September 24, 2001

Mr. Otto L. Maynard
President and Chief Executive Officer
Wolf Creek Nuclear Operating Corporation
Post Office Box 411
Burlington, KS 66839

SUBJECT: WOLF CREEK GENERATING STATION - ISSUANCE OF AMENDMENT REGARDING DELETION OF LICENSE CONDITIONS AND REVISION TO STEAM GENERATOR TUBE INSPECTION TABLE 5.5.9-2 (TAC NO. MB1611)

Dear Mr. Maynard:

The Commission has issued the enclosed Amendment No. 141 to Facility Operating License No. NPF-42 for the Wolf Creek Generating Station. The amendment consists of changes to the Technical Specifications (TSs) in response to your application dated March 23, 2001, (CO 01-0013).

The amendment deletes (1) certain conditions of the operating license, and (2) reporting requirements in Table 5.5.9-2, "Steam Generator Tube Inspection," in Section 5.5.9, "Steam Generator (SG) Tube Surveillance Program," of the TSs. License Conditions 2.C.(4), 2.C.(6) through 2.C.(14), Section 2.F, and Attachments 2 and 3 to the operating license are deleted. The list of the attachments and appendices to the operating license are revised to reflect the deletion of the attachments.

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

Sincerely,

**/RA/**

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

Docket No. 50-482

Enclosures: 1. Amendment No. 141 to NPF-42
2. Safety Evaluation

cc w/encls: See next page

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President and Chief Executive Officer
Wolf Creek Nuclear Operating Corporation
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cc w/encls: See next page

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WBateman
JCalvo

[DAllison added to SE Section 2.12]
* See previous concurrence

**ACCESSION NO.: ML012680230**

| OFFICE | PDIV-2/PM | PDIV-2/LA | TSB/BC | OGC | PDIV-2/SC |
|---|---|---|---|---|---|
| NAME | JDonohew:sp | EPeyton | WBeckner* | DCummings | SDembek |
| DATE | 8/29/2001 | 7/11/01 | 08/13/2001 | 9/17/01 | 9/20/01 |

| OFFICE | IOLB SC | SRXB/BC | EMCB/BC | EMEB/BC | EEIB/BC |
|---|---|---|---|---|---|
| NAME | DTrimble* | JWermiel* | ESullivan for WBateman* | GImbro* | JCalvo* |
| SE Section | 2.2 | 2.10 | 2.5 and 2.12 | 2.5 | 2.9 |
| DATE | 07/19/2001 | 07/20/2001 | 07/30/2001 | 08/01/2001 | 08/10/2001 |

DOCUMENT NAME: C:\Program Files\Adobe\Acrobat 4.0\PDF Output\AMD-MB1611-License Conditio~.wpd
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WOLF CREEK NUCLEAR OPERATING CORPORATION

WOLF CREEK GENERATING STATION

DOCKET NO. 50-482

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 141
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 March 23, 2001, 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 Operating License and 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. 141, 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 within 60 days of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

**/RA/**

Stephen Dembek, Chief, Section 2
Project Directorate IV
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical
Specifications and Operating License

Date of Issuance: September 24, 2001

ATTACHMENT TO LICENSE AMENDMENT NO. 141

FACILITY OPERATING LICENSE NO. NPF-42

DOCKET NO. 50-482

Replace the following pages of the Operating License and the Appendix A Technical Specifications (TSs) with the attached pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change. The corresponding overleaf page for the TSs is also provided to maintain document completeness.

| REMOVE | INSERT |
|---|---|
| Operating License | Operating License |
| 4 | 4 |
| 5 | 5 |
| 6 | 6 |
| 7 | --- |
| Attachment 2 | --- |
| Attachment 3 | --- |
| Technical Specifications | Technical Specifications |
| 5.0-17 | 5.0-17 |

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, as revised through Amendment No. 141, 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) Antitrust Conditions

Kansas Gas & Electric Company and Kansas City Power & Light Company shall comply with the antitrust conditions delineated in Appendix C to this license.

(4) Environmental Qualification (Section 3.11, SSER #4, Section 3.11, SSER #5)*

Deleted per Amendment No. 141

(5) Fire Protection (Section 9.5.1, SER, Section 9.5.1.8, SSER #5)

(a) The Operating Corporation shall maintain in effect all provisions of the approved fire protection program as described in the SNUPPs Final Safety Analysis Report for the facility through Revision 17, the Wolf Creek site addendum through Revision 15, and as approved in the SER through Supplement 5, subject to provisions b and c below.

(b) The licensee may make changes to the approved fire protection program without prior approval of the Commission only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire.

(c) Deleted per Amdt. #15, dated 2-24-88.

(6) Qualification of Personnel (Section 13.1.2, SSER #5, Section 18, SSER #1)

Deleted per Amendment No. 141

*The parenthetical notation following the title of many license conditions denotes the section of the Safety Evaluation Report and/or its supplements wherein the license condition is discussed.

