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

August 18, 2020

Mr. John Dent, Jr.
Vice President and Chief
Nuclear Officer
Nebraska Public Power District
72676 648A Avenue
P.O. Box 98
Brownville, