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

September 11, 2015

Mr. Adam C. Heflin
President, Chief Executive Officer,
and Chief Nuclear Officer
Wolf Creek Nuclear Operating Corporation
P.O. Box 411
Burlington, KS 66839

SUBJECT: WOLF CREEK GENERATING STATION - ISSUANCE OF AMENDMENT RE: REVISE THE FIRE PROTECTION PROGRAM RELATED TO ALTERNATIVE SHUTDOWN CAPABILITY (TAC NO. MF3112)

Dear Mr. Heflin:

The U.S. Nuclear Regulatory Commission (the Commission) has issued the enclosed Amendment No. 214 to Renewed Facility Operating License No. NPF-42 for the Wolf Creek Generating Station. The amendment consists of changes to the license in response to your application dated November 21, 2013, as supplemented by letters dated December 8, 2014, and January 21, 2015.

The amendment revises Paragraph 2.C.(5)(a) of the renewed facility operating license and the approved Fire Protection Program as described in the Updated Safety Analysis Report, based on the reactor coolant system thermal hydraulic response evaluation of a postulated control room fire, performed for changes to the alternative shutdown methodology.

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,

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. Amendment No. 214 to NPF-42
2. Safety Evaluation

cc w/encls: Distribution via Listserv

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

WOLF CREEK NUCLEAR OPERATING CORPORATION

WOLF CREEK GENERATING STATION

DOCKET NO. 50-482

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 214
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) Renewed Facility Operating License No. NPF-42 filed by the Wolf Creek Nuclear Operating Corporation (the Corporation), dated November 21, 2013, as supplemented by letters dated December 8, 2014, and January 21, 2015, 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.

Enclosure 1

2. Accordingly, the license is amended as follows:

Paragraph 2.C.(2) of Renewed Facility Operating License No. NPF-42 is hereby amended to read as follows:

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, as revised through Amendment No. 214, 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.

In addition, Paragraph 2.C.(5)(a) of Renewed Facility Operating License No. NPF-42 is hereby amended to read as follows:

(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, as approved in the SER through Supplement 5, Amendment No. 189, Amendment No. 191, Amendment No. 193, Amendment No. 205, and Amendment No. 214, subject to provisions b and c below.

3. The license amendment is effective as of its date of issuance and shall be implemented within 45 days of the date of issuance. In addition, the licensee shall include the revised information in the next Final Safety Analysis Report update submitted to the NRC in accordance with 10 CFR 50.71(e), as described in the licensee's application dated November 21, 2013, as supplemented by letters dated December 8, 2014, and January 21, 2015, and evaluated in the staff's safety evaluation enclosed with this amendment.

FOR THE NUCLEAR REGULATORY COMMISSION

Michael T. Markley

Michael T. Markley, Chief
Plant Licensing Branch IV-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Renewed Facility
Operating License

Date of Issuance: September 11, 2015

ATTACHMENT TO LICENSE AMENDMENT NO. 214

RENEWED FACILITY OPERATING LICENSE NO. NPF-42

DOCKET NO. 50-482

Replace the following pages of the Renewed Facility Operating License No. NPF-42 with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Renewed Facility Operating License

REMOVE	INSERT
4	4
5	5

(5) The Operating Corporation, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess, and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components; and

(6) The Operating Corporation, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

C. This renewed operating license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission, now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

The Operating Corporation is authorized to operate the facility at reactor core power levels not in excess of 3565 megawatts thermal (100% power) in accordance with the conditions specified herein.

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, as revised through Amendment No. 214, 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.

---

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

(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, as approved in the SER through Supplement 5, Amendment 189, Amendment No. 191, Amendment No. 193, Amendment No. 205, and Amendment No. 214, 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.

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

Deleted per Amendment No. 141.

(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.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 214 TO

RENEWED FACILITY OPERATING LICENSE NO. NPF-42

WOLF CREEK NUCLEAR OPERATING CORPORATION

WOLF CREEK GENERATING STATION

DOCKET NO. 50-482

1.0 INTRODUCTION

By application November 21, 2013 (Reference 1), as supplemented by letters dated December 8, 2014, and January 21, 2015 (References 2 and 3, respectively), Wolf Creek Nuclear Operating Corporation (WCNOC, the licensee) requested changes to the license for Wolf Creek Generating Station (WCGS). The supplemental letters dated December 8, 2014, and January 21, 2015, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the U.S. Nuclear Regulatory Commission (NRC) staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on March 18, 2014 (79 FR 15151).

The proposed changes would revise Paragraph 2.C.(5)(a) of the renewed facility operating license and the fire protection program as described in the Updated Safety Analysis Report (USAR), based on the reactor coolant system (RCS) thermal hydraulic response evaluation of a postulated control room fire, performed for changes to the alternative shutdown methodology. Specifically, the changes would involve including credit for an automatic feedwater isolation signal (FWIS) to close; and the main feedwater isolation valves (MFIVs), or the main feedwater regulating valves (MFRVs) and the MFRV bypass valves, to terminate main feedwater flow in the event of a control room fire and if control room operators are unsuccessful in manually isolating the feedwater system.

The licensee identified required changes to the approved fire protection program as described in the USAR as follows:

1. Revision to USAR Appendix 9.5B, "Fire Hazards Analysis," to include incorporation of Drawing E-1F9915 Revision 5, as the licensing basis document for alternative shutdown following a control room fire in lieu of the licensee's letter SLNRC 84-0109 dated August 23, 1984 (Reference 4).

2. Revision to Calculation XX-E-013, Revision 3, "Post-Fire Safe Shutdown (PFSSD) Analysis," Assumption 3-A-4 regarding application of loss of automatic

functions, specific to automatic feedwater isolation in the event of a control room fire. Calculation XX-E-013 is incorporated by reference in USAR Appendix 9.5B.

3. Deviation from the licensee's commitment (see Regulatory Evaluation discussion for an explanation of the commitment) to Title 10 of the *Code of Federal Regulations* (10 CFR), Part 50, Appendix R, "Fire Protection Program for Nuclear Power Facilities Operating Prior to January 1, 1979," Section III.L.1, "Alternative and Dedicated Shutdown Capability," comparison response, as described in Appendix 9.5E, "WCGS Fire Protection Comparison to 10 CFR 50 Appendix R," of the WCGS USAR, specific to maintaining RCS process variables within those predicted for a loss of normal alternating current (AC) power.

4. Deviation from the licensee's commitment to 10 CFR Part 50, Appendix R, Section III.L.2 comparison response, as described in Appendix 9.5E of the WCGS USAR, specific to maintaining pressurizer level on scale.

WCNOC determined that the proposed changes adversely affect the ability to achieve and maintain safe shutdown in the event of a fire. Therefore, prior NRC approval is required, since the licensee's commitments to 10 CFR Part 50, Appendix R, Section III.L.1 and Section III.L.2 are not satisfied.