(7) NUREG-0737 Supplement 1 Conditions (Section 22, SER)

Deleted per Amendment No. 141

(8) Post-Fuel-Loading Initial Test Program (Section 14, SER Section 14, SSER #5)

Deleted per Amendment No. 141

(9) Inservice Inspection Program (Sections 5.2.4 and 6.6, SER)

Deleted per Amendment No. 141

(10) Emergency Planning

Deleted per Amendment No. 141

(11) Steam Generator Tube Rupture (Section 15.4.4, SSER #5)

Deleted per Amendment No. 141

(12) LOCA Reanalysis (Section 15.3.7, SSER #5)

Deleted per Amendment No. 141

(13) Generic Letter 83-28

Deleted per Amendment No. 141

(14) Surveillance of Hafnium Control Rods (Section 4.2.3.1(10), SER and SSER #2)

Deleted per Amendment No. 141

(15) Additional Conditions

The Additional Conditions contained in Appendix D, as revised through Amendment No. 123, are hereby incorporated into this license. Wolf Creek Nuclear Operating Corporation shall operate the facility in accordance with the Additional Conditions.

D. Exemptions from certain requirements of Appendix J to 10 CFR Part 50, and from a portion of the requirements of General Design Criterion 4 of Appendix A to 10 CFR Part 50, are described in the Safety Evaluation Report. These exemptions are authorized by law and will not endanger life or property or the common defense and security and are otherwise in the public interest. Therefore, these exemptions are hereby granted pursuant to 10 CFR 50.12. With the granting of these exemptions the facility will operate, to the extent authorized herein, in conformity with the application, as amended, the provisions of the Act, and the rules and regulations of the Commission.

E. The licensee shall fully implement and maintain in effect all provisions of the Commission-approved physical security, guard training and qualification, and safeguards contingency plans including amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR 27817 and 27822) and to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The plans, which contain Safeguards Information protected under 10 CFR 73.21, are entitled: "Wolf Creek Generating Station Physical Security Plan," with revisions submitted through August 2, 1988: "Wolf Creek Generating Station Security Training and Qualification Plan," with revisions submitted through August 2, 1988; and "Wolf Creek Generating Station Safeguards Contingency Plan," with revisions submitted through August 2, 1988. Changes made in accordance with 10 CFR 73.55 shall be implemented in accordance with the schedule set forth therein.

F. Deleted per Amendment No. 141

G. The licensees shall have and maintain financial protection of such type and in such amounts as the Commission shall require in accordance with Section 170 of the Atomic Energy Act of 1954, as amended, to cover public liability claims.

H. This license is effective as of the date of issuance and shall expire at Midnight on March 11, 2025.

FOR THE NUCLEAR REGULATORY COMMISSION

"Original Signed By"

Harold R. Denton, Director
Office of Nuclear Reactor Regulation

Attachments/Appendices:
1. Attachment 1 - Deleted
2. Attachment 2 - Deleted
3. Attachment 3 - Deleted
4. Appendix A - Technical
   Specifications (NUREG-1136)
5. Appendix B - Environmental
   Protection Plan
6. Appendix C - Antitrust Conditions
7. Appendix D - Additional Conditions

Date of Issuance: June 4, 1985

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 141 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 March 23, 2001, 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).

The proposed changes would (1) delete certain license conditions from the operating license, and (2) revise Table 5.5.9-2, "Steam Generator Tube Inspection," in Section 5.5.9, "Steam Generator (SG) Tube Surveillance Program," of the TSs. License Conditions 2.C.(4), 2.C.(6) through 2.C.(14) of the operating license are considered to have been completed and obsolete, or duplicate other license requirements, and are proposed to be deleted. Attachments 2 and 3 to the facility operating license are also proposed to be deleted. Section 2.F of the operating license is considered to duplicate the reporting requirements in 10 CFR 50.72 and 50.73 and is proposed to be deleted. The reporting requirements in two "Action Required" columns of TS Table 5.5.9-2 are also considered to duplicate the reporting requirements in 10 CFR 50.72 and 50.73 and are proposed to be deleted.

The list of the attachments and appendices to the facility operating license would also be revised to reflect the proposed deletion of Attachments 2 and 3.

## 2.0 EVALUATION

The licensee has proposed to delete certain conditions of the operating license and the reporting requirements in Table 5.5.9-2 of the TSs. These changes are discussed below.

### 2.1 Condition 2.C.(4) - Environmental Qualification

Condition 2.C.(4) states that: "All electrical equipment within the scope of 10 CFR 50.49 shall be qualified by November 30, 1985."

The licensee stated that it notified the NRC that the electric equipment required to be qualified under 10 CFR 50.49 had been evaluated and determined to be qualified per the provisions of 10 CFR 50.49 in its letters of November 29, 1985, and January 17, 1986. The NRC responded by letter dated March 17, 1986, stating that License Condition 2.C.(4) had been fulfilled. Based on the staff's letter of March 17, 1986, which concluded that Condition 2.C.(4) had been fulfilled, the staff concludes that Condition 2.C.(4) has been met and the proposed deletion of the condition is acceptable.

