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

December 7, 2020

Mr. Cleveland Reasoner
Chief Executive Officer and
Chief Nuclear Officer
Wolf Creek Nuclear Operating Corporation
P.O. Box 411
Burlington, KS 66839

SUBJECT: WOLF CREEK GENERATING STATION, UNIT 1 - ISSUANCE OF AMENDMENT NO. 226 RE: EXTENSION OF TYPE A AND TYPE C LEAK RATE TEST FREQUENCIES (EPID L-2020-LLA-0083)

Dear Mr. Reasoner:

The U.S. Nuclear Regulatory Commission (the Commission) has issued the enclosed Amendment No. 226 to Renewed Facility Operating License No. NPF-42 for the Wolf Creek Generating Station, Unit 1. The amendment consists of changes to the Technical Specifications in response to your application dated April 20, 2020.

The amendment revises Technical Specification 5.5.16, “Containment Leakage Rate Testing Program,” for permanent extension of Type A and Type C leak rate test frequencies.

A copy of the related Safety Evaluation is enclosed. Notice of Issuance will be included in the Commission’s monthly *Federal Register* notice.

Sincerely,

/RA/

Samson S. Lee, Project Manager
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-482

Enclosures:
1. Amendment No. 226 to NPF-42
2. Safety Evaluation

cc: Listserv

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

WOLF CREEK NUCLEAR OPERATING CORPORATION

WOLF CREEK GENERATING STATION, UNIT 1

DOCKET NO. 50-482

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 226
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, Unit 1 (the facility) Renewed Facility Operating License No. NPF-42 filed by the Wolf Creek Nuclear Operating Corporation (the Corporation), dated April 20, 2020, 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 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 Technical Specifications as indicated in the attachment to this license amendment and 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. 226, 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 90 days of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Jennifer L. Dixon-Herrity

Digitally signed by Jennifer L. Dixon-Herrity
Date: 2020.12.07 08:12:12 -05'00'

Jennifer L. Dixon-Herrity, Chief
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Renewed Facility
Operating License and
Technical Specifications

Date of Issuance: December 7, 2020

ATTACHMENT TO LICENSE AMENDMENT NO. 226 TO

RENEWED FACILITY OPERATING LICENSE NO. NPF-42

WOLF CREEK GENERATING STATION, UNIT 1

DOCKET NO. 50-482

Replace the following pages of the Renewed Facility Operating License No. NPF-42 and Appendix A Technical Specifications 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

Technical Specifications

REMOVE	INSERT
5.0-20	5.0-20

(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. 226, 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.

5.5 Programs and Manuals

5.5.15 Safety Function Determination Program (SFDP) (continued)

The SFDP identifies where a loss of safety function exists. If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

5.5.16 Containment Leakage Rate Testing Program

a. A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Nuclear Energy Institute (NEI) Topical Report (TR) NEI 94-01, "Industry Guideline for Implementing Performance-Based Option of 10 CFR 50, Appendix J," Revision 3-A, dated July 2012, and the conditions and limitations specified in NEI 94-01, Revision 2-A, dated October 2008, as modified by the following exceptions:

    1. The visual examination of containment concrete surfaces intended to fulfill the requirements of 10 CFR 50, Appendix J, Option B testing, will be performed in accordance with the requirements of and frequency specified by ASME Section XI Code, Subsection IWL, except where relief has been authorized by the NRC.

    2. The visual examination of the steel liner plate inside containment intended to fulfill the requirements of 10 CFR 50, Appendix J, Option B testing, will be performed in accordance with the requirements of and frequency specified by ASME Section XI Code, Subsection IWE, except where relief has been authorized by the NRC.

b. The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 48 psig.

c. The maximum allowable containment leakage rate, L_a, at P_a, shall be 0.20% of containment air weight per day.

d. Leakage rate acceptance criteria are:

    1. Containment leakage rate acceptance criterion is $\leq 1.0\ L_a$. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are $< 0.60\ L_a$ for the Type B and Type C tests and $\leq 0.75\ L_a$ for Type A tests;

(continued)

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 226 TO

RENEWED FACILITY OPERATING LICENSE NO. NPF-42

WOLF CREEK NUCLEAR OPERATING CORPORATION

WOLF CREEK GENERATING STATION, UNIT 1

DOCKET NO. 50-482

1.0 INTRODUCTION

By application dated April 20, 2020 (Reference 1), Wolf Creek Nuclear Operating Corporation (the licensee) requested changes to the Technical Specifications (TSs) for Wolf Creek Generating Station (Wolf Creek or WCGS).

The proposed change would revise TS 5.5.16, "Containment Leakage Rate Testing Program," to extend the Type A and Type C leak rate test frequencies. Specifically, the proposed change to TS 5.5.16 would allow the extension of the Type A integrated leakage rate test (ILRT) containment test interval to 15 years, and the extension of the Type C local leakage rate test (LLRT) interval to 75 months.

2.0 REGULATORY EVALUATION

2.1 Description of Containment

In Section 3.1, "Description of Primary Containment System," of Attachment I to the license amendment request (LAR) dated April 20, 2020, the licensee stated, in part, that:

> The containment structure for WCGS is a prestressed, post-tensioned concrete structure with a cylindrical wall, a hemispherical dome, and a flat foundation slab. The wall and dome form a prestressed, post-tensioned system consisting of horizontal tendons in the wall and inverted U-shaped vertical tendons in the wall and dome. The foundation slab is reinforced with carbon steel. The inside surface of the structure is lined with a carbon steel liner to ensure a high degree of leak tightness. The containment structure completely encloses the reactor and reactor coolant system, i.e., the reactor pressure vessel, the steam generators, the reactor coolant loops and portions of the associated auxiliary systems, the pressurizer, accumulator tanks, and associated piping. The design ensures that the containment structure is protected against postulated missiles from both equipment failures and external sources. The containment design provides means for the integrated leak rate testing of the containment structure and for

Enclosure 2

local leak rate testing of individual piping, electrical, and access penetrations of the containment.

2.2 Licensee's Proposed Changes

The licensee stated in Section 1.0, "Summary Description," of Attachment I to the LAR that the proposed change would revise TS 5.5.16 to reflect the following:

- Increases the existing Type A integrated leakage rate test (ILRT) program test interval from 10 years to 15 years in accordance with Nuclear Energy Institute (NEI) Topical Report (TR) NEI 94-01, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," Revision 3-A [(Reference 2)] and the conditions and limitations specified in NEI 94-01, Revision 2-A [(Reference 3)].

- Adopts an extension of the containment isolation valve (CIV) leakage rate testing (Type C) frequency from the 60 months currently permitted by 10 CFR 50, Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors," Option B, to a 75-month frequency for Type C leakage rate testing of selected components, in accordance with NEI 94-01, Revision 3-A.

- Adopts the use of American National Standards Institute/American Nuclear Society (ANSI/ANS) 56.8-2002, "Containment System Leakage Testing Requirements" [(LaGrange Park, Illinois, November 2002)].

- Adopts a more conservative allowable test interval extension of nine months, for Type A, Type B and Type C leakage rate tests in accordance with NEI 94-01, Revision 3-A.

Specifically, the proposed change contained herein, would revise WCGS TS 5.5.16, paragraph a., by replacing the references to Regulatory Guide (RG) 1.163, "Performance-Based Containment Leak-Test Program," [(Reference 4)] and NEI 94-01, Revision 0, [(Reference 5)] with a reference to NEI 94-01, Revision 3-A, and the limitation and conditions specified in NEI 94-01, Revision 2-A, dated October 2008, as the documents used by WCGS to implement the performance-based leakage testing program in accordance with Option B of 10 CFR 50, Appendix J.

2.3 Regulatory Requirements

The regulations in Title 10 of the *Code of Federal Regulations* (10 CFR) 50.54(o) require that the primary reactor containments for water cooled power reactors shall be subject to the requirements set forth in 10 CFR Part 50, Appendix J. Appendix J to 10 CFR Part 50 specifies containment leakage testing requirements, including the types required to ensure the leak-tight integrity of the primary reactor containment and systems and components, which penetrate the containment. In addition, Appendix J to 10 CFR Part 50 discusses leakage rate acceptance criteria, test methodology, frequency of testing, and reporting requirements for each type of test.

Appendix J to 10 CFR Part 50 includes two options: "Option A—Prescriptive Requirements," and "Option B—Performance-Based Requirements," either of which can be chosen for meeting the requirements of the Appendix. The testing requirements in 10 CFR Part 50, Appendix J, ensure that: (a) leakage through containments or systems and components penetrating containments does not exceed allowable leakage rates specified in the TS and (b) integrity of the containment structure is maintained during the service life of the containment.

Wolf Creek adopted Option B in TS 5.5.16. The adoption of the Option B performance-based containment leakage rate testing for Type A, B, and C testing does not alter the basic method by which Appendix J leakage rate testing is performed; however, it does alter the frequency at which Type A, B, and C containment leakage tests must be performed. Under the performance-based option of 10 CFR Part 50, Appendix J, the test frequency is based upon an evaluation that reviewed the as-found leakage history to determine the frequency for leakage testing, which provides assurance that leakage limits will be maintained.

The regulations in 10 CFR 50.55a, "Codes and standards," contain the containment inservice inspection (ISI) requirements, which, in conjunction with the requirements of 10 CFR Part 50, Appendix J, ensure the continued leaktight and structural integrity of the containment during its service life.

The regulations in 10 CFR 50.65, "Requirements for monitoring the effectiveness of maintenance at nuclear power plants," paragraph (a)(1), state, in part, that the licensee:

> … shall monitor the performance or condition of structures, systems, or components, against licensee-established goals, in a manner sufficient to provide reasonable assurance that these structures, systems, and components, …are capable of fulfilling their intended functions. These goals shall be established commensurate with safety and, where practical, take into account industry-wide operating experience.

The regulations in 10 CFR 50.36, "Technical specifications," state that the TSs must include items in the following five specific categories: (1) safety limits, limiting safety system settings, and limiting control settings; (2) limiting conditions for operations; (3) surveillance requirements; (4) design features; and (5) administrative controls.

2.4 Regulatory Guidance

2.4.1 NEI-94-01

NEI 94-01, Revision 0, dated July 1995, provides methods for complying with Option B of 10 CFR Part 50, Appendix J, and allows for the extension of the performance-based Type A test interval for up to 10 years, based upon two consecutive successful tests. NEI 94-01, Revision 0, is endorsed by the U.S. Nuclear Regulatory Commission (NRC) in RG 1.163 with some conditions.

NEI 94-01, Revision 2-A, dated October 2008, incorporated the NRC conditions in RG 1.163 and added provisions for extending Type A test intervals up to 15 years. This revision of NEI 94-01 was supported by Electric Power Research Institute (EPRI) Report No. 1009325, Revision 2, “Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals,” dated August 2007 (Reference 6). The EPRI report provides a generic assessment of the risks associated with permanently extending the ILRT interval to 15 years, and it provides a risk-informed methodology to be used to confirm the risk impact of the ILRT extension on a plant-specific basis. Probabilistic risk assessment (PRA) methods are used in combination with ILRT performance data and other considerations to justify the extension of the ILRT interval. This is consistent with the guidance provided in RG 1.174, “An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis,” Revision 3, dated January 2018 (Reference 7) and RG 1.177, “An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications,” Revision 1, dated May 2011 (Reference 8) (to support changes to test intervals.

The NRC staff’s review of both NEI 94-01, Revision 2, and EPRI Report No. 1009325, Revision 2, is described in the NRC safety evaluation (SE) dated June 25, 2008 (Reference 9). As stated in the NRC SE, NEI 94-01, Revision 2 describes an acceptable approach for implementing the optional performance-based requirements of 10 CFR Part 50, Appendix J, Option B. The NRC staff concluded that NEI 94-01, Revision 2 is acceptable for referencing by licensees proposing to amend their containment leakage rate testing TSs, subject to the conditions listed in Section 4.1 of the SE.

NEI 94-01, Revision 3-A, dated June 8, 2012, added guidance for extending Type C LLRT intervals beyond 60 months and incorporated the two conditions from the NRC staff’s SE (Reference 10).

2.4.2 PRA Quality

Consistent with the information provided in RIS 2007-06, “Regulatory Guide 1.200 Implementation,” dated March 22, 2007 (Reference 11), the NRC staff will use RG 1.200, Revision 2, to assess technical adequacy of the PRA used to support risk-informed applications received by the end of 2009. In Section 3.2.4.1, “Quality of the PRA,” of the NRC SE for EPRI Report No. 1009325, the NRC staff stated that Capability Category I of the ASME PRA Standard (i.e., ASME/ANS-RA-Sa-2009) shall be applied as the standard for assessing PRA quality for ILRT extension applications, since approximate values of core damage frequency (CDF) and large early release frequency (LERF) and their distribution among release categories are sufficient to support the evaluation of changes to ILRT frequencies.

3.0 TECHNICAL EVALUATION

3.1 Type A, B, and C Leak Rate Test Program and Historical Test Results

The licensee stated that Wolf Creek TS 5.5.16 maximum allowable containment leakage rate acceptance criteria (L_a) are 0.20 percent of containment air weight per day. The peak calculated containment internal pressure for the design basis loss-of-coolant accident is 48 pounds per square inch gauge.

The containment leakage rate testing program leakage rate acceptance criteria is less than or equal to (≤) 1.0 L_a. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are less than (<) 0.60 L_a for the Type B and Type C tests and ≤ 0.75 L_a for Type A tests.

Four ILRTs have been performed on the Wolf Creek containment since start up and these tests resulted in satisfactory leakage rates being observed. The licensee provided the test results in Section 3.3.5, "Integrated Leakage Rate Testing (ILRT) History," and summarized the test results in Tables 3.3.5-1 and 3.3.5-2 of Attachment I to the LAR.

3.2 Containment Inspection and Testing Program

In Section 3.5.2, "Containment Inservice Inspection (CISI) Program," of Attachment I to the LAR, the licensee provided information related to the ISI performed at Wolf Creek. The licensee stated that the CISI Program complies with the 2013 Edition of American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (BPV) Code (ASME Code), Section XI, Subsections IWA, IWE, and IWL, along with the appropriate requirements of 10 CFR 50.55a, and to the conditions specified in 10 CFR Part 50.55a(b)(2), "Conditions on ASME BPV Code, Section XI."

The licensee incorporated RG 1.147, "Inservice Inspection Code Case Acceptability, ASME Section XI, Division 1," endorsed ASME Code Case N-532-5, "Alternative Requirements to Repair and Replacement Documentation Requirements and Inservice Summary Report Preparation and Submission as Required by IWA-4000 and IWA-6000, Section XI, Division 1" into the Wolf Creek ISI Program. In accordance with 10 CFR 50.55a(b), ASME Section XI Code Cases referenced in RG 1.147 may be incorporated into the WCGS ISI Program.

In Section 3.5.3, "Supplemental Inspection Requirements," of Attachment I to the LAR, the licensee stated that supplemental inspections will not be required. Rather, inspections of the exterior containment concrete surfaces and the steel liner plate inside containment will be conducted in accordance with Wolf Creek TS 5.5.16 as modified by Amendment No. 152 (Reference 12) by adding the following exceptions to RG 1.163:

- The visual examination of containment concrete surfaces intended to fulfill the requirements of 10 CFR 50, Appendix J, Option B testing, will be performed in accordance with the requirements of and frequency specified by ASME Section XI Code, Subsection IWL, except where relief has been authorized by the NRC.

- The visual examination of the steel liner plate inside containment intended to fulfill the requirements of 10 CFR 50, Appendix J, Option B testing, will be performed in accordance with the requirements of and frequency specified by the ASME Section XI Code, Subsection IWE, except where relief has been authorized by the NRC.

In Section 3.5.4, "Results of Recent Containment Examinations," of Attachment I to the LAR, the licensee presented the results of recent visual inspections for ASME Section XI Code, Subsections IWE and IWL examinations conducted during refueling outage (RFO) RFO22 (spring 2018). The licensee evaluated the ASME Section XI Code, Subsection IWE inspection results and concluded that they were acceptable, and that the containment pressure boundary continues to perform its intended function as a leaktight barrier. The licensee also presented the results of the most recent ASME Section XI Code, Subsection IWL examination performed in October of 2015. This examination was the 30th Year ASME Section XI Code, Subsection IWL Tendon Surveillance of the Wolf Creek containment building's post-tensioning system and concrete structure. This inspection is an implementation of IWL examination requirements and is completed once every 5 years per the program schedule. This examination is a systematic means of assessing the quality and structural performance of the post-tensioning system. In summary, the licensee stated that the Final Report for the 30th Year tendon surveillance at Wolf Creek has concluded that the functional integrity of the selected post-tensioning system has met the applicable Code requirements, unless noted otherwise with non-conformance items, which were recorded, identified, and dispositioned as required.

In Section 3.6, "Operating Experience (OE)," of Attachment I to the LAR, the licensee evaluated the following site-specific and industry events for applicability to the Subsection IWE Program:

- NRC Information Notice (IN) 1992-20, "Inadequate Local Leak Rate Testing," dated March 3, 1992 (Reference 13);

- IN 2010-12, "Containment Liner Corrosion," dated June 18, 2010 (Reference 14);

- IN 2014-07, "Degradation of Leak-Chase Channel Systems for Floor Welds of Metal Containment Shell and Concrete Containment Metallic Liner," dated May 5, 2014 (Reference 15);

- NRC Regulatory Issue Summary (RIS) 2016-07, "Containment Shell or Liner Moisture Barrier Inspection," dated May 9, 2016 (Reference 16); and

- Licensee Event Report (LER) 2020-001-00, "Plant Shutdown Due to Inoperable Containment Purge Isolation Valves," dated April 1, 2020 (Reference 17).

- In Table 3.5.2-3 of the LAR, the licensee added one augmented examination for containment surfaces for Items E4.11 and E1.12 under examination Category E-C for the third containment ISI interval. Results of RFO22 IWE examination identified damage/degradation of these components but the licensee's corrective action report concluded that the noted damage/degradation does not affect the structural integrity of the containment pressure boundary. The components are added to the augmented inspections Category E-C.

The licensee provided the results of the review of the applicable INs and RIS and demonstrated how such information is used to inform the programs for maintaining the overall containment integrity at Wolf Creek:

- In Section 3.6.1 of Attachment I to the LAR, the licensee described IN 1992-20 and determined it did not apply to Wolf Creek since it applied to boiling-water reactor plants for two cases and Wolf Creek has procedures in place to address the third case.

- In Section 3.6.2 of Attachment I to the LAR, the licensee described IN 2010-12 and determined that the IN is applicable to Wolf Creek, and concluded that the existing visual inspection procedure within the ASME Section XI Code, Subsection IWE program has provisions to identify corrosion and bulging of the containment liner plate.

- In Section 3.6.3 of Attachment I to the LAR, the licensee described IN 2014-07 and determined that it is not applicable to Wolf Creek because Wolf Creek does not have leak-chase test connections that are below the concrete floor level that would have water intrusion as noted. The Wolf Creek leak-chase test connections are above floor grade. However, because of the increased awareness of water intrusion into the leak-chase system, the inspection description of test point locations was revised in the second 10-year Interval of the Wolf Creek CISI Program, Revision 6, to include the test plug.

- In Section 3.6.4 of Attachment I to the LAR, the licensee discussed NRC RIS 2016-07 that identified several instances in which the containment shell or liner plate moisture barrier materials (e.g., caulking, flashing, and other sealants used for this application) were not properly inspected in accordance with the ASME Section XI Code, Subsection IWE Program. In order to address the RIS, the licensee performed a plant equipment location drawing review and determined that the concrete and the grating steel do not come in contact with the liner. A walkdown of the containment was also performed in RFO 21 (October 11, 2016) to determine if there are locations where the containment liner is in contact with concrete or steel and has some type of moisture barrier installed that is not presently identified in the Wolf Creek CISI Program Plan. The licensee did not identify any moisture barrier locations from the walkdown, and therefore, no further action was necessary.

- In Section 3.6.5 of Attachment I to the LAR, the licensee discussed LER 2020-001-00. The event occurred on February 1, 2020, when the licensee, during the surveillance testing of CIVs associated with the containment shutdown purge supply piping, discovered that the leakage rate through the penetration was greater than that allowed by the TS. Two CIVs in series were determined to be inoperable and due to the high leakage rate, containment was declared inoperable and was shutdown. Both valves were returned to service the following day, and Wolf Creek subsequently returned to Mode 1 on February 3, 2020.

- In Section 3.7, “License Renewal Aging Management,” of Attachment I to the LAR, the licensee, per the requirement of 10 CFR 54.21(d), described the license renewal commitments for aging management programs in the Updated Safety Analysis Report, Chapter 18, Appendix A, “Introduction and License Renewal Commitments,” and described enhancements to the ISI programs beyond the requirements of ASME Section XI. The licensee provided a summary of the applicable ASME Section XI and the 10 CFR Part 50 Appendix J commitments in Table 3.7-1, “License Renewal Commitments Supplementing ASME Section XI Requirements,” of the LAR, for ASME Section XI ISI Subsection IWE, ASME Section XI ISI Subsection IWL, 10 CFR Part 50, Appendix J, Concrete Containment Tendon Prestress, Containment Liner Plate, Polar Crane Bracket, and Penetration Load Cycles.

- In Section 3.5.1, “Nuclear Coatings Program,” of Attachment I to the LAR, the licensee described the protective coatings program of Service Level I coatings applied to the structures, systems and components located inside the primary containment that are performed during every refueling outage. The regulatory requirements for the coatings program are based on RG 1.54, Revision 3, “Service level I, II, III, and In-Scope License Renewal Protective Coatings Applied to Nuclear Power Plants,” dated April 2017 (Reference 18), and defines the Service Level I as applicable to coating failures (detached coatings) that adversely affect the operations of post-accident fluid systems and impair safe shutdown (e.g., emergency core cooling system (ECCS)). The licensee described the tracking process to quantify “unqualified” coatings to ensure that the documented quantity does not exceed the postulated maximum allowable quantity in the Wolf Creek Containment Building.

The NRC staff reviewed the information summarized above and finds that the licensee acceptably addressed the relevant regulatory requirements, guidance, and operating experience described above through inspection and aging management programs. Therefore, the licensee's CISI Program provides reasonable assurance that the containment will maintain its capability to perform its safety-related function.

3.3 NEI 94-01 Conditions

3.3.1 NEI 94-01, Revision 2-A

NEI 94-01, Revision 2-A, contains six conditions. The NRC staff evaluated whether the licensee adequately addressed these conditions.

3.3.1.1 NEI 94-01, Revision 2-A, Condition 1

Limitation and Condition 1 of NEI 94-01, Revision 2-A, states:

> For calculating the Type A leakage rate, the licensee should use the definition [of the performance leakage rate] in the NEI TR 94-01, Revision 2, in lieu of that in ANSI/ANS-56.8-2002. (Refer to SE Section 3.1.1.1).

The licensee stated in Section 3.8.1 of Attachment I to the LAR that it will use the definition in Section 5.0 of NEI 94-01, Revision 3-A. The definition of the performance leakage rate in Revision 2, Revision 2-A, and Revision 3-A of NEI 94-01 has remained unchanged. Therefore, the NRC staff concludes that the licensee adequately addressed Condition 1.

3.3.1.2 NEI 94-01, Revision 2-A, Condition 2

Limitation and Condition 2 of NEI 94-01, Revision 2-A, states:

> The licensee submits a schedule of containment inspections to be performed prior to and between Type A tests. (Refer to SE Section 3.1.1.3).

The licensee provided a schedule for the Subsection IWE examinations of Section XI of the ASME Code at Wolf Creek for Category E-A, E-C and E-G components. The schedule is contained in the CISI Plan, Appendix B. The Appendix B schedule is detailed and componentized by outage and period within the current 10-year interval. The IWE examination schedule is summarized in Table 3.5.2-5, "IWE Examination Schedule," of the LAR.

The schedule for the Code Subsection IWL examinations at Wolf Creek for concrete Category L-A and tendons Category L-B is contained in the CISI Program Plan, Appendix C. The Appendix C schedule is detailed and componentized by outage and period within the current 10-year interval. This IWL examination schedule is summarized in Table 3.5.2-8, "IWL Examination Schedule," of the LAR.

The NRC staff reviewed Tables 3.5.2-5 and 3.5.2-8 of the LAR and finds that NEI 94-01 Revision 2-A SE Condition 2 has been met.

3.3.1.3 NEI 94-01, Revision 2-A, Condition 3

Limitation and Condition 3 of NEI 94-01, Revision 2-A, states:

> The licensee addresses the areas of the containment structure potentially subjected to degradation. (Refer to SE Section 3.1.3).

As discussed in Section 3.2 of this SE regarding actions taken by the licensee in response to containment structure operating experience, the NRC staff finds that the licensee provided an acceptable level of information regarding the implementation of the Examination Categories of E-A and E-C of ASME Section XI, Subsection IWE, that exhibit and/or would indicate the presence of potential degraded conditions in the accessible and inaccessible areas of the containment concrete and liner plate. Therefore, the NRC staff concluded that the licensee has adequately addressed Condition 3.

3.3.1.4 NEI 94-01, Revision 2-A, Condition 4

Limitation and Condition 4 of NEI 94-01, Revision 2-A, states:

> The licensee addresses any tests and inspections performed following major modifications to the containment structure, as applicable. (Refer to SE Section 3.1.4).

In Table 3.8.1-1 of the LAR, the licensee stated that "There have been no major or minor containment repairs or modifications performed nor are any repairs or modifications planned for the containment structure."

Based on the information above, the NRC staff finds that that Condition 4 is not applicable since there have been no major or minor containment repairs or modifications performed nor are any repairs or modifications planned for the containment structure.

3.3.1.5 NEI 94-01, Revision 2-A, Condition 5

Limitation and Condition 5 of NEI 94-01, Revision 2-A, states:

> The normal Type A test interval should be less than 15 years. If a licensee has to utilize the provision of Section 9.1 of NEI 94-01, Revision 2, related to extending the ILRT interval beyond 15 years, the licensee must demonstrate to the NRC staff that it is an unforeseen emergent condition. (Refer to SE Section 3.1.1.2)

In response to Condition 5, the licensee stated that "WCGS will follow the requirements of NEI 94-01 Revision 3-A, Section 9.1. This requirement has remained unchanged from Revision 2-A to Revision 3-A of NEI 94-01."

In accordance with the requirements of NEI 94-01, Revision 2-A, SE Section 3.1.1.2, the licensee committed in the LAR to demonstrate to the NRC staff that an unforeseen emergent condition exists in the event an extension beyond the 15-year interval is required.

Therefore, the NRC staff concludes that the licensee has adequately addressed Condition 5.

3.3.1.6 NEI 94-01, Revision 2-A, Condition 6

Limitation and Condition 6 of NEI 94-01, Revision 2-A, states:

> For plants licensed under 10 CFR Part 52, applications requesting a permanent extension of the ILRT surveillance interval to 15 years should be deferred until after the construction and testing of containments for that design have been completed and applicants have confirmed the applicability of NEI TR 94-01, Revision 2 and EPRI Report No. 1009325, Revision 2, including the use of past containment ILRT data,

Condition 6 is not applicable to Wolf Creek because it was not licensed under 10 CFR Part 52. Wolf Creek was licensed under 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities."

3.3.2 NEI 94-01, Revision 3-A

NEI 94-01, Revision 3-A, contains two conditions in the NRC SE that the licensee responded to.

3.3.2.1 NEI 94-01, Revision 3-A, Condition 1

Condition 1, Issue 1

> The allowance of an extended interval for Type C LLRTs of 75 months carries the requirement that a licensee's post-outage report include the margin between the Type B and Type C leakage rate summation and its regulatory limit.

In response to Issue 1, the licensee stated in the LAR that:

> The post-outage report shall include the margin between the Type B and Type C MNPLR [minimum pathway leakage rate] summation value, as adjusted to include the estimate of applicable Type C leakage understatement, and its regulatory limit of 0.60 L_a.

Condition 1, Issue 2

> In addition, a corrective action plan shall be developed to restore the margin to an acceptable level.

In response to Issue 2, the licensee stated in the LAR that:

> When the potential leakage understatement adjusted Types B and C MNPLR total is greater than the WCGS administrative leakage summation limit of 0.5 L_a, but less than the regulatory limit of 0.6 L_a, then an analysis and determination of a corrective action plan shall be prepared to restore the leakage summation margin to less than the WCGS leakage limit. The corrective action plan will focus on those components which have contributed the most to the increase in the leakage summation value and what manner of timely corrective action, as deemed appropriate, best focuses on the prevention of future component leakage performance issues so as to maintain an acceptable level of margin.

Condition 1, Issue 3

> Use of the allowed 9-month extension for eligible Type C valves is only authorized for non-routine emergent conditions with exceptions as detailed in NEI 94-01, Revision 3-A, Section 10.1.

In response to Issue 3, the licensee stated in the LAR that:

> WCGS will apply the 9-month allowable interval extension period only to eligible Type C components and only for non-routine emergent conditions. Such occurrences will be documented in the record of tests.

The NRC staff reviewed the licensee's responses and finds that each of the three issues has been satisfactorily addressed, and therefore Condition 1 of the NEI 94-01 Revision 3-A SE has been satisfactorily addressed.

3.3.2.2 NEI 94-01, Revision 3-A, Condition 2

Condition 2, Issue 1

> Extending the LLRT intervals beyond 5 years to a 75-month interval should be similarly conservative provided an estimate is made of the potential understatement and its acceptability determined as part of the trending specified in NEI TR 94-01, Revision 3, Section 12.1.

In response to Issue 1, the licensee stated in the LAR that:

> The change in going from a 60-month extended test interval for Type C tested components to a 75-month interval, as authorized under NEI 94-01, Revision 3-A, represents an increase of 25% in the LLRT periodicity. As such, WCGS will conservatively apply a potential leakage understatement adjustment factor of 1.25 to the actual As-left leak rate, which will increase the As-left leakage total for each Type C component currently on greater than a 60-month test interval up to the 75-month extended test interval. This will result in a combined conservative Type C total for all 75-month LLRTs being "carried forward" and will be included whenever the total leakage summation is required to be updated (either while on-line or following an outage).
>
> When the potential leakage understatement adjusted leak rate total for those Type C components being tested on greater than a 60-month test interval up to the 75-month extended test interval is summed with the non-adjusted total of those Type C components being tested at less than or equal to a 60-month test interval. . ., and the total of the Type B tested components, results in the MNPLR being greater than the WCGS administrative leakage summation limit of 0.50 L_a, but less than the regulatory limit of 0.6 L_a, then an analysis and corrective action plan shall be prepared to restore the leakage summation value to less than the WCGS leakage limit. The corrective action plan should focus on those components which have contributed the most to the increase in the leakage summation value and what manner of timely corrective action, as deemed appropriate, best focuses on the prevention of future component leakage performance issues.

Condition 2, Issue 2

> When routinely scheduling any LLRT valve interval beyond 60 months and up to 75 months, the primary containment leakage rate testing program trending or monitoring must include an estimate of the amount of understatement in the Types B and C total, and must be included in a licensee's post-outage report. The report must include the reasoning and determination of the acceptability of the extension, demonstrating that the LLRT totals calculated represent the actual leakage potential of the penetrations.

In response to Issue 2, the licensee stated in the LAR that:

> If the potential leakage understatement adjusted leak rate MNPLR is less than the WCGS administrative leakage summation limit of 0.50 L_a, then the acceptability of the greater than a 60-month test interval up to the 75-month LLRT extension for all affected Type C components has been adequately demonstrated, and the calculated local leak rate total represents the actual leakage potential of the penetrations. . . . A post-outage report shall be prepared presenting results of the previous cycle's Type B and Type C tests, and Type A, Type B and Type C tests, if performed during that outage.

The NRC staff reviewed the licensee's responses and finds they satisfactorily address Condition 2 of the NEI 94-01 Revision 3-A SE.

3.4 Plant Specific Risk Evaluation

The licensee provided a plant specific risk assessment for permanently extending the currently allowed containment Type A ILRT interval from 10 years to 15 years in the Enclosure, "Evaluation of Risk Significance of Permanent ILRT Extension," to the LAR.

In Section 3.4.1, "Methodology," of Attachment I to the LAR, the licensee stated that the plant-specific risk assessment follows the guidance in NEI 94-01, Revision 3-A; the NEI "Interim Guidance for Performing Risk Impact Assessments in Support of One-Time Extensions for Containment Integrated Leakage Rate Test Surveillance Intervals," dated November 2001; RG 1.200, Revision 2, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," dated March 2009 (Reference 19), as applied to ILRT extensions; RG 1.174, Revision 3, risk insights in support of plant's licensing basis request; the methodology described in EPRI Report No. 1018243 (also identified as EPRI Report No. 1009325, Revision 2-A). Additionally, the licensee applied the methodology from Calvert Cliffs Nuclear Power Plant to estimate the likelihood and risk implications of corrosion-induced leakage of steel liners going undetected during extended test interval (Reference 20).

The licensee addressed each of the four conditions for the use of EPRI Report No. 1009325, Revision 2, which are listed in Section 4.2 of the NRC SE dated June 25, 2008. A summary of how each condition is met is provided in Sections 3.4.1 through 3.4.4 below.

3.4.1 PRA Quality – Condition 1

The first condition stipulates that the licensee submit documentation indicating that the technical adequacy of its PRA is consistent with the guidance in RG 1.200 relevant to the ILRT extension application. This RG describes one acceptable approach for determining whether the technical adequacy of the PRA, in total or the parts that are used to support an application, is sufficient to provide confidence in the results such that the PRA can be used in regulatory decisionmaking for LWRs.

The licensee addressed Wolf Creek's PRA technical adequacy in Section 3.4.2.1, "PRA Quality Statement for Permanent 15-Year ILRT Extension," of Attachment I to the LAR. As discussed in Section 3.4.2.1, the Wolf Creek risk assessment performed to support the ILRT application utilized the current internal events PRA model of record, which the licensee completed in February 2020. The licensee explains its approach to establishing and maintaining the technical adequacy and plant fidelity of the PRA models. This approach includes both a proceduralized PRA maintenance and update process, and the use of self-assessments and independent peer reviews. In Section 3.4.2.2, "PRA Maintenance and Update," of Attachment I to the LAR, the licensee provides further details of the process used to ensure its PRA model reflects the as-built and as-operated plant.

Wolf Creek PRA model for internal events received a formal industry peer review in June 2019. Following the peer review, an independent assessment of the facts and observations (F&Os) closures was performed between November 2019 and March 2020. The independent assessment followed the guidance of NEI 05-04, "Process for Performing Internal Events PRA Peer Reviews Using the ASME/ANS PRA Standard," Appendix X (Reference 21). At the conclusion of the independent assessment, three of the original F&Os remained open, with one F&O deemed to require a PRA upgrade. The PRA upgrade required a focused scope peer review, which assigned and opened a fourth F&O. Details of the four open F&Os were provided in the enclosure, "Evaluation of Risk Significance of Permanent ILRT Extension," to the Wolf Creek LAR. An NRC staff review of these findings and observations found that there is no, if any material impact, on the LAR.

With respect to external events, RG 1.174 stipulates that established acceptance guidelines are intended for comparison with a full-scope assessment of the change in the applicable risk metrics and recognizes that many PRAs are not full scope and PRA information of less than full scope may be acceptable. The methodology described in EPRI Report No. 1009325, which the NRC found, satisfies the key principles of risk-informed decisionmaking of RG 1.174, explains that if the external event analysis is not of sufficient quality or detail to allow direct application of the methodology, the quality or detail will be increased or a suitable estimate of the risk impact from the external events should be performed. This assessment can be taken from existing, previously submitted and approved analyses or another alternate method of assessing an order-of-magnitude estimate for contribution of the external event to the impact of the changed interval. Based on this, the licensee performed a bounding, order-of-magnitude, analysis of the potential impacts from external events. This analysis references the currently available information for external events models and information to develop an "external events multiplier" to be applied to the internal events results. The licensee's external events contribution assessment quantified an estimated LERF fire value based on the Individual Plant Examination of External Events fire PRA calculated CDF adjusted by the internal events ratio of LERF/CDF from the latest PRA Model revision. The fire LERF was estimated as 5.48E-8/yr. Regarding seismic contributions, the licensee used the seismic CDF determined by Generic Issue 199 (GI-199), "Implications of Updated Probabilistic Seismic Hazard Estimates in Central and Eastern United States on Existing Plants: Safety/Risk Assessment," Table D-1, "Seismic Core-Damage Frequencies Using 2008 USGS Seismic Hazards Curves," to estimate the LERF value (Reference 22). Using the same LERF/CDF ratio, seismic LERF was estimated as 4.34E-8/yr.

Based on review of the above information, the NRC staff finds that the licensee has addressed the relevant findings and gaps from the peer reviews and that they have no impact on the results of this LAR. Therefore, the NRC staff concludes that the internal events PRA model used by the licensee is of sufficient quality to support the evaluation of changes to ILRT frequencies. Accordingly, the first condition is met.

3.4.2 Estimated Risk Increase – Condition 2

The second condition stipulates that the licensee submit documentation indicating that the estimated risk increase associated with permanently extending the ILRT interval to 15 years is "small," consistent with the guidance in RG 1.174 and the clarification provided in the NRC SE for EPRI Report No. 1009325. Specifically, a "small" increase in population dose should be defined as an increase in population dose of less than or equal to either 1.0 person-rem per year or 1 percent of the total population dose, whichever is less restrictive. In addition, a "small" increase in conditional containment failure probability (CCFP) should be defined as a value marginally greater than that accepted in previous one-time 15-year ILRT extension requests. This would require that the increase in CCFP be less than or equal to 1.5 percentage point.

The licensee reported the results of the plant-specific risk assessment in Section 5.2 and sensitivity calculations in Section 5.3 of the Enclosure to the LAR. The reported risk impacts are based on a change in the Type A containment ILRT frequency from three tests in 10 years (the test frequency under 10 CFR 50 Appendix J, Option A) to one test in 15 years. The following conclusions can be drawn from the licensee's analysis associated with extending the Type A ILRT frequency:

1. LERF is the relevant risk metric for ILRT Type A testing for plants with no reliance on over-pressure of containment to ensure adequate net positive suction head of the ECCS pumps. Since the Wolf Creek design does not rely on over-pressure of containment to ensure adequate net positive suction head of the ECCS pumps, LERF is the relevant risk metric for this LAR. RG 1.174 defines "very small" changes in risk as resulting in an increase of LERF of less than 1.0E-7/year respectively. RG 1.174 considers a "small" change in LERF to be between 1E-7/year and 1E-6/year with a total LERF less than 1E-5. The increase in LERF resulting from a change in the Type A ILRT test interval from 3-in-10 years to 1-in-15 years is estimated as 1.48E-7/year using the EPRI guidance, with a negligible increase from on-going and undetected corrosion-induced leakage of the steel liners, as a result of the test interval extension. The total internal events LERF is 2.58E-7/yr. Therefore, the estimated change in LERF is determined to be "small" using the acceptance guidelines of RG 1.174. When external event risk is included, the increase in LERF resulting from a change in the Type A ILRT test interval from 3-in-10 years to 1-in-15 years is estimated as 2.81E-7/year using the EPRI guidance, and total LERF is 4.89E-7/year. As such, the estimated change in LERF is also determined to be "small" using the acceptance guidelines of RG 1.174.

2. The effect resulting from changing the Type A test frequency to 1-per-15 years, measured as an increase to the total integrated plant risk for those accident sequences influenced by Type A testing, is 0.645 person-rem/year. NEI 94-01 states that a small total population dose is defined as an increase of ≤ 1.0 person-rem/year, or ≤ 1 percent of the total population dose, whichever is less restrictive for the risk impact assessment of the extended ILRT intervals. The reported increase in total

population dose is below the acceptance criteria provided in the NRC SE for EPRI Report No. 1009325. Thus, the increase in the total integrated plant risk for the proposed change is considered "small" and supportive of the proposed change.

3. The increase in the CCFP due to the change in test frequency from 3 in 10 years to 1 in 15 years is 0.911 percent. NEI 94-01 states that increases in CCFP of ≤ 1.5 percent is small. This value is below the acceptance guidelines in Section 3.2.4.6 of the NRC SE for NEI 94-01, Revision 2 and supportive of the proposed change.

Based on the review of the licensee's risk assessment results, the NRC staff concludes that the increase in LERF is "small" and consistent with the acceptance guidelines of RG 1.174, and the increase in the total population dose and the magnitude of the change in the CCFP for the proposed change are "small." The defense-in-depth philosophy is maintained as the independence of barriers will not be degraded because of the requested change, and the use of the quantitative risk metrics collectively ensures that the balance between prevention of core damage, prevention of containment failure, and consequence mitigation is preserved. Accordingly, the second condition is met.

3.4.3 Leak Rate for the Large Pre-Existing Containment Leak Rate Case – Condition 3

The third condition stipulates that for the methodology in EPRI Report No. 1009325 to be acceptable, the average leak rate for the pre-existing containment large leak rate accident case (i.e., accident case 3b) used by the licensees shall be 100 L_a instead of 35 L_a. As noted by the licensee in Section 4 of the Enclosure to the LAR, the methodology in EPRI Report No. 1009325, Revision 2-A, incorporated the use of 100 L_a as the average leak rate for the pre-existing containment large leakage rate accident case (accident case 3b). Therefore, the 100 L_a value was used in the Wolf Creek plant-specific risk assessment. Accordingly, the third condition is met.

3.4.4 Containment Overpressure is Relied Upon for ECCS Performance – Condition 4

The fourth condition stipulates that in instances where containment over-pressure is relied upon for ECCS performance, a LAR is required to be submitted. In Section 3.2 of the LAR, the licensee stated that Wolf Creek does not rely on containment over-pressure for ECCS performance. Accordingly, the fourth condition is not applicable.