Additionally, the licensee proposed to add a reference to Amendment No. 189 in License Condition 2.C.(5)(a). Amendment No. 189 was issued on September 30, 2010 (Reference 5), and revised the WCGS fire protection program for the use of fire-resistive cable for certain power and control cables with two motor-operated valves on the component cooling water system. Subsequent to Amendment No. 189, amendments that approved changes to WCGS fire protection program were included in License Condition 2.C.(5)(a). This is an administrative change to the license condition to consistently reflect license amendments that are applicable to the fire protection program.

2.0 REGULATORY EVALUATION

WCGS was licensed after January 1, 1979, and therefore is not required by 10 CFR 50.48(b) to meet 10 CFR Part 50, Appendix R. Although Appendix R does not apply, WCGS's commitment to Appendix R, Section III.L is established in Appendix 9.5E as part of the approved fire protection program documented in the USAR. Appendix R, Section III.L.3, requires that alternative safe shutdown capability shall be independent of the fire area.

The requested amendment involves reliance on an automatic signal within the affected fire area, contrary to 10 CFR 50, Appendix R, Section III.L. The requested amendment addresses a postulated fire within one of the cabinets in the control room. If the fire is not detected in time, the fire could cause loss of automatic function of valves and pumps with control circuits that could be affected by a control room fire. The scenario assumes that the fire causes evacuation of the control room and operators enter alternative shutdown procedure OFN RP-017 Revision 42. Before the operators evacuate the control room, they will trip the reactor using the hand switches located on panels RL003 and RL006. The operators will also attempt to close the main steam isolation valves (MSIVs) by using the hand switches that are located on panel RL006. However, in accordance with the current licensing basis, credit is not given for

manual MSIV closure since spurious actuation could occur and prevent closure as a result of the control room fire. The licensee has determined that there is insufficient time for operators to take actions outside of the control room to stop the main feedwater pumps from overfilling the steam generators (SGs).

The technical evaluation in Section 3.0 of this safety evaluation (SE) addresses the tools in place to prevent, detect, and suppress fires. In the event that a fire within the control room is not prevented, and is not suppressed, that necessitates control room evacuation. The licensee has provided analysis to show that an unrecoverable condition will not occur due to an automatic FWIS for closure of the main feedwater isolation valves and/or the main feedwater regulating valves.

The licensee requested this amendment in accordance with 10 CFR 50.90, which allows licensees to request a change to their license. The licensee's submittal fully describes the change desired. The NRC staff reviewed this license amendment request (LAR) based on the information in 10 CFR Part 50, Appendix R to determine if the changes provide adequate protection of public health and safety.

NRC staff guidance for the postulated control room fire scenario and the associated acceptance criteria are given in NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, LWR Edition" (SRP) Section 9.5.1.1, "Fire Protection Program," Revision 0, February 2009 (Reference 6). The guidance in Regulatory Guide 1.189, Revision 2, "Fire Protection for Nuclear Power Plants," October 2009 (Reference 7), provides an approach for satisfying the above criteria that is acceptable to the NRC staff.

3.0 TECHNICAL EVALUATION

3.1 NRC Staff Evaluation of the Prevention, Detection, and Suppression of Fires

In its submittal, the licensee provided an evaluation for crediting the automatic FWIS in the event of a control room fire and subsequent evacuation. A spurious actuation due to a cabinet fire in the control room has the potential to defeat an operator action (performed in the control room) to close the main steam isolation valves (MSIVs). The licensee's evaluation intended to show that even in the event of the failure of the operator actions to close the MSIVs, the automatic functions controlled by the FWIS are sufficiently separated from the postulated fire that the automatic functions would not be impacted. The licensee requested a license amendment because reliance on automatic actions is not consistent with the plant's licensing basis.

3.1.1 Background

The licensee's letter SLNRC 84-0109 dated August 23, 1984 (Reference 4), described the original strategy for shutting down the plant and maintaining a safe hot standby condition in order to meet the Appendix R Section III.L criteria in the event of a fire in the control room that requires evacuation. Letter SLNRC 84-0109 is part of the approved fire protection program and is described in Section 9.5.1.5 of NUREG-0881, Supplement 5, "Safety Evaluation Report

Related to the Operation of the Wolf Creek Generating Station Unit No. 1," March 1985 (Reference 8).

The NRC issued an apparent violation on February 1, 2006, during an NRC triennial fire protection inspection (Reference 9) that resulted in changes to the WCGS's shutdown methodology for fires that are postulated to involve a control room evacuation. In order to demonstrate the adequacy of these revised shutdown procedures, the licensee developed Drawing E-1F9915, "Design Basis Document for OFN RP-017, Control Room Evacuation," Revision 0, and Evaluation SA-08-006, "RETRAN-3D Post-Fire Safe Shutdown (PFSSD) Consequence Evaluation for a Postulated Control Room Fire," Revision 3 (Reference 10).

During a subsequent NRC triennial fire protection inspection (see NRC letter to the licensee dated January 2, 2009; Reference 11) the inspection team identified an unresolved item related to these changes to the fire protection program: URI 0500482/2008010-03, "Changes to the Approved Fire Protection Program May Not Meet NRC Acceptance Criteria." The inspection team stated that the licensee changed the fire protection program in a manner that could adversely affect the ability to achieve and maintain safe shutdown in the event of a fire. In accordance with the WCGS license condition 2.C.(5)(b), such changes require prior NRC approval.

WCGS has determined that the change to credit an automatic FWIS in the event of a control room fire and evacuation represented an adverse effect on the ability to achieve and maintain safe shutdown in the event of a fire. Therefore, prior NRC approval is required as WCGS commitment to 10 CFR 50 Appendix R, Section III.L.1 and Section III.L.2, is not satisfied. In addition, crediting automatic action violated assumption 2.2.5, in Section 2.2 of Drawing E-1F9915, enclosure to the LAR dated November 21, 2013, which states,

> Automatic functions capable of mitigating spurious actuations are assumed to be defeated by damage to cables located in the area associated with the automatic function.

3.1.2 Defense-in-Depth Review

The licensee stated that the ability to achieve and maintain safe shutdown is preserved following a fire event by extending the concept of defense-in-depth to:

1. Prevent fires from starting;

2. Detect rapidly, control, and extinguish promptly those fires that do occur; and,

3. Provide protection of structures, systems, and components important to safety so that a fire that is not promptly extinguished by fire suppression activities will not prevent safe shutdown of the plant.