2.2 Condition 2.C.(6) - Qualification of Personnel

Condition 2.C.(6) states that: "The Operating Corporation shall have on each shift operators who meet the requirements described in Attachment 2 [of the license]." Attachment 2, which lists operating staff experience requirements, states the following:

> The Operating Corporation shall have a licensed senior operator on each shift who has had at least six months of hot operating experience on a same type plant, including at least six weeks at power levels greater than 20% of full power, and who has had startup and shutdown experience. For those shifts where such an individual is not available on the plant staff, an advisor shall be provided who has had at least four years of power plant experience, including two years of nuclear plant experience, and who has had at least four years of power plant experience, including two years of nuclear plant experience, and who has at least one year of experience on shift as a licensed senior operator at a similar type facility. Use of advisors who were licensed only at the RO (reactor operator) level will be evaluated on a case-by-case basis. Advisors shall be trained on plant procedures, technical specifications and plant systems, and shall be examined on these topics at a level sufficient to assure familiarity in the role of the advisors. These advisors, or fully trained and qualified replacements, shall be retained until the experience levels identified in the first sentence above have been achieved. The names of any replacement advisors shall be certified by the Operating Corporation prior to these individuals being placed on shift. The NRC shall be notified at least 30 days prior to the date the Operating Corporation proposes to release the advisors from further service.

The licensee stated that it notified the NRC of its intent to release the shift advisors from further service on December 20, 1985, in its letter of November 19, 1985. It stated further that TS 5.3 and Updated Safety Analysis Report (USAR) Chapter 13, "Conduct of Operations," provides the shift staffing and qualification requirements for operations personnel. The regulations in 10 CFR 50.54(m) provide the minimum requirements for on-site staffing of nuclear power plants for licensed operators and senior operators.

In its letter of November 19, 1985, the licensee stated that after December 20, 1985, it would no longer rely on advisors (or shift consultants, as stated in the letter) for further service when there is not a licensed senior operator on shift who has had at least six months of hot operating experience on a same type plant, including at least six weeks at power levels greater than 20 percent of full power, and who has had startup and shutdown experience. The licensee stated that in the future there will always be a licensed senior operator onshift with the required operating experience. Therefore, removing this license condition would increase requirements on the senior licensed operators onshift at WCGS because advisors would no longer be allowed to meet the operating experience requirements. Also, the letter notifying the NRC that the licensee would be releasing the advisors from further service was sent at least 30 days prior to the date they were released, which was required by Condition 2.C.(6). Because the license condition is no longer needed to allow advisors to assist the licensed operators and the required notification of the NRC of the change was met, the staff concludes that the proposed deletion of Condition 2.C.(6) is acceptable.

2.3 Condition 2.C.(7) - NUREG-0737 Supplement I Conditions

Condition 2.C.(7) states that: "The Operating Corporation shall complete the requirements described in Attachment 3 to the satisfaction of the NRC." These conditions reference the appropriate items in Section 22, "TMI Action Plan Requirements for Applicants for Operating Licenses," in the safety evaluation report (NUREG-0881) and Supplements 1, 2, 3, 4, and 5 of NUREG-0881. The safety evaluation report and supplements of NUREG-0881 are the safety evaluations issued by the staff that are related to the operation of WCGS and the approval of the facility operating license for the plant. Attachment 3 to the operating license specifies the following requirements:

(1) Functional and Task Analysis

Prior to startup following the first refueling outage, the Operating Corporation shall submit for staff review and approval, a description of the process used to complete the functional and task analysis, including a description and justification for all information and control deviations from the Westinghouse Owners Group Emergency Response Guidelines, Revision 1.

(2) Emergency Response Capabilities

Prior to restart following the first refueling outage, the Operating Corporation shall have a fully functional Technical Support Center and Emergency Operations Facility and a fully operable Emergency Response Facilities Information System (ERFIS).

(3) Regulatory Guide 1.97

Prior to restart following the first refueling outage, the Operating Corporation shall have installed and operable the following instrumentation:

(a) Source range instrumentation qualified to post-accident conditions

(b) Reactor vessel water level instrumentation

(c) Subcooling monitors

(d) Radiation monitors for releases from steam generator safety/relief valves or atmospheric dump valves, and

(e) Auxiliary feedwater pump turbine exhaust monitor

For the functional task analysis in (1) above, the licensee stated that the staff provided in its letter of December 2, 1986, its review of the process to complete the functional and task analysis, including a description and justification for all information and control deviations from the Westinghouse Owners Group (WOG) Emergency Response Guidelines (ERGs), Revision 1.

In its letter of December 2, 1986, the staff concluded that the licensee had satisfactorily described the process by which it conducted a function and task analysis for deriving the instrumentation and control characteristics of Revision 1 of the ERGs and related background information. Based on its review, the staff also concluded that the licensee had adopted procedural changes at WCGS that provide adequate guidance and information to the operator to cope with emergencies and achieve the pertinent objectives of the Westinghouse generic ERGs. Based on the staff's conclusions in its letter of December 2, 1986, the staff concludes that the licensee has met number (1) of Condition 2.C.(7) and the proposed deletion of (1) is acceptable.

For the emergency response capabilities in (2) above, the licensee stated that the staff in its letter of July 31, 1989, concluded that the requirements of this license condition were met. The licensee stated that this was based on the licensee's letter of December 15, 1986, and on emergency preparedness inspections and related exercises that made use of the Technical Support Center, the Emergency Operations Facility, and the ERFIS.