3.5 Technical Evaluation Conclusion

The NRC staff reviewed the proposed change to Wolf Creek TS 5.5.16, "Containment Leakage Rate Testing Program," to extend the Type A and Type C leak rate test frequencies. Specifically, the proposed change to TS 5.5.16 would allow the extension of the Type A ILRT containment test interval to 15 years, and the extension of the Type C LLRT interval to 75 months.

The proposed change would adopt NEI 94-01, Revisions 2-A and 3-A, which were accepted by the NRC staff with conditions. As discussed above in this SE, the licensee has adequately addressed those conditions. Based on the regulatory and technical evaluations in Sections 2.0 and 3.0 of this SE, the NRC staff finds that the licensee has adequately justified the proposed TS changes in its application.

The NRC staff finds that the proposed TS changes will be adequate to maintain the containment leakage limits. The NRC staff concludes there is reasonable assurance the requirements of Appendix J to 10 CFR Part 50, 10 CFR 50.55a, 10 CFR 50.65(a)(1), and 10 CFR 50.36 will continue to be met. In conclusion, the NRC staff finds the proposed changes in the LAR acceptable.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Kansas State official was notified of the proposed issuance of the amendment on September 21, 2020. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes requirements with respect to installation or use of facility components 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 June 2, 2020 (85 FR 33753). 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. McCoy, J. H., Wolf Creek Nuclear Operating Corporation, letter to U.S. NRC, "Docket No. 50-482: License Amendment Request to Revise Technical Specification 5.5.16 for Permanent Extension of Type A and Type C Leak Rate Test Frequencies," dated April 20, 2020 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML20111A327).

2. Nuclear Energy Institute, NEI 94-01, Revision 3-A, Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," dated July 2012 (ADAMS Accession No. ML12221A202).

3. Nuclear Energy Institute, NEI 94-01, Revision 2-A, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J, dated October 2008 (ADAMS Accession No. ML100620847).

4. U.S. NRC, Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995 (ADAMS Accession No. ML003740058).

5. Nuclear Energy Institute, NEI 94-01, Revision 0, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," dated July 21, 1995 (ADAMS Accession No. ML11327A025).

6. Electric Power Research Institute (EPRI) Report No. 1009325, Revision 2, "Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals," dated August 2007 (ADAMS Accession No. ML072970208).

7. U.S. NRC, Regulatory Guide 1.174, Revision 3, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," dated January 2018 (ADAMS Accession No. ML17317A256).

8. U.S. NRC, Regulatory Guide 1.177, Revision 1, "An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications," dated May 2011 (ADAMS Accession No. ML100910008).

9. U.S. NRC, "Final Safety Evaluation for Nuclear Energy Institute (NEI) Topical Report (TR) 94-01, Revision 2, 'Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J' and Electric Power Research Institute (EPRI) Report No. 1009325, Revision 2, August 2007, 'Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals,'" dated June 25, 2008 (ADAMS Accession No. ML081140105).

10. U.S. NRC, "Final Safety Evaluation of Nuclear Energy Institute (NEI) Report, 94-01 Revision 3, 'Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J,'" dated June 8, 2012 (ADAMS Accession No. ML121030286).

11. U.S. NRC, Regulatory Issue Summary 2007-06, "Regulatory Guide 1.200 Implementation," dated March 22, 2007 (ADAMS Accession No. ML070650428).

12. U.S. NRC, "Wolf Creek Generating Station – Issuance of Amendment Re: Containment Tendon Surveillance Program and Containment Leakage Rate Testing Program" (Amendment No. 152), dated March 17, 2004 (ADAMS Package Accession No. ML040820952).

13. U.S. NRC, Information Notice 1992-20, "Inadequate Local Leak Rate Testing," dated March 3, 1992 (ADAMS Accession No. ML031200473).

14. U.S. NRC, Information Notice 2010-12, "Containment Liner Corrosion," dated June 18, 2010 (ADAMS Accession No. ML100640449).

15. U.S. NRC, Information Notice 2014-07, "Degradation of Leak-Chase Channel Systems for Floor Welds of Metal Containment Shell and Concrete Containment Metallic Liner," dated May 5, 2014 (ADAMS Accession No. ML14070A114).

16. U.S. NRC, Regulatory Issue Summary 2016-07, "Containment Shell or Liner Moisture Barrier Inspection," dated May 9, 2016 (ADAMS Accession No. ML16068A436).

17. U.S. NRC, Licensee Event Report 2020-001-00, “Plant Shutdown Due to Inoperable Containment Purge Isolation Valves,” dated April 1, 2020 (ADAMS Accession No. ML20092N985).

18. U.S. NRC, Regulatory Guide 1.54, Revision 3, “Service level I, II, III, and In-Scope License Renewal Protective Coatings Applied to Nuclear Power Plants,” dated April 2017 (ADAMS Accession No. ML17031A288).

19. U.S. NRC, Regulatory Guide 1.200, Revision 2, “An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities,” dated March 2009 (ADAMS Accession No. ML090410014).

20. Cruse, C. H, Constellation Nuclear, letter to U.S. NRC, “Calvert Cliffs Nuclear Power Plant, Unit No. 1; Docket No. 50-317, Response to Request for Additional Information Concerning the License Amendment Request for a One-Time Integrated Leakage Rate Test Extension,” dated March 27, 2002 (ADAMS Accession No. ML020920100).

21. Nuclear Energy Institute, NEI 05-04, “Process for Performing Internal Events PRA Peer Reviews Using the ASME/ANS PRA Standard,” Appendix X (ADAMS Package Accession No. ML17086A431).

22. U.S. NRC, “Results of Safety/Risk Assessment of Generic Issue 199, ‘Implications of Updated Probabilistic Seismic Hazard Estimates in Central and Eastern United States on Existing Plants,’” dated September 2, 2010 (ADAMS Package Accession No. ML100270582).

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Date: December 7, 2020

SUBJECT: WOLF CREEK GENERATING STATION, UNIT 1 - ISSUANCE OF AMENDMENT NO. 226 RE: EXTENSION OF TYPE A AND TYPE C LEAK RATE TEST FREQUENCIES (EPID L-2020-LLA-0083) DATED DECEMBER 7, 2020

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