3.1.2.1 Fire Prevention Features

The licensee has stated that except under strictly controlled conditions, hot work activities are not permitted in the control room during power operation. The licensee has administrative

controls in place to minimize the introduction of transient combustibles within the control room. Cables in the control room are limited to those that terminate in the control room for instrumentation and control circuits as well as lighting and other ancillary uses. In the original submittal dated November 21, 2013, the licensee stated that cables carrying voltages greater than 120 Volts alternating current (VAC)/125 Volts direct current (VDC) are not run in the control room. In a supplement dated December 8, 2014, the licensee stated that during a review of external operating experience, it was identified that there are cables carrying voltages of 250 VDC in the control room. These cables are Institute of Electrical and Electronics Engineers (IEEE) 383, "Standard for Type Test of Class 1E Electric Cables, Field Splices, and Connections for Nuclear Power Generating Stations" (Reference 12) qualified, which is an NRC accepted standard for cables. The presence of the 250 VDC cables in the control room does not adversely impact the conclusion that the fire protection defense-in-depth features within the control room provide reasonable assurance that a severe fire that causes the evacuation of the control room is unlikely.

The carpet material used in the control room meets or exceeds the surface flammability criteria per American Society for Testing and Materials (ASTM) E-84, "Standard Test Method for Surface Burning Characteristics of Building Materials" (Reference 13), or Consumer Product Safety Commission Standard FFI-70, "Standard for the Surface Flammability of Carpets and Rugs" (Reference 14); the static propensity rating per ASTM D-2679, 'Standard Test Method for Electrostatic Charge" (Reference 15), or American Association of Textile Chemists and Colorists (AATCC) 134, "Electrostatic Propensity of Carpets" (Reference 16); smoke development rating per ASTM E-662, "Standard Test Method for Specific Optical Density of Smoke Generated by Solid Materials" (Reference 17), and the critical radiant flux rating per ASTM E-648, "Standard Test Method for Critical Radiant Flux of Floor Covering Systems Using a Radiant Heat Energy Source" (Reference 18), or National Fire Protection Association (NFPA) 253, "Standard Method of Test for Critical Radiant Flux of Floor Covering Systems Using a Radiant Heat Energy Source" (Reference 19).

3.1.2.2 Detection and Extinguishing Features

The licensee states that the control room is continuously manned; therefore operators should be able to quickly respond to a fire event in the control room. Additionally, fixed spot type ionization smoke detectors installed per NFPA 72D, "Standard for the Installation, Maintenance and Use of Proprietary Protective Signaling Systems" (Reference 20), are provided within some control room cabinets and at the ceiling level throughout the control room. There are duct smoke detectors installed on the control room back panel area return ductwork. A Halon 1301 system installed per NFPA 12A, "Halogenated Extinguishing Agent Systems – Halon 1301" (Reference 21), that is actuated by fixed spot-type ionization smoke detectors is provided inside the cable trenches beneath the control room floor. Portable fire extinguishers installed per NFPA 10, "Installation of Portable Fire Extinguishers" (Reference 22), and hose stations installed per NFPA 14, "Standpipe and Hose Systems" (Reference 23), are also provided.

3.2 NRC Staff Evaluation of Postulated Control Room Fire Scenarios

In the event of a control room fire that causes the operators to abandon the control room, WCNOC is crediting the automatic FWIS to terminate main feedwater flow and prevent SG overfill. In the LAR, the licensee provided an analysis to justify the use of the automatic FWIS.

The justification includes an analysis of the availability of the automatic FWIS based on the separation of cables, equipment, and electrical cabinets within the control room such that one set of equipment needed for shut down will be available. The licensee also provided information from electrical cabinet fire testing to support its analysis that should a fire start within one of the cabinets within the control room, there will still be adequate separation such that one set of equipment needed for shut down will be available.

The licensee evaluated the bounding scenarios that may occur in the event of a fire in the control room. One scenario includes the inability of the control room operators to credit the closure of the MSIVs using the hand switches in the control room. Closing the MSIVs terminates main steam flow to the feedwater pump turbines thereby stopping the main feedwater pumps. However, if the main feedwater pumps continue to operate, the SGs would overfill in a matter of minutes and operators outside of the control room would not have sufficient time to stop the pumps. Therefore, WCNOC is proposing a revision to credit the automatic FWIS for closure of the MFIVs and/or the main feedwater regulating valves (MFRVs), and the MFRV bypass valves to terminate main feedwater flow and prevent SG overfill.

The NRC staff evaluated postulated control room fire scenarios regarding:

A. Separation of critical cabinets and separation of Train A and Train B cables as they relate to the LAR analysis of loss of automatic functions, specific to automatic feedwater isolation in the event of a control room fire.

B. Adequacy of the assessment of spurious and multiple spurious operations as related to the justification of the request to revise the USAR Appendix 9.5B by the incorporation of Drawing E-1F9915 as the licensing basis document for alternative shutdown following a control room fire.

C. The adequacy of the selection of sequences for the PFSSD consequence evaluation and adequacy of the thermo-hydraulic calculation in support of the PFSSD consequence evaluation, specifically the use and implementation of the RETRAN-3D analysis to justify the request to: deviate from 10 CFR 50, Appendix R, Section III.L.2, specific to maintaining pressurizer level on scale; and to deviate from 10 CFR 50, Appendix R, Section III.L.1, specific to maintaining RCS process variables within those predicted for a loss of normal AC power.

A. Separation of critical cabinets and separation of Train A and Train B cables as they relate to the LAR analysis of loss of automatic functions, specific to automatic feedwater isolation in the event of a control room fire.

The licensee used historical fire testing information to show that a control room fire would not impact either of the manual switches available to the operator or the automatic FWIS cabinets. The decision to manually trip the plant will likely be made early in the event as environmental and plant conditions warrant. The licensee assumes that the FWIS actuates 15 seconds after reactor trip (Section 3.7.3 of Attachment I of the application). The information below supports why a fire in the control room would be unlikely to spread and damage the automatic signal

capability, via fire or smoke damage, in the 15 seconds that it would take for the FWIS to be generated.

The licensee provided information from NUREG/CR-4527, "An Experimental Investigation of Internally Ignited Fires in Nuclear Power Plant Control Cabinets: Part 1: Cabinet Effects Tests," April 1987 (Reference 24), to show that fires in vertical control cabinets are not expected to spread to adjacent cabinets where IEEE-383 qualified and unqualified cable is used and where the cabinets are separated only by the metal enclosure of each cabinet forming a double wall metal barrier. In these tests, the cabinets were placed side-by-side leaving only a 1-inch air gap between cabinets. The licensee stated that cabinets tested as part of NUREG/CR-4527 are representative of the control cabinets used at WCGS. Also, the configuration of the cabinets at WCGS is such that there is at least a 1-inch air gap and a double metal barrier between cabinets.

The licensee also provided information from NUREG/CR-4596, "Screening Tests of Representative Nuclear Power Plants Components Exposed to Secondary Environments Created by Fires," June 1986 (Reference 25), to demonstrate that smoke conditions would not affect redundant trains of the FWIS. The primary objective of NUREG/CR-4596 was to assess the functionality of representative nuclear power plant components when subjected to smoke and hot gases created by fire. The test results showed that most components survived the environments created by the cabinet fires and that higher ventilation rates and a larger room size resulted in significantly less combustion products being deposited on the components. The ceiling height in the equipment cabinet area of the control room is approximately 25 feet above the floor.

The air conditioning system in the control room and equipment cabinet area is not equipped to shut down in the event of smoke entering the ductwork. During normal operation the return air flow is approximately 22,650 cubic feet per minute (cfm) from the control room envelope. The volume of the control room envelope is conservatively estimated at 156,000 cubic feet, with a ceiling height of 25 feet. There are approximately eight air changes per hour. Since the control room air conditioning system does not automatically shut down in the event of smoke in the ductwork, the system will continue to operate to remove products of combustion from the room until the ventilation system is shut down. The decision to trip the plant will likely be made early in the event as environmental and plant conditions warrant. Since the FWIS occurs automatically within 15 seconds of a plant trip, it is not likely that smoke conditions would be sufficient to affect the automatic FWIS.

The NRC staff reviewed critical cable separation using NUREG/CR-4527, "An Experimental Investigation of Internally Ignited Fires in Nuclear Power Plant Control Cabinets," Volume 1, April 1987, and Volume 2, November 1988, and determined that the separation between protection cabinets SB038 and SB037 cannot be compared to the criteria established in NUREG/CR-4527. In order to evaluate the actual cabinet separation, the, NRC staff requested that the licensee explain how the electrical cabinet separation criteria in NUREG/CR-4527 for adjacent cabinets compared to the main control room cabinet layout at the site.

In its letter dated January 21, 2015, the licensee clarified (SRXB-RAI-4) that its assumption is that the fire will not spread to adjacent cabinets containing opposite train circuits. The licensee does not propose to take credit for cabling in adjacent cabinets, as is implied in the licensee's

submittal. The implication of the licensee's submittal was that the physical separation criteria specified in NUREG/CR-4527 were satisfied, because "The cabinets used in the fire tests are representative of the control cabinets used at WCGS" (from Section 3.6.5 of Attachment I to the licensee's submittal dated November 21, 2013). Specifically, the licensee stated that protection set cabinets SB037 and SB038 both contain only Train A components and circuits. Further discussion with the licensee during an on-site audit conducted by NRC staff and its contractor, Sandia National Laboratory (SNL) (ADAMS Accession No. ML15069A646; not publically available, contains proprietary information) confirmed that their analysis assumes both of these cabinets, located in the same cabinet bank, would be consumed by a fire, as they are not sufficiently separated. However, the physical separation requirement between cabinets is instead met by a 4-foot wide walkway between cabinet banks. The Train B cabinets, SB041 and SB042, are located across the 4-foot walkway, and it is presumed that the fire will not spread to these cabinets. This physical separation was verified by the NRC staff during a walkdown of the main control room during the audit. During the audit visit, the staff and the licensee discussed the protection priority status of Train A versus Train B, and the licensee clarified that Train B is the protected train as established in the initial licensing application.

The NRC staff verified information relative to the control room layout document (J-14001, Attachment V (not publically available, contains security-related information) to the licensee's application dated November 21, 2013), referencing the conduit path through the control room from protection cabinet SB042 to solid state protection system (SSPS) cabinet SB029A and power supply cables for cabinets NN0110 and NN0309. The power supply cables to NN0110 and NN0309 are routed from the lower cable spreading room directly up into the cabinet and are not exposed in the control room.

The NRC staff also verified the protection of cables and cable runs for each safety train where it is exposed in the control room. During the audit, the NRC inspected 6 areas identified in the licensee's application dated November 21, 2013, Sections 3.6.2, 3.6.3, and 3.6.4, where power or control cables associated with the FWIS were routed through the control room. The protection of these raceways was verified by NRC and SNL staff during the audit as follows:

a. Conduit carrying Train B low-low input signals from electric chase near columns C-5 and C-7 to SB042. The NRC staff and SNL visually verified that the cables are contained in conduit labeled 2J1B1A and that the conduit is fully sealed throughout its length and passes over three banks of cabinets, which do not carry safety-related circuits associated with Train A or B.

b. Conduit carrying Train B temperature inputs associated with Loop 2 (BBTE0421B, BBTE0421A1, BBTE0421A2, and BBTE0421A3) from the electric chase near columns C-5 and C-7 to SB042. The NRC staff and SNL visually verified that the cables are contained in conduit labeled 2J1070 and that the conduit is fully sealed throughout its length and passes over three banks of cabinets, which do not carry safety-related circuits associated with Train A or B.

c. Cable tray and conduit carrying Train A and Train B signals from SB041 and SB042 to the SSPS cabinets. The NRC staff and SNL verified that conduit 2J3019 exits from the top of SB042, proceeds vertically for approximately 6 feet, turns to the west until it is approximately 4 feet west of the Train A Protection

cabinet SB038, turns to the SB29A where it enters that cabinet. This conduit is sealed for its entire length. Conduit 2J3020 exits from the top of SB042, proceeds vertically approximately 10 feet, leads west past the end of the cabinet bank row, turns north and proceeds to SB032A where it drops into this cabinet. This conduit is sealed for the entire length. Cable tray 4J1J17 exits from the top of SB041 and proceeds vertically directly into the upper cable spreading room. This cable tray is fully enclosed with metal panels on all sides and fire barrier penetration sealant installed on all cracks and edges, except for the final 4 feet before the cable tray passes through the ceiling. In these final few feet, the cable tray is not enclosed by metal covers and the cables are visible from the floor level. The cables from SB041 then re-enter the Control Room from the upper spreading room to SB029A via cable tray 4J1J35, which is fully enclosed and sealed. Also, cables from SB041 re-enter the control room from the upper spreading room to SB32A via cable tray 4J1J19, are fully enclosed and sealed.

d. The NRC staff and SNL verified that conduit carrying cables associated with Loops 2 and 4 (Train B) temperature sensor signals to SB041 and SB042 are carried in conduit labeled 4J1J17, discussed above, and were fully sealed.

e. Cables carrying reactor trip signals from SB102B (located in the plant) to SB032B are carried in conduit labeled 4J3016 and are sealed along the entire length.

f. Power cables for NN0410 are located in a combination of fully sealed conduit and fully enclosed and sealed cable tray 4C8J1G.

Based on its review and audit as described above, the NRC staff concluded that the licensee adequately provides fire separation between manual and automatic functions of FWIS.

B. Adequacy of the assessment of single and multiple spurious operations as related to the justification of the request to revise the USAR Appendix 9.5B by the incorporation of Drawing E-1F9915 as the licensing basis document for alternative shutdown following a control room fire.

The NRC staff reviewed the licensee's selection of fire sequences provided in the application dated November 21, 2013, Attachment 1, Section 2.4, and Drawing E-1F9915, Section 2.2, "Assumptions." Attachment 1, Section 2.4 states, in part, that:

> The PFSSD analysis assumes a loss of off-site power in some scenarios coupled with one spurious operation (failure) and no automatic actuation of safety components. These two additional conservative assumptions ensure the transient will be more severe than a loss of normal AC power.