In its letter of July 31, 1989, the staff concluded, based on its review of the licensee's letter of December 15, 1986, and on past emergency preparedness inspections and related exercises conducted at WCGS, that number (2) of Condition 2.C.(7) has been met. Based on this conclusion by the staff, the staff concludes that the proposed deletion of number (2) of Condition 2.C.(7) is acceptable.

For Regulatory Guide 1.97 items in (3) above, the licensee stated that the staff in its letter of July 5, 1989, concluded that the licensee's response to the license condition was acceptable. The licensee stated that this was based on NRC Inspection Report 50-482/88-37 dated December 23, 1988, and the licensee's letter of December 15, 1986.

In its letter of July 5, 1989, the staff concluded, based on the licensee's letter of December 15, 1986, and Inspection Report No. 50-482/88-37 dated December 23, 1988, that the licensee's response to number (3) of Condition 2.C.(7) was acceptable. Based on this conclusion by the staff, the staff concludes that the proposed deletion of (3) of Condition 2.C.(7) is acceptable.

### 2.4 Condition 2.C.(8) - Post-Fuel Loading Initial Test Program

Condition 2.C.(8) states that: "Any changes in the Initial Test Program described in Section 14 of the FSAR [Final Safety Analysis Report] made in accordance with the provisions of 10 CFR 50.59 shall be reported in accordance with 50.59(b) within one month of such change." The FSAR for WCGS is the licensee's USAR.

The licensee stated that this license condition is obsolete because the initial test program is complete and WCGS is currently in operating cycle 12. Because WCGS is currently in operating cycle 12 and the initial test program, as described in USAR Chapter 14, is complete with the startup tests prior to the first operating cycle, the staff concludes that Condition 2.C.(8)

on the post-fuel loading initial test program has been met and the proposed deletion of the condition is acceptable.

2.5 Condition 2.C.(9) - Inservice Inspection Program

Condition 2.C.(9) states that: "By December 11, 1985, KG&E [licensee of WCGS] shall submit for staff review and approval, the inservice inspection program which conforms to the ASME [American Society of Mechanical Engineers] Code in effect on March 11, 1984."

In its application, the licensee stated that its letters of July 29 and December 11, 1985, which provided the inservice testing (IST) program and inservice inspection (ISI) program plan, respectively, for WCGS, satisfied Condition 2.C.(9).

Because the licensee provided its IST and ISI programs for WCGS in its letters of July 29 and December 11, 1985, respectively, the staff concludes that Condition 2.C.(9) on the inservice inspection program has been met and the proposed deletion of the condition is acceptable.

2.6 Condition 2.C.(10) - Emergency Planning

Condition 2.C.(10) states that: "In the event that the NRC finds that the lack of progress in completion of the procedures in the Federal Emergency Management Agency's final rule, 44 CFR Part 350, is an indication that a major substantive problem exists in achieving or maintaining an adequate state of emergency preparedness, the provisions of 10 CFR Section 50.54(s)(2) will apply."

The licensee stated that this license condition duplicates the requirements in 10 CFR 50.54(s) that are applicable to WCGS and are enforceable; therefore, deleting this license condition would not reduce any requirements on the plant.

The staff issued a letter dated May 11, 1989, to the licensee that provided the findings of the Federal Emergency Management Agency (FEMA) on 44 CFR 350 for the Kansas State and local emergency plans for WCGS. In the letter dated April 4, 1989, from FEMA that was enclosed in the staff's letter, FEMA found no problems with the Kansas State and local plans for WCGS with respect to 40 CFR 350.

The regulations in 10 CFR 50.54(s)(2)(ii) and 50.54(s)(3) state the following:

> If after April 1, 1981, the NRC finds that the state of emergency preparedness does not provide reasonable assurance that adequate protective measures can and will be taken in the event of a radiological emergency ... and if the deficiencies are not corrected within four months of that finding, the Commission will determine whether the reactor will be shut down until such deficiencies are remedied or whether other enforcement action is appropriate. [10 CFR 50.54(s)(2)(ii)]

> The NRC will base its finding on a review of the FEMA findings and determinations as to whether State and local emergency plans are adequate and capable of being implemented, and on the NRC assessment as to whether the licensee's emergency plans are adequate and capable of being implemented. [10 CFR 50.54(s)(3)]

Based on the above, the staff concludes that, because Condition 2.C.(10) duplicates the regulations in 10 CFR 50.54(s) and the proposed deletion of the license condition would not remove any requirements from the plant, the proposed deletion of the condition is acceptable.

2.7 Condition 2.C.(11) - Steam Generator Tube Rupture

Condition 2.C.(11) states that: "Prior to restart following the first refueling outage, the Operating Corporation shall submit for NRC review and approval an analysis which demonstrates that the steam generator single-tube rupture (SGTR) analysis presented in the FSAR is the most severe case with respect to the release of fission products and calculated doses. Consistent with the analytical assumptions, the licensee shall propose all necessary changes to Appendix A [the TSs] to this license."

