail to meet the performance criteria is less than 0.01. Alternatively, an acceptance criterion applicable to the probability of not meeting the performance criteria of each individual tube must be developed consistent with the overall goal of ensuring that the probability that one or more tubes out of a population of 22,500 tubes will fail to meet the performance criteria is less than 0.01.

Revisions to the acceptance criteria approach, or an explanation why the approach in the report is correct, is needed.

12. Reference 4, Enclosure I, page 19 - The stacked uncertainty analysis in Table 2-1 considers each of six parameters be at their mean + 1 sigma value (+ or – depending on which is more conservative) with one exception, tubesheet hole/tube dimension. This parameter was considered at the mean + 2 sigma value. These mean + X sigma values were the values used in the reference H*/B* analyses shown in Tables 2-2 and 2-3 of the report. The probability that all six of these parameters could be at or outside these mean + 1 sigma values is 2.6E-06. The NRC staff is concerned that this does not provide a conservative representation of the limiting H*/B* distance. The approach assumes that the H*/B* distances are equally sensitive to all six input parameters. Independent NRC staff calculations, discussed at December 13, 2007 meeting, shows that the H*/B* distances are strongly sensitive to residual contact pressure associated with hydraulic expansion (which in turn was considered to be a function of yield strength), tube TEC, and tubesheet TEC, but to be only weakly sensitive to Young's Modulus for the tube and tubesheet, respectively, and the tubesheet hole/tube dimension. To ensure a conservative analysis, a robust "extreme value" uncertainty analysis is needed. Such an analysis should focus on those parameters which dominate the H*/B* uncertainties (e.g., residual contact pressure, tube TEC, and tubesheet TEC), both individually and in various combinations. Alternatively, a comprehensive Monte Carlo analysis of the probability density functions of all input parameters affecting H*/B* may be submitted in lieu of an extreme value analysis. [The NRC staff acknowledges that pages 24 and 25 of the report do touch on "extreme value" considerations. However, this treatment is not complete. For example, a 5 sigma variation on yield strength reducing residual pressure to essentially zero is discussed, but equally realistic cases such as 2 sigma variations on residual contact pressure, tube TEC, and tubesheet TEC, or 3 sigma variations on tube TEC and tubesheet TEC are not discussed. In addition, a robust extreme value analysis should reflect the responses to questions above dealing with residual contact pressure and tube and tubesheet TECs.]

13. Reference 4, Enclosure I, pages 19 and 20 - The report addresses the uncertainties on the degree of constraint provided by the divider plate on tubesheet vertical deflection. However, for a given divider plate assumption, the report does not address the accuracy of the finite element analyses in terms of how they affect tubesheet hole dilation, resulting increase or decrease in tube to tubesheet contact pressure, and the H*/B* distances. (For example, finite element Model O-1 is an axisymmetric model to which a divider plate factor (from 3-D model analysis) is applied. The tubesheet and its boundary conditions (including the divider plate connection) are not truly axisymmetric.) Discuss the significance of the finite element accuracy on calculated tubesheet hole dilation, resulting increase or decrease in tube to tubesheet contact pressure, and the H*/B* distances and, in light this significance, how is the conservatism of the H*/B* distances is being assured.

14. At the meeting on December 13, 2007, the licensee's contractor, Westinghouse, stated that it had discovered an error in the way the unit load finite element analyses were being combined which led to over-conservative results for steam line break. A description of the error, which was found and how it was corrected, is needed.

15. The variability of some input parameters such as tube yield strength or tube expansion pressure may vary somewhat randomly from tube to tube whereas other input parameters, particularly those associated with the tubesheet, may have a more systematic variation from a nominal value. Discuss how this affects the implementation of the uncertainty analyses being employed in support of the proposed H*/B* distances.

16. Reference 4, Enclosure I, pages 34 and 39 - As discussed at the meeting with the licensee on November 28, 2007 (Reference 5), these pages of the report and information presented at the meeting do not provide sufficient evidence that incremental slippage of the tubing within the tubesheet will not occur under heatup/cooldown and other operational loading cycles. Potential monitoring strategies were discussed at the meeting. A description of any changes to the previously proposed technical specification amendment request that define the necessary monitoring program is needed.

17. Reference 4, Enclosure I, page 40 – The report lists the accidents for which primary to secondary leakage is modeled in the Wolf Creek Update Final Safety Analysis Report (USAR). Feed line break is not included in the list, presumably because a qualitative comparison with steam line break was performed in accordance with NUREG-800, Standard Review Plan, and it determined feed line break to be less limiting from a radiological consequence standpoint. Is the previous finding in the USAR that steam line break is most limiting still valid in the context of the B* analysis and, if yes, why? If not, how will this situation be remedied? [Basis for question: The NRC staff notes that the key variables affecting the relative flow resistance in the tube to tubesheet crevice between steam line break and normal operating conditions are viscosity and the effective crevice length, since loss coefficient is now assumed to be constant with increasing positive contact pressure. While there is a significant viscosity difference between steam line break temperatures and normal operating temperatures, there is only a small viscosity difference between feed line break temperatures and normal operating temperatures. Thus, B* distances for feed line break would tend to be larger for feed line break than steam line break.]

18. Reference 4, Enclosure I, pages 66 to 71 - These pages address, in part, the conservatism of the “limiting median crevice pressure approach” for modeling the pressure within the crevice. As discussed at the meetings with the licensee on November 28 and December 13, 2007, the NRC staff has performed independent calculations using a more realistic distribution of crevice pressure as a function of elevation roughly based on the results of the Westinghouse crevice test results. The staff’s calculations indicated that the limiting median crevice pressure approach leads to non-conservative H*/B* predictions at certain radial locations including the most limiting locations. What is needed is to incorporate realistic crevice pressure assumptions, based on the test data, to verify the conservatism of the limiting median crevice pressure approach or, if not conservative at all locations, to update its H*/B* calculations to reflect realistic or conservative crevice pressure distributions. Details of the analysis should be provided.

19. At the meeting on December 13, 2007, Westinghouse identified the use of a “beta factor” which is embedded in at least some of the H*/B* analyses. As the NRC staff understands the explanation at the meeting, this beta factor has been used to adjust the crevice pressure to define an effective crevice pressure acting on the inside surfaces of

the tubesheet holes. It is the staff's understanding that the beta factor is intended to account for the fact that radial displacement of the surface of the tubesheet holes is governed not just by the crevice pressure within the tubesheet hole in question but by crevice pressures acting on nearby tubesheet holes. The staff's understanding of the "beta factor" needs to be addressed. A detailed description of how the beta factor was developed and how it is used in the development of the proposed H*/B* distances need to be provided. The NRC staff notes that the crevice pressures may vary significantly among the tubesheet holes depending on whether or not the tubes contain through wall cracks or whether or not they are completely severed. For example, the tubesheet hole in question may contain a fully severed tube. The adjacent tube holes may contain tubes with no cracks and thus with an entirely different crevice pressure than the tubesheet hole in question. Thus, the description of the development of the beta factor needs to explain how the beta factor is affected by the spectrum of possibilities regarding the crevices pressures in neighboring tube holes.

20. An updated set of Figures and Tables (similar to those in Reference 2) describing the results of the H* and B* analyses and all accompanying input parameters (including crevice pressure and beta factor assumptions) needs to be provided. This is needed for the cases (1) where the stub runner to divider plate weld is 100% degraded and (2) where the divider plate is fully effective in restricting tubesheet vertical deflection.