Drawing E-1F9915, Section 2.2, Assumption 2.2.3 states:

> Prior to transfer of control to the Auxiliary Shutdown System only a single spurious actuation is assumed to occur at a time, except in the case of two redundant valves in a high/low pressure interface line. All potential spurious

actuations are mitigated/prevented using OFN RP-017 but timing is based on the spurious actuations occurring one at a time, or two at a time in the case of high/low pressure interface lines.

Based on its review, the NRC staff requested additional information regarding whether or not the licensee evaluated the potential for multiple spurious operations (MSOs) upon the safe-shut down path after control of the plant has been transferred from the control room to the alternative shutdown system, as discussed in RG 1.189, Revision 2, Sections 5.4.1 and 5.4.4. The staff also requested that the licensee describe how it identified and considered the potential impact of spurious actuations that could defeat the alternate safe shutdown system, as discussed in RG 1.189, Revision 2, and Section 5.4.1.

In its letter dated January 21, 2015, the licensee clarified that the assessment of MSOs occurring after control of the plant has been achieved were considered as documented in Drawing E-1F9915, Appendix 3 of the licensee's application dated November 21, 2013.

In its letter dated January 21, 2015, the licensee stated that the fire protection program assumes that the timing of spurious actuations is based on single or multiple spurious operations (MSOs) occurring after control of the plant is achieved from the alternative location. Prior to transfer of control to the alternative shutdown system, only a single spurious operation is assumed. This is consistent with RG 1.189, Section 5.4.4 (Reference 7). The licensee proposed to revise Drawing E-1F9915 Assumption 2.2.3 as follows:

2.2.3 Before transfer of control is achieved by the alternative and dedicated shutdown system only a single spurious actuation is assumed to occur, except in the case of two redundant valves in a high/low pressure interface line. All potential spurious actuations are mitigated or prevented using procedure OFN RP-017 but timing is based on one spurious actuation occurring prior to transfer of control to the alternative and dedicated shutdown system, or two spurious actuations in the case of high/low pressure interface lines.

The NRC staff reviewed the licensee's response and concludes it is consistent with the guidance of RG 1.189, Revision 2.

During the audit visit, the NRC staff reviewed the licensee's assessment of MSOs in each of the accident scenarios, consistent with the criteria in RG 1.189, Revision 2, Sections 5.4.1 and 5.4.4, which specify that such actuations should be considered after control has been transferred from the control room to the alternative or dedicated shutdown system and after control of the plant has been achieved.

Appendix 3 of Drawing E-1F9915 records the licensee's assessment of MSOs consistent with the guidance in RG 1.189, Revision 2, Sections 5.4.1 and 5.4.4, by following the process documented in Nuclear Energy Institute (NEI) 00-01, Revision 2, "Guidance for Post Fire Safe Shutdown Circuit Analysis," May 2009, Appendices D and G (Reference 28). The NRC staff verified that the MSO scenarios documented in Appendix 3 of Drawing E-1F9915 correspond to the guidance of NEI 00-01, Revision 2, Appendix G.

Based on the NRC staff's review of Drawing E-1F9915, Appendix 3, the information provided by the licensee in its letter dated January 21, 2015, and its on-site audit, the staff concludes that the licensee's treatment of single spurious operations and MSOs subsequent to establishing control of the plant, as documented in Appendix 3 of Drawing E-1F9915, adequately addresses all of the pressurized-water reactor generic MSO scenarios identified in Appendix G of NEI 00-01, Revision 3, "Guidance For Post Fire Safe Shutdown Circuit Analysis," October 2011 (Reference 29), as discussed below:

a. In its letter dated January 21, 2015, the licensee clarified that the assessment of spurious actuations that could defeat the alternate safety shutdown systems was documented in Drawing E-1F9915, Table 7.1. During the audit, the NRC staff verified the licensee's assumption that the Feedwater Chemical Injection system does not need further assessment regarding MSOs or as a potential source for water to flood up the SGs, as discussed in the licensee's application dated November 21, 2013, Attachment I, Sections 3.6.1 and 3.6.4.

b. The NRC staff verified the licensee's assessment of two potential leakage paths from the RCS, the RHR pump suction valves and the number 1 RCP seal return valves (BBHV8141A, B, C and D).

c. The NRC staff verified the licensee's assessment of the redundancy of the process monitoring function as discussed in Attachment I, Section 3.6.2, "Evaluation of SG Water Level Low-Low Feedwater Isolation Input Signals."

The licensee provided clarifying information to the NRC staff during the on-site audit regarding the assessment of spurious actuations in drawing E-1F9915, Table 7.1. Specifically, the assumption that the Feedwater Chemical Injection system did not need further assessment regarding MSOs or as a potential source for water to flood up the SGs was discussed. FWIS isolates the Feedwater Chemical Injection system, but the isolation actuation is not necessary because of the system is isolated by a normally closed manual valve. The NRC staff verified the system configuration by reviewing drawing P&ID 12AE02. Therefore, the staff concludes that the Feedwater Chemical Injection system did not need further assessment regarding MSOs or as a potential source for water to flood up the SGs.

The NRC staff also reviewed information relative to reactor coolant makeup and inventory control discussed in the application dated November 21, 2013, Attachment 1, Section 6.3. The licensee identified two potential sources for loss of inventory in Section 6.3:

- The RHR pump suction valves, which are normally closed and in a normally de-energized state, and therefore this path is eliminated as a potential leakage path. The NRC staff verified the system configuration by reviewing drawings P&ID 12EJ01 and 12AE02.

- The potential exists for the No. 1 reactor coolant pump (RCP) seal return valves (BBHV8141A, B, C, D) to spuriously fail closed, which may place the No. 2 RCP seals beyond their design basis pressure at high temperature. The licensee clarified that the potentially harmful impacts of spurious closure of the No. 1 RCP seal return valves are mitigated by steps in procedure OFN-RP-017 to trip the

> RCPs (Step B1) and isolate charging through the pump seals (Step B11), thus reducing the flow rate through each seal to 21 gpm. The licensee stated that this flow rate is well within the tolerance of the No. 2 seals even if the No. 1 seal return valves should spuriously close. Based on engineering judgement, the NRC staff concludes that the procedure provides reasonable assurance that the impact of a spurious closure of the No. 1 RCP seal return valves will be adequately mitigated.

Based on the NRC staff's review of E-1F9915, Table 7.1, the information provided by the licensee in its letter dated January 21, 2015, and the staff's on-site audit, the staff concludes that the licensee satisfactorily addressed the guidance of RG 1.189, Revision 2, Section 5.4.1 regarding the assessment of single and multiple spurious operations.

C. The adequacy of the selection of sequences for the post-fire safe shutdown (PFSSD) consequence evaluation and adequacy of the thermo-hydraulic calculation in support of the PFSSD consequence evaluation, specifically the use and implementation of the RETRAN-3D analysis to justify the request to: deviate from 10 CFR Part 50, Appendix R, Section III.L.2, specific to maintaining pressurizer level on scale; to deviate from 10 CFR Part 50, Appendix R, Section III.L.1, specific to maintaining RCS process variables within those predicted for a loss of normal AC power.