The licensee stated that it provided a report which demonstrated that the SGTR analysis presented in the USAR is the most severe case with respect to the release of fission products and calculated doses in its letter of January 8, 1986. The licensee also submitted a license amendment request, in its letter of December 7, 1986, to incorporate the requirements in the TSs for the steam generator atmospheric relief valves to assure the availability of equipment assumed in the SGTR analysis. The TS amendment request was approved in Amendment No. 30, dated April 20, 1989. The SGTR analysis is described in USAR Section 15.6.3, and any changes to the analysis would be reviewed in accordance with the criteria in 10 CFR 50.59. The licensee concluded that the requirements of the license condition have been met, and, therefore, the license condition is obsolete and can be deleted.

In Amendment No. 30, the staff stated that the SGTR analysis that met the license condition was submitted in the licensee's letters of January 8, February 11, and April 1, 1986. The analysis takes credit for the operation of an atmospheric relief valve (ARV) to mitigate the consequences of a SGTR accident. Because the ARVs had not previously been relied upon to mitigate postulated accidents, there were no requirements in the TSs at that time relating to the operability and surveillance requirements of the ARVs. Therefore, the licensee submitted proposed TSs for the ARVs in its letter of November 7, 1986. Based on the application, as amended on March 30, 1989, TSs on the ARVs for the SGTR accident were approved.

In the safety evaluation for Amendment No. 30, the staff stated that it would address the acceptability of the SGTR analysis in future correspondence. This was addressed in the staff's letter of May 7, 1991, where the staff stated that the licensee in its letter of January 15, 1991, committed to additional demonstration runs by plant operators on the WCGS simulator to confirm operator action times in the SGTR analysis. The staff concluded that the issuance of Amendment No. 30 and the safety evaluation dated May 7, 1991, enclosed in the letter of May 7, 1991, completes the staff's action on Condition 2.C.(11) and the license condition has been satisfied. The licensee submitted the results of the additional demonstrations on operator action times in its letter of May 5, 1992, and stated that the operator action times obtained from the simulator SGTR scenarios have demonstrated that the times assumed in the analysis for WCGS are realistic and representative of the then current operator population.

Therefore, based on the staff's letter of May 7, 1991, the staff concludes that the requirements of Condition 2.C.(11) on the SGTR analysis have been met, and the proposed deletion of the condition is acceptable.

2.8 Condition 2.C.(12) - LOCA Reanalysis

Condition 2.C.(12) states that: "Prior to restart from the first refueling outage, the Operating Corporation shall submit for NRC review and approval a reanalysis for the worst break LOCA using an approved ECCS evaluation model."

The licensee stated that it provided the staff with the large break loss-of-coolant accident (LOCA) analysis for WCGS in its letter of October 1, 1986. The large break LOCA would be the worst LOCA for WCGS. The staff responded in a letter dated April 1, 1987, and stated that the results of the LOCA analysis were acceptable and the license condition had been satisfied. The licensee concluded that the requirements of the license condition have been met, and, therefore, the license condition is obsolete and should be deleted.

In its letter of April 1, 1987, the staff concluded, based on its review of the licensee's letter of October 1, 1986, that the analysis of a large break LOCA for WCGS was acceptable and Condition 2.C.(12) had been met. The staff's conclusions were based on the following: (1) the licensee's analysis was performed using methodologies and codes which have been previously approved by the NRC and which satisfy the criteria of Appendix K to 10 CFR Part 50, and (2) the results using the analysis are within the acceptance criteria of 10 CFR 50.46. Therefore, based on the staff's letter of April 1, 1987, the staff concludes that the requirements of Condition 2.C.(12) on LOCA reanalysis have been met and the proposed deletion of the condition is acceptable.

2.9 Condition 2.C.(13) - Generic Letter 83-28

Condition 2.C.(13) states that: "The Operating Corporation shall submit responses to and implement the requirements of Generic Letter 83-28 on a schedule which is consistent with that given in their February 29, 1984 and February 6, 1985 letters."

The licensee provided a summary of the responses and the NRC review of the requirements of Generic Letter (GL) 83-28. The licensee listed the following items of the generic letter and the NRC letters accepting the licensee's response to that item of the generic letter:

| **Item** | **NRC Response Accepting Item** |
|---|---|
| Item 1.1 - Post Trip Review Program Description and Procedure | NRC letter dated June 26, 1985. |

| Item | NRC Response Accepting Item |
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| Item 1.2 - Post Trip Review Data and Information Capability | NRC letter dated July 24, 1986. (Not July 14, 1986, as stated in the application). |
| Items 2.1.1 and 2.1.2 - Equipment Classification and Vendor Interface Reactor Trip System Components | NRC letter dated September 1, 1988. |
| Item 2.2.1 - Equipment Classification Other Safety-Related Components | NRC letter dated May 31, 1989. |
| Item 2.2.2 - Vendor Interface Other Safety-Related Components | NRC letter dated October 18, 1990. (Not December 18, as stated in the application). |
| Items 3.1.1, 3.1.2, and 3.1.3 - Post Maintenance Testing Reactor Trip System Components | NRC letters dated October 7 and 22, 1986. |
| Items 3.2.1, 3.2.2, and 3.2.3 - Post Maintenance Testing Other Safety-Related Components | NRC letters dated October 7 and 22, 1986. |
| Items 4.1, 4.2.1, and 4.2.2 - Reactor Trip System Reliability | NRC letter dated July 21, 1986. |
| Item 4.3 - Reactor Trip System Reliability Automatic Actuation of Shunt Trip Attachment for Westinghouse Plants | NRC Amendment No. 26 (March 1, 1989). NRC letter dated July 19, 1988. |
| Item 4.5.1 - Reactor Trip System Reliability System Functional Testing | NRC letter dated October 7, 1986. |
| Items 4.5.2 and 4.5.3 - Reactor Trip System Reliability On-line System Functional Testing | NRC letter dated May 31, 1989. |

