UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

January 28, 2014

Mr. Matthew W. Sunseri
President and Chief Executive Officer
Wolf Creek Nuclear Operating Corporation
Post Office Box 411
Burlington, KS 66839

SUBJECT: WOLF CREEK GENERATING STATION – REQUEST FOR ADDITIONAL INFORMATION RE: TRANSITION TO WESTINGHOUSE CORE DESIGN AND SAFETY ANALYSIS (TAC NO. MF2574)

Dear Mr. Sunseri:

By application dated August 13, 2013, to the U.S. Nuclear Regulatory Commission (NRC) (Agencywide Documents Access and Management System (ADAMS) Accession No. ML13247A075), Wolf Creek Nuclear Operating Corporation (the licensee) requested a license amendment for Wolf Creek Generating Station, to revise the Technical Specifications to support transition to the Westinghouse core design and safety analysis. Portions of the letter dated August 13, 2013, contain proprietary information and, therefore, those portions have been withheld from public disclosure.

The NRC staff has reviewed the information provided in your application and determined that additional information is required in order to complete its review. The enclosed questions were provided to Mr. S. Wideman of your staff on January 17, 2014. Please provide a response to the questions by February 28, 2014.

The NRC staff considers that timely responses to requests for additional information help ensure sufficient time is available for staff review and contribute toward the NRC's goal of efficient and effective use of NRC staff resources. If circumstances result in the need to revise

NOTICE: Enclosure 1 to this letter contains Proprietary Information. Upon separation from Enclosure 1, this letter is DECONTROLLED.

the requested response date, please contact me at 301-415-2296 or via e-mail at Fred.Lyon@nrc.gov.

Sincerely,

CFLyon

Carl F. Lyon, Project Manager
Plant Licensing Branch IV-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-482

Enclosures:
1. Request for Additional Information (proprietary)
2. Request for Additional Information (non-proprietary)

cc w/Enclosure 2: Distribution via Listserv

ENCLOSURE 2

REQUEST FOR ADDITIONAL INFORMATION (NON-PROPRIETARY)

TRANSITION TO WESTINGHOUSE CORE DESIGN AND SAFETY ANALYSIS

WOLF CREEK NUCLEAR OPERATING CORPORATION

WOLF CREEK GENERATING STATION

DOCKET NO. 50-482

REQUEST FOR ADDITIONAL INFORMATION

TRANSITION TO WESTINGHOUSE CORE DESIGN AND SAFETY ANALYSIS

WOLF CREEK NUCLEAR OPERATING CORPORATION

WOLF CREEK GENERATING STATION

DOCKET NO. 50-482

1.0 INTRODUCTION

By letter dated August 13, 2013 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML13247A075), Wolf Creek Nuclear Operating Corporation (WCNOC, the licensee) submitted a license amendment request (LAR) to revise Safety Limits (SLs) 2.1.1, "Reactor Core SLs," Technical Specification (TS) 3.3.1, "Reactor Trip System (RTS) Instrumentation," TS 3.3.2, "Engineered Safety Feature Actuation System (ESFAS) Instrumentation," TS 3.3.5, "Loss of Power (LOP) Diesel Generator (DG) Start Instrumentation," TS 3.4.1, "RCS [Reactor Coolant System] Pressure, Temperature, and Flow Departure from Nucleate Boiling Limits," TS 3.7.1, "Main Steam Safety Valves (MSSVs)," and Specification 5.6.5, "CORE OPERATING LIMITS REPORT (COLR)," to replace the existing WCNOC methodology for performing core design, non-loss-of-coolant-accident (non-LOCA) and LOCA safety analyses (for Post-LOCA Subcriticality and Cooling only) with standard Westinghouse developed and U.S. Nuclear Regulatory Commission (NRC)-approved analysis methodologies. As part of the transition to the generic Westinghouse NRC-approved methodologies, the licensee performed instrumentation setpoint and control uncertainty calculations based on the current Westinghouse Setpoint Methodology. This amendment request also includes the adoption of Option A of Technical Specification Task Force (TSTF) TSTF-493-A, Revision 4, "Clarify Application of Setpoint Methodology for LSSS [Limiting Safety System Settings] Functions."

2.0 REGULATORY REQUIREMENTS

The NRC staff evaluated the LAR against the regulatory requirements and guidance listed below to ascertain whether there is reasonable assurance that the systems and components affected by the LAR will perform their required safety functions when called upon to do so.

2.1 Regulatory Requirements

The NRC staff considered the following regulatory requirements:

Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, "Domestic Licensing of Production and Utilization Facilities," establishes the fundamental regulatory requirements. Specifically, Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50 provides, in part, that

> ...an application for a design certification, combined license, design approval, or manufacturing license, respectively, must include the principal design criteria for a proposed facility. The principal design criteria establish the minimum necessary design, fabrication, construction, testing, and performance

> requirements for structures, systems, and components important to safety; that is, structures, systems, and components that provide reasonable assurance that the facility can be operated without undue risk to the health and safety of the public.

General Design Criterion (GDC) 13, "Instrumentation and Control," of Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50 requires that instrumentation be provided to monitor variables and systems and that controls be provided to maintain these variables and systems within prescribed operating ranges.