The NRC staff evaluation of the adequacy of the PFSSD is divided into three parts: (C1) the bounding scenarios, (C2) the use of the RETRAN-3D systems model, Chexal-Lellouche correlation, and VIPER sub-channel model, and (C3) the results of the consequence analysis.

C1. The Adequacy of the Bounding Scenarios

The NRC staff reviewed the licensee's selection of bounding accident scenarios for the PFSSD consequence evaluation as provided in the application dated November 21, 2013, Attachment 1, Section 3.7, and concluded that the licensee's approach followed accepted industry practice. The licensee evaluated a range of plausible accident scenarios on the licensee's simulator to identify and select those scenarios that would result in maximum RCS mass inventory loss and pressure reduction following an uncontrolled cool down of the primary system. This satisfies the intent of RG 1.189, Regulatory Position 1.8.2 to establish a sound technical basis for the current license amendment request.

C2. The adequacy of the RETRAN-3D systems model, Chexal-Lellouche drift flux model, and VIPRE-01 sub-channel model

In its application dated November 21, 2013, Attachment I, Section 3.7.1, the licensee stated, in part, that,

> The results presented in [licensee] Evaluation SA-08-006 were developed using a four-loop best-estimate RETRAN-3D model of the plant used in the RETRAN-02 mode. The only exception to the RETRAN-02 mode was that the Chexal-Lellouche drift flux model option was used to better represent depleted mass distributions on the SG secondary and to simulate vapor collecting in the upper regions of the RCS should boiling occur.

The licensee stated that this is a conservative treatment by enhancing the possibility for the model to predict flow stagnation in either/both the upper plenums in the reactor vessel and/or the upper tube region of the SGs. To determine the minimum departure from nucleate boiling ratio (MDNBR) as required in 10 CFR Part 50, Appendix R, Section III.L.2, computer simulations using the NRC-approved (WCGS USAR Section 15.0.11.8, "VIPRE-01") VIPRE-01 Code were conducted for a one-eighth core model using thermal-hydraulic and neutronic boundary conditions from the RETRAN-02 simulations.. Using the VIPRE-01 Code, MDNBR analyses were performed for the scenarios using system boundary conditions from the RETRAN-3D system. The VIPRE-01 Code uses the transient boundary conditions supplied by RETRAN-3D to calculate fuel, cladding, and sub-channel thermal hydraulic conditions in one-fourth of the assemblies to determine if fuel damage has the potential to occur in any of the bounding scenarios.

In Section 3.7.1 of Attachment 1 of its letter dated November 21, 2013, the licensee documented the NRC's acceptance of the RETRAN-3D computer code and conditions placed by the NRC staff on the use of the Chexal-Lellouche drift flux model. The licensee stated, in part, that,

> The NRC approved the use of RETRAN-02 in the Safety Evaluation Report dated September 30, 1993 [(Reference 30)], for the WCNOC Topical Report NSAG-006 "Transient Analysis Methodology for the Wolf Creek Generating Station." The NRC has accepted the RETRAN-3D computer code for use in analyzing Chapter 15 accidents and transients subject to some conditions and limitations. This acceptance is documented in a letter from Stuart Richards, NRC, to Gary Vine, EPRI, dated January 25, 2001, entitled "Safety Evaluation Report (SER) on EPRI Topical Report NP-7450(P), Revision 4, 'RETRAN-3D – A Program for Transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems' TAC No. MA4311)," [(Reference 31)].
>
> . . .
>
> Section 8.0 of the NRC Safety Evaluation Report indicates that use of the Chexal-Lellouche drift flux model will result in the need to assure its use is in conformance with Condition 16. Condition 16 indicates that the results of the analysis using the model must be carefully reviewed. WCNOC has reviewed the results of the model.

The NRC staff reviewed the licensee's use of the Chexal-Lellouche drift flux model in Evaluation SA-08-006 and asked clarifying questions to verify that the Chexal-Lellouche drift flux model was not used by the licensee outside of its validation bounds. The NRC staff also reviewed the licensee's assumptions, calculations, and use of the RETRAN-3D, Chexal-Lellouche, and VIPRE-01 models during an on-site audit in order to verify that the models were appropriately used. Based on its review of the information provided by the licensee by letter dated January 21, 2015 (SRXB-RAI-3) and the staff's on-site audit, the staff concludes that the Chexal-Lellouche drift flux model was not used by the licensee outside of its validation bounds. The NRC staff previously accepted the use of the RETRAN-3D systems model in RETRAN-02 mode. Therefore, based on its review of the licensee's use of the RETRAN-3D systems model,

Chexal-Lellouche drift flux model, and VIPRE-01 sub-channel model, the NRC staff concludes that the methodology used by WCNOC for analyzing the PFSSD is acceptable to determine the MDNBR as required in 10 CFR Part 50, Appendix R.

C3. The results of the consequence analysis

In its application dated November 21, 2013, Attachment 1, Section 3.7.1, the licensee provided the summary results of its PFSSD consequence analysis, including RETRAN-02 calculations and VIPRE-01 calculations. The NRC staff reviewed the consequence analysis and asked clarifying questions, to which the licensee responded in RAI-2 of its letter dated January 21, 2015, and which the staff evaluated during its on-site audit. The staff's questions concerned the range of transient boundary conditions spanned by the VIPRE-01 scenarios, including the normalized power, core outlet pressure, core input flow, and enthalpy.

The NRC staff reviewed the degree to which void formations appeared in boiling scenarios (1, 1A, and 1C) and the potential impact on natural circulation if the voids were to reach the top of the SG U-tubes. The licensee provided clarifying information relative to void growth and behavior in the original and modified boiling scenarios. The RETRAN-02 simulations for scenario 1C were extended past the 7200-second (120-minute) simulation time due to continuing void growth formation in the core. The licensee stated that 10 minutes after the original simulation time (i.e., 130 minutes), the void levels in the primary system were simulated to be sufficient for voids to rise up the SG tube and only condense after the apex of the SG tube. The Chexal-Lellouche drift flux model predicts that the voids will flow past the apex of the SG tube without binding in the apex of the tubes. The NRC staff verified that the scenarios did not result in degradation of the natural circulation pathway. The staff also verified that none of the reported void fractions in the scenarios would prevent sufficient natural circulation in the primary system.