For Item 1.1 of GL 83-28, the staff concluded in its letter of June 26, 1985, that the post-trip review program and procedures for WCGS were acceptable.

For Item 1.2 of GL 83-28, the staff concluded in its letter of July 24, 1986, that the post-trip review data and information capabilities for WCGS were acceptable.

For Items 2.1.1 and 2.1.2 of GL 83-28, the staff concluded in its letter of September 1, 1988, that the program to identify classify, and treat components that are required for the safety-related reactor trip function (Item 2.1.1) and the vendor interface program for the same components (Item 2.1.2) for WCGS were acceptable.

For Item 2.2.1 of GL 83-28, the staff concluded in its letter of May 31, 1989, that the program for identifying and classifying safety-related equipment for WCGS was acceptable.

For Item 2.2.2 of GL 83-28, the staff concluded in its letter of October 18, 1990, that the vendor interface program for safety-related components for WCGS was acceptable. The letter stated that the program would be modified to conform to the guidance provided in the GL by January 31, 1991. In a memo-to-file dated April 30, 1991, the then project manager for Wolf Creek stated that he confirmed on April 26, 1991, that the modification of the program to the GL was completed; however, the licensee has confirmed that it closed out its commitment to revise the vendor interface program on February 7, 1991, by the issuance of procedure modifications for the program.

For Items 3.1.1 and 3.1.2 of GL 83-28, the staff concluded in its letter of October 7, 1986, that the post-maintenance testing program for safety-related reactor trip system components at Wolf Creek was acceptable. This was based on letters dated February 29, 1984, and May 29, 1986, from the licensee, and NRC inspections documented in Inspection Reports 50-482/84-44 (dated January 23, 1985) and 50-482/85-11 (dated May 8, 1985).

For Items 3.2.1 and 3.2.2 of GL 83-28, the staff concluded in its letter of October 7, 1986, that the post-maintenance testing program for safety-related components other than reactor trip system components at Wolf Creek was acceptable. This was based on letters dated February 29, 1984, and May 29, 1986, from the licensee, and NRC inspections documented in Inspection Reports 50-482/84-44 (dated January 23, 1985) and 50-482/85-11 (dated May 8, 1985).

For Items 3.1.3 and 3.2.3 of GL 83-28, the staff concluded in its letter of October 22, 1986, that post maintenance testing of reactor trip system and other safety-related components for WCGS was acceptable. In the safety evaluation enclosed with the October 22, 1986, letter, the staff stated that technical specification changes addressing concerns about diesel generator testing would be submitted in response to GL 84-15 guidance. The licensee confirmed that Amendment No. 8 issued May 28, 1987, addressed the diesel generator testing in response to GL 84-15. The licensee's application for Amendment No. 8 was its letters of November 14, 1986, and April 17, 1987. Therefore, the staff concludes that the licensee has completed Items 3.1.3 and 3.2.3 of GL 83-28.

For Items 4.1, 4.2.1, and 4.2.2 of GL 83-28, the staff concluded in its letter of July 21, 1986, that the programs at WCGS on reactor trip system reliability, for (1) review of vendor-recommended reactor trip breaker modifications, (2) periodic maintenance, and (3) trending of parameters affecting operation to forecast degradation, were acceptable.

For Item 4.3 of GL 83-28, the staff concluded in its letter of July 19, 1988, that the shunt trip modification for WCGS was acceptable. In the safety evaluation enclosed with the July 19, 1988, letter, the staff stated that the licensee had proposed technical specification changes which address surveillance testing of the shunt and undervoltage trip attachments of the reactor trip breakers in response to GL 85-09, which was a followup to Item 4.3 of GL 83-28. The proposed changes were approved in Amendment No. 26 dated March 1, 1989, for WCGS to meet GL 85-09. Therefore, the staff's letter of July 19, 1988, and Amendment No. 26

completed the staff's actions for Item 4.3 of GL 83-28.

For Item 4.5.1 of GL 83-28, the staff concluded in its letter of October 7, 1986, that the online functional testing of the reactor trip system for WCGS was acceptable. This was based on letters dated February 29, 1984, and May 29, 1986, from the licensee and NRC inspections documented in Inspection Reports 50-482/84-44 (dated January 23, 1985) and 50-482/85-11 (dated May 8, 1985).

For Item 4.5.2 of GL 83-28, the staff concluded in a safety evaluation enclosed with the May 31, 1989, letter that WCGS is designed to permit on-line functional testing of the reactor trip system, including independent testing of the diverse trip features of the reactor trip breakers and, therefore, the licensee met Item 4.5.2 of GL 83-28.