GDC 20, "Protection System Functions," of Appendix A to 10 CFR Part 50 requires that the protection system be designed to initiate the operation of appropriate systems to ensure that specified acceptable fuel design limits are not exceeded.

In 10 CFR 50.36, "Technical Specifications," the Commission established its regulatory requirements related to the contents of the TS. Specifically, 10 CFR 50.36 states that "each applicant for a license authorizing operation of a production or utilization facility shall include in his application proposed technical specifications in accordance with the requirements of this section." Specifically, the regulations in 10 CFR 50.36(c)(1)(ii)(A) state, in part, that

> Where a limiting safety system setting is specified for a variable on which a safety limit has been placed, the setting must be so chosen that automatic protective action will correct the abnormal situation before a safety limit is exceeded. If, during operation, it is determined that the automatic safety system does not function as required, the licensee shall take appropriate action, which may include shutting down the reactor.

Additionally, 10 CFR 50.36(c)(3) states,

> Surveillance requirements are requirements relating to test, calibration, or inspection to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions for operation will be met.

The NRC staff reviewed the proposed LAR against these requirements to ensure that there is reasonable assurance that the systems affected by the proposed LAR will perform their required safety functions.

2.2 Regulatory Guidance

Regulatory Guide (RG) 1.105, Revision 3, "Setpoints for Safety-Related Instrumentation," December 1999 (ADAMS Accession No. ML993560062), describes a method that the NRC staff finds acceptable for use in complying with the NRC's regulations for ensuring that setpoints for safety-related instrumentation are initially within, and will remain within, the TS limits. RG 1.105 endorses Part I of Instrument Society of America (ISA)-S67.04-1994, "Setpoints for Nuclear Safety Instrumentation," which is subject to NRC staff clarifications.

In Regulatory Issue Summary (RIS) 2006-17, "NRC Staff Position on the Requirements of 10 CFR 50.36, "Technical Specifications," Regarding Limiting Safety System Settings During Periodic Testing and Calibration of Instrument Channels," dated August 24, 2006 (ADAMS Accession No. ML051810077), the NRC addresses requirements on limiting safety system settings that are assessed during the periodic testing and calibration of instrumentation.

By letter dated September 7, 2005, from Patrick L. Hiland (NRC) to the Nuclear Energy Institute's Setpoint Methods Task Force, "Technical Specification for Addressing Issues Related to Setpoint Allowable Values" (ADAMS Accession No. ML052500004), footnotes are described that should be added to surveillance requirements related to setpoint verification for instrument functions on which a safety limit has been placed. This letter also addresses the information that should be included within TSs to ensure operability of the instruments following surveillance tests related to instrument setpoints.

2.3 Supplemental Guidance

The PWR [Pressurized-Water Reactor] and BWR [Boiling-Water Reactor] Owner's Groups' TSTF-493, Revision 4, dated January 5, 2011, and an errata sheet dated April 23, 2010 (ADAMS Accession No. ML100060064), addresses the NRC staff's concerns stated in RIS 2006-17, and the May 11, 2010, "Notice of Availability of the Models for Plant-Specific Adoption of Technical Specifications Task Force Traveler TSTF–493, Revision 4, 'Clarify Application of Setpoint Methodology for LSSS Functions'" (75 FR 26294), documents NRC's position on adoption of TSTF-493, Revision 4.

3.0 REQUEST FOR ADDITIONAL INFORMATION

The NRC staff has reviewed the instrumentation and controls aspects of the licensee's LAR and concludes that additional information is needed to complete the review:

EICB-RAI-1

On page 56 of WCAP-17602-P, Revision 0, "Westinghouse Setpoint Calculations for the Wolf Creek Generating Station Control, Protection, and Indication Systems," August 2013 (non-proprietary version available in ADAMS at Accession No. ML13247A079), for **[[**

]] Please provide the necessary information to demonstrate how these numbers have been established to comply with the 95/95 confidence level specified in RG 1.105. If they are based on plant drift analysis, then please provide sample calculations to demonstrate the validity of the data.

EICB-RAI-2

Please provide similar information for **[[**

]] Explain how THE same value for ALT and AFT will provide adequate detection of instrument system degradation during surveillance testing. Provide the criteria beyond which ALT and AFT are assigned different values.

EICB-RAI-3

Please provide samples of plant procedures to demonstrate that adequate measures are being implemented to ensure compliance to TSTF-493-A, Revision 4, Option A, requirements.

the requested response date, please contact me at 301-415-2296 or via e-mail at Fred.Lyon@nrc.gov.

Sincerely,

/RA/

Carl F. Lyon, Project Manager
Plant Licensing Branch IV-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-482

Enclosures
1. Request for Additional Information (proprietary)
2. Request for Additional Information (non-proprietary)

cc w/Enclosure 2: Distribution via Listserv

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ADAMS Accession Nos.: ML14027A053 (proprietary); ML14027A164 (non-proprietary); *memo dated 1/14/14

OFFICE	NRR/DORL/LPL4-1/PM	NRR/DORL/LPL4-1/LA	NRR/DE/EICB/BC*	NRR/DORL/LPL4-1/BC	NRR/DORL/LPL4-1/PM
NAME	FLyon	JBurkhardt	JThorp	MMarkley	FLyon
DATE	1/27/14	1/27/14	1/14/14	1/28/14	1/28/14

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