The licensee identified three scenarios (1, 1A, and 1C) with a failed-open pressurizer power-operated relief valve resulting in core boiling and one scenario (3A) with a failed open steam generator atmospheric relief valve, in which pressurizer-water level is not maintained within level indication for 15.7 minutes, which does not meet the requirements of 10 CFR Part 50, Appendix R, Sections III.L.1 and III.L.2. The NRC staff verified that the event timing for scenarios 1, 1A, 1C, and 3A are consistent with the scenarios and the procedure in licensee document OFN-RP-017. The licensee assumes that the FWIS actuates 15 seconds after reactor trip (Section 3.7.3 of Attachment I of the application). The boiling predicted in scenarios 1, 1A, and 1C did not reach void levels which would impede natural circulation in either the upper plenum of the reactor vessel or the apex of the SG tubes. In scenario 3A, the dry pressurizer only impedes the ability for the pressurizer to control RCS pressure and does not impede natural circulation through the RCS. Although the licensee's analysis does not meet the requirements of 10 CFR Part 50, Appendix R, Sections III.L.1 and III.L.2, the results do not impede natural circulation through the reactor vessel, SGs, or RCS. Therefore, based on its review of the information provided by the licensee and its on-site audit, the NRC staff concludes that the licensee's consequence analysis is acceptable, and that the licensee's assumption that the FWIS actuates 15 seconds after reactor trip does not impede the plant from achieving safe shutdown conditions.

3.3 NRC Staff Conclusion

The NRC staff reviewed the postulated fire scenarios, described in the licensee's submittal dated November 21, 2013, Attachment I, Section 3.5, using RG 1.189, "Fire Protection for Nuclear Power Plants." The licensee's postulated fire scenarios and PFSSD consequence analysis are adequate because they meet the objectives of a postulated fire detailed in RG 1.189, which states that an assessment be made for fire damage to safe shutdown equipment on the basis of a single fire, including an exposure fire. Additionally, the effects of a fire on the capability to shut down the plant or prevent a release must be considered. Specifically, a fire affecting one train, which may represent an exposure fire to the alternate train, should be postulated. The licensee has met these criteria with the postulated fire scenario; therefore, the scenarios are acceptable.

The proposed configuration represents an adverse effect on the licensee's ability to safely shut down WCGS, since it does not meet the criteria in 10 CFR Part 50, Appendix R, Section III.L.1 and Section III.L.2. However, based on its review of the information provided by the licensee in this application, the NRC staff concludes that due to the prevention detection, and extinguishing features, and separation of circuits associated with the FWIS within the control room, there is reasonable assurance that a credible fire in the control room would not prevent both automatic and manual isolation of the feedwater system. Therefore, the proposed changes to the fire protection program as described in the WCGS USAR are acceptable.

3.4 Changes to License Condition

The current License Condition 2.C.(5)(a) states that:

> 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, as approved in the SER through Supplement 5, Amendment No. 191, Amendment No. 193, and Amendment No. 205 subject to provisions b and c below.

The revised License Condition 2.C(5)(a) states that:

> 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, as approved in the SER through Supplement 5, Amendment No. 189, Amendment No. 191, Amendment No. 193, Amendment No. 205, and Amendment No. 214, subject to provisions b and c below.

The licensee proposed to add a reference to Amendment No. 189 in License Condition 2.C.(5)(a). Amendment No. 189 was issued on September 30, 2010 (Reference 5), and revised the WCGS fire protection program for the use of fire-resistive cable for certain power and control cables with two motor-operated valves on the component cooling water system. Subsequent to Amendment No. 189, amendments that approved changes to WCGS fire protection program were included in License Condition 2.C.(5)(a). This is an administrative change to the license

condition to consistently reflect license amendments that are applicable to the fire protection program and, therefore, is acceptable to the staff.

The change to the license condition reflects the approved fire protection program based on the issuance of the license amendment approving the proposed change.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, on September 1, 2015, the Kansas State official, Ms. K. Steves, was notified of the proposed issuance of the amendment. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding published in the *Federal Register* on March 18, 2014 (79 FR 15151). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

6.0 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) there is reasonable assurance that 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.

7.0 REFERENCES

1. Broschak, J. P., Wolf Creek Nuclear Operating Corporation, letter to U.S. Nuclear Regulatory Commission, "License Amendment Request (LAR) for Revision to the Wolf Creek Generating Station Fire Protection Program Related to Alternative Shutdown Capability," dated November 21, 2013 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML13331A728).

2. McCoy, J. H., Wolf Creek Nuclear Operating Corporation, letter to U.S. Nuclear Regulatory Commission, "Supplemental Information Regarding License Amendment Request for Revision to the Wolf Creek Generating Station Fire Protection Program Related to Alternative Shutdown Capability," dated December 8, 2014 (ADAMS Accession No. ML14349A433).

3. McCoy, J. H., Wolf Creek Nuclear Operating Corporation, letter to U.S. Nuclear Regulatory Commission, "Response to Request for Additional Information Regarding License Amendment Request to Revise the Fire Protection Program Related to Alternative Shutdown Capability," dated January 21, 2015 (ADAMS Accession No. ML15027A497).

4. Petrick, N. A., Standardized Nuclear Unit Power Plant System (SNUPPS), letter to U.S. Nuclear Regulatory Commission, SLNRC 84-0109, "Fire Protection Review," dated August 23, 1984 (ADAMS Legacy Accession No. 8408300090).

5. Singal, B. K., U.S. Nuclear Regulatory Commission, letter to Matthew W. Sunseri, Wolf Creek Nuclear Operating Corporation, Wolf Creek Generating Station -Issuance of Amendment Re: License Amendment Request for Use of Fire-Resistive Electrical Cable (TAC No. ME2966)," dated September 30, 2010 (ADAMS Accession No. ML102560498).

6. U.S. Nuclear Regulatory Commission, NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, LWR Edition" (SRP) Section 9.5.1.1, "Fire Protection Program," Revision 0, February 2009 (ADAMS Accession No. ML090510170).

7. U.S. Nuclear Regulatory Commission, Regulatory Guide 1.189, "Fire Protection for Nuclear Power Plants," Revision 2, October 2009 (ADAMS Accession No. ML092580550).

8. U.S. Nuclear Regulatory Commission, NUREG-0881, Supplement 5, "Safety Evaluation Report Related to the Operation of Wolf Creek Generating Station, Unit No. 1," March 1985 (ADAMS Accession No. ML091380409).

9. Smith, L. J., U.S. Nuclear Regulatory Commission, letter to Rick A. Muench, Wolf Creek Nuclear Operating Corporation, "Wolf Creek Generating Station – NRC Triennial Fire Protection Inspection Report (05000482/2005008)," dated February 1, 2006 (ADAMS Accession No. ML060330616).

10. Evaluation SA-08-006, "RETRAN-3D Post-Fire Safe Shutdown (PFSSD) Consequence Evaluation for a Postulated Control Room Fire," Revision 3, October 17, 2012.

11. O'Keefe, N., U.S. Nuclear Regulatory Commission, letter to Rick A. Muench, Wolf Creek Nuclear Operating Corporation, "Wolf Creek Generating Station – NRC Triennial Fire Protection Inspection Report (05000482/2008010)," dated January 2, 2009 (ADAMS Accession No. ML090020490).