For Item 4.5.3 of GL 83-28, the staff concluded in a safety evaluation enclosed with the May 31, 1989, letter that the existing intervals for on-line functional testing of the reactor trip system for WCGS are consistent with achieving high reactor trip system reliability. Based on this, the staff concludes that the licensee met Item 4.5.3 of GL 83-28.

Based on the above evaluation on how the licensee has completed the different items of GL 83-28 in Condition 2.C.(13), the staff concludes the licensee has completed Condition 2.C.(13) and the licensee's proposed deletion of the condition is acceptable.

2.10 Condition 2.C.(14) - Surveillance of Hafnium Rods

Condition 2.C.(14) states that: "The Operating Corporation shall perform a visual inspection of a sample of hafnium control rods during one of the first five refueling outages. A summary of the results of these inspections shall be submitted to the NRC." WCGS was the second Westinghouse plant to begin operation with hafnium control rods and Section 4.2.3.1(10) of the safety evaluation report that licensed WCGS stated that "the Nuclear Regulatory Commission (NRC) staff believes that a minimal surveillance program consisting of a visual inspection of representative rods should be carried out at the first two plants to have the new hafnium control rods."

The licensee stated that it submitted, in its letter of August 26, 1991, the results of the eddy current testing and visual inspections performed during the examination of the rod cluster control assemblies during refueling outage No. 3. In the attachment to its letter, the licensee stated the following:

> An amendment to WCGS Technical Specifications has been received which will allow the use of silver-indium-cadmium control rods, hafnium control rods, or a mixture of both. This change supplements [the licensee's] options for future replacement of hafnium control rods that are in operation. Pending further operating experience and control rod examination[,] hafnium will remain a viable option as a neutron adsorbing material in control rods at WCGS.

In its application, the licensee concluded that the requirements of the license condition have been met, and, therefore, the license condition is obsolete and should be deleted.

The licensee confirmed that Amendment No. 48 issued August 22, 1991, is the amendment that it referred to in its statement above from its letter of August 26, 1991. Amendment No. 48 revised Technical Specification 5.3.2, "Control Rod Assemblies," to allow the use of silver-indium-cadmium as the absorber material in the rod cluster control assemblies in addition to hafnium. The statement that "All control rod assemblies shall be hafnium, silver-indium-cadmium, or a mixture of both types" that was approved in Amendment No. 48 is now the following statement in current Technical Specification 4.2.2: "The control rod material shall be silver indium cadmium or hafnium metal as approved by the NRC."

Because the visual inspection of hafnium controls rods was submitted by the licensee in its letter of August 26, 1991, the staff concludes that the licensee has met the requirements of Condition 2.C.(14). Based on this conclusion, the staff further concludes that the proposed deletion of the condition is acceptable.

2.11 Section 2.F

Section 2.F states the following:

> Except as otherwise provided in the Technical Specifications or Environmental Protection Plan, the licensee shall report any violations of the requirements contained in Section 2.C of this license in the following manner: initial notification shall be made within 24 hours to the NRC Operations Center via the Emergency Notification System with written followup within thirty days in accordance with the procedures described in 10 CFR 50.73(b), (c) and (e).

The licensee stated that the reporting requirements in Section 2.F duplicate the requirements in 10 CFR 50.72, on immediate notification requirements for operating nuclear power reactors, and 10 CFR 50.73, on the licensee event report system. The deviations from maximum power level (Condition 2.C(1)), fire protection (Condition 2.C(5)), and additional conditions (Condition 2.C(15)) are adequately addressed by the requirements of 10 CFR 50.72 and 50.73. The condition regarding antitrust (Condition 2.C(3)) is an administrative issue and has no safety significance; therefore, it does not warrant the reporting requirements in Section 2.F. The remaining license conditions of Condition 2.C(4) and Conditions 2.C(6) through 2.C(14) are proposed to be deleted. The licensee concludes that the requirements of Section 2.F are adequately addressed by the reporting requirements of 10 CFR 50.72 and 50.73, and that Section 2.F should be deleted from the license. The licensee pointed out in its application that the staff had previously approved such a deletion in Amendment Nos. 220 and 97 for Beaver Valley Power Station, Units 1 and 2, respectively. These amendments were issued on March 26, 1999.

The staff agrees that the requirements for immediate notification with written follow-up of events at operating nuclear plants have been incorporated into the regulations in 10 CFR 50.72 and 50.73. Therefore, the staff concludes that the requirements of Section 2.F are redundant to the requirements in these regulations. Based on this conclusion, the staff further concludes that the proposed deletion of Section 2.F is acceptable.

2.12 TS Table 5.5.9-2, "Steam Generator Tube Inspection"

In TS Table 5.5.9-2, the statement that "Notification to NRC pursuant to §50.72(b)(2) of 10 CFR Part 50" is given in two columns near the bottom of the table for Category C-3 results of steam generator tube inspections. The licensee has proposed to delete both statements. In support of its proposal, the licensee stated that (1) the reference to 50.72(b)(2) is incorrect because the NRC issued a final rule on October 25, 2000, that changed the paragraph (65 FR 63769), (2) it duplicates reporting requirements in 10 CFR 50.72 of the regulations, and (3), in addition, the reporting requirement in TS Table 5.5.9-2 also duplicates TS 5.6.10.c, which requires that "Results of steam generator tube inspections, which fall into Category C-3, shall be reported in a Special Report to the Commission within 30 days and prior to resumption of plant operation."