12. Institute for Electrical and Electronics Engineers, IEEE-383, "Standard for Type Test of Class 1E Electric Cables, Field Splices, and Connections for Nuclear Power Generating Stations."

13. ASTM International, ASTM E-84, "Standard Test Method for Surface Burning Characteristics of Building Materials."

14. Consumer Product Safety Commission Standard FFI-70, "Standard for the Surface Flammability of Carpets and Rugs," Title 16 of the *Code of Federal Regulations*, Section 1630.

15. ASTM International, ASTM D-2679, "Standard Test Method for Electrostatic Charge."

16. American Association of Textile Chemists and Colorists, AATCC Test Method 134, "Electrostatic Propensity of Carpets."

17. ASTM International, ASTM E-662, "Standard Test Method for Specific Optical Density of Smoke Generated by Solid Materials."

18. ASTM International, ASTM E-648, "Standard Test Method for Critical Radiant Flux of Floor Covering Systems Using a Radiant heat Energy Source."

19. National Fire Protection Association, NFPA 253, "Standard Method of Test for Critical Radiant Flux of Floor Covering Systems Using a Radiant Heat Energy Source."

20. National Fire Protection Association, NFPA 72D, "Standard for the Installation, Maintenance and Use of Proprietary Protective Signaling Systems," 1975.

21. National Fire Protection Association, NFPA 12a, "Halogenated Extinguishing Agent Systems – Halon 1301," 1973.

22. National Fire Protection Association, NFPA 10, "Installation of Portable Fire Extinguishers," 1975.

23. National Fire Protection Association, NFPA 14, "Standpipe and Hose Systems," 1976.

24. U.S. Nuclear Regulatory Commission, NUREG/CR-4527, "An Experimental Investigation of Internally Ignited Fires in Nuclear Power Plant Control Cabinets: Part 1: Cabinet Effects Tests," April 1987 (ADAMS Accession No. ML060960351).

25. U.S. Nuclear Regulatory Commission, NUREG/CR-4596, "Screening Tests of Representative Nuclear Power Plants Components Exposed to Secondary Environments Created by Fires," June 1986 (ADAMS Accession No. ML062260142).

26. Lyon, C. F., U.S. Nuclear Regulatory Commission, letter to Adam C. Heflin, "Wolf Creek Nuclear Operating Corporation, Wolf Creek Generating Station - Request for Additional Information Re: License Amendment Request to Revise the Fire Protection Program Related to Alternative Shutdown Capability (TAC No. MF3112)," dated December 4, 2014 (ADAMS Accession No. ML14323A574).

27. U.S. Nuclear Regulatory Commission, "Regulatory Audit Report – Review of Wolf Creek Generating Station Fire Protection Program Related to Alternative Shutdown Capability (TAC No. MF3112)," dated March 31, 2015 (not publicly available – proprietary).

28. Nuclear Energy Institute, NEI 00-01, Revision 2, "Guidance for Post Fire Safe Shutdown Circuit Analysis," May 2009 (ADAMS Accession No. ML091770265).

29. Nuclear Energy Institute, NEI 00-01, Revision 3, "Guidance for Post Fire Safe Shutdown Circuit Analysis," May 2009 (ADAMS Accession No. ML112910147).

30. Reckley, W. D., U.S. Nuclear Regulatory Commission, letter to N. S. Carns, Wolf Creek Nuclear Operating corporation, "Wolf Creek Nuclear Operating Corporation – Transient Analysis Methodology for the Wolf Creek Generating Station (TAC No. M79740)," dated September 30, 1993 (ADAMS Legacy Library No. 9310070215).

31. Richards, S., U.S. Nuclear Regulatory Commission, letter to Gary Vine, Electric Power Research Institute, "Safety Evaluation Report on EPRI Topical Report NP-7450(P), Revision 4, 'RETRAN-3D – A Program for Transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems' (TAC No. MA4311)," dated January 25, 2001 (ADAMS Accession No. ML010470342).

32. U.S. Nuclear Regulatory Commission, Generic Letter 83-11, "Licensee Qualification for Performing Safety Analyses in Support of Licensing Actions," dated February 8, 1983 (ADAMS Accession No. ML031080334).

Principal Contributors: D. Frumkin, NRR/DRA/AFPB
R. Vettori, NRO/DSRA/SPSB
F. Forsaty, NRR/DSS/SRXB

Date: September 11, 2015

September 11, 2015

Mr. Adam C. Heflin
President, Chief Executive Officer,
and Chief Nuclear Officer
Wolf Creek Nuclear Operating Corporation
P.O. Box 411
Burlington, KS 66839

SUBJECT: WOLF CREEK GENERATING STATION - ISSUANCE OF AMENDMENT RE: REVISE THE FIRE PROTECTION PROGRAM RELATED TO ALTERNATIVE SHUTDOWN CAPABILITY (TAC NO. MF3112)

Dear Mr. Heflin:

The U.S. Nuclear Regulatory Commission (the Commission) has issued the enclosed Amendment No. 214 to Renewed Facility Operating License No. NPF-42 for the Wolf Creek Generating Station. The amendment consists of changes to the license in response to your application dated November 21, 2013, as supplemented by letters dated December 8, 2014, and January 21, 2015.

The amendment revises Paragraph 2.C.(5)(a) of the renewed facility operating license and the approved Fire Protection Program as described in the Updated Safety Analysis Report, based on the reactor coolant system thermal hydraulic response evaluation of a postulated control room fire, performed for changes to the alternative shutdown methodology.

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/

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. Amendment No. 214 to NPF-42
2. Safety Evaluation

cc w/encls: Distribution via Listserv

DISTRIBUTION:
PUBLIC
LPL4-1 r/f
RidsAcrsAcnw_MailCTR Resource
RidsNrrDorlDpr Resource
RidsNrrDorlLpl4-1 Resource
RidsNrrDraAfpb Resource
RidsNrrDraAphbResource
RidsNrrDssSrxb Resource
RidsNrrLAJBurkhardt Resource
RidsNrrPMWolfCreek Resource
RidsRgn4MailCenter Resource
FForsaty, NRR/DSS/SRXB
DFrumkin, NRR/DRA/AFPB
RVettori, NRO/DSRA/SPSB
GLapinsky, NRR/DRA/APHB

ADAMS Accession No. ML15183A052 *via memo dated

OFFICE	NRR/DORL/LPL4-1/PM	NRR/DORL/LPL4-1/LA	NRR/DSS/SRXB/BC	NRR/DRA/AFPB/BC
NAME	FLyon	JBurkhardt	CJackson	AKlein (w/comments)
DATE	8/4/15	8/3/15	8/13/15	8/31/15

OFFICE	OGC - NLO	NRR/DORL/LPL4-1/BC	NRR/DORL/LPL4-1/PM
NAME	AGhosh	MMarkley	FLyon
DATE	9/10/15	9/11/15	9/11/15

OFFICIAL RECORD COPY