In the current regulations, 10 CFR 50.72(b)(2) requires reporting of the declaration of an emergency class specified in the approved emergency plan, a deviation from the technical specifications pursuant to 10 CFR 50.54(x), any plant shutdown required by the technical specifications, any event that results or should have resulted in emergency core cooling system discharge as a result of a valid signal, any event that results in actuation of the reactor protection system when the reactor is critical, and any event related to the health and safety of the public or onsite personnel, or protection of environment, for which a news release is planned. Therefore, 50.72(b)(2) no longer applies to the condition in TS Table 5.5.9-2.

Prior to the rule change on October 25, 2000, 10 CFR 50.72(b) contained seven 4-hour reporting requirements, one of which applied to steam generator tubes that were found to be degraded. Paragraph 50.72(b)(2)(i) required a 4-hour notification of the NRC Operations Center of “Any event, found while the reactor is shut down, that, had it been found while the reactor was in operation, would have resulted in the nuclear power plant, including its principal safety barriers, being seriously degraded or being in an unanalyzed condition that significantly comprises plant safety.” This regulation was deleted by the rule change.

The current 50.72(b)(2)(i) requires a 4-hour notification only for “The initiation of any plant shutdown required by the plant’s Technical Specifications.” In its place, the current rule contains a reporting requirement which applies to steam generator tubes found to be degraded. Paragraph 50.72(b)(3)(ii) requires an 8-hour report for “Any event or condition that results in: (A) The condition of the nuclear power plant, including its principal safety barriers, being seriously degraded; or (B) The nuclear power plant being in an unanalyzed condition that significantly degrades plant safety.” A licensee must comply with Paragraph 50.72(b)(3)(ii) regardless of whether or not reporting of degraded steam generator tubes is required by the TSs.

In promulgating the October 25, 2000, rule change, the NRC determined that reporting events of the type discussed in the previous paragraph within 8 hours is acceptable. Therefore, the difference between the previous 4-hour report in 50.72(b)(2) and the current 8-hour report in 50.72(b)(3)(ii) is not considered important because the rule change for 50.72 approved the 8-hour reporting requirement for a nuclear power plant being in an unanalyzed condition or having principal safety barriers seriously degraded and the licensee must comply with 50.73(b)(3)(ii) even if TS Table 5.5.9-2 does not list a reference to 50.72(b). However, as stated in NUREG-1022, revision 2, reporting in accordance with 50.72(b)(3)(ii) would mean that SG tube structural or leakage integrity safety margins are exceeded. This condition is not equivalent to Category C-3, which only means that more than 1 percent of the tubes are defective and are required to be plugged. In other words, a plant's SG tubes can be in Category C-3 without exceeding structural or leakage integrity safety margins. Therefore, it is not sufficient that 10 CFR 50.72(b)(3)(ii) exists for the staff to conclude that the reporting requirements in TS Table 5.5.9-2 for Category C-3 are duplicated by the current regulations.

Although it is true that TS 5.6.10.c requires that results of steam generator tube inspections, which fall into Category C-3, shall be reported to NRC, this requirement does not duplicate the reporting requirement in TS Table 5.5.9-2 because the two reporting requirements differ as to when and how the reports should be made. The reporting requirement in Table 5.5.9-2 has an 8-hour requirement to notify the NRC by 50.72(b)(3) and the reporting requirement in TS 5.6.10.c has the requirement to submit a Special Report to the Commission within 30 days and prior to resumption of plant operation. One is to notify within 8 hours and the other is to submit a report prior to resumption of plant operation, but no later than 30 days. The SG tube inspections are conducted in plant shutdowns, typically during refueling outages.

Although TS 5.6.10.c does not duplicate the reporting requirements in TS Table 5.5.9-2, the staff concludes that the notification on a Category C-3 condition prior to restart, but no later than 30 days, is an appropriate time frame because the staff will be promptly notified per the current 50.72(b)(3)(ii) if the plant's SGs have the more significant inspection result of having seriously degraded safety barriers. Because the staff concludes that the special report in TS 5.6.10.c on the Category 3 condition requires reporting of the Category C-3 condition in an appropriate time frame, the staff also concludes that the proposed deletion of the requirement to notify the Commission pursuant to 10 CFR 50.72(b)(2) from TS Table 5.5.9-2 is acceptable. In addition, further rule changes to 50.72 will not require changes to TS Table 5.5.9-2.

## 3.0 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 no comments.

## 4.0 ENVIRONMENTAL CONSIDERATION

The amendment involves a change to a requirement with respect to the installation and 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 previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (66 FR 22035). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). This amendment also involves changes in recordkeeping, reporting or administrative procedures or requirements. Accordingly, with respect to these items, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(10). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

## 5.0 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: Jack Donohew

Date: September 24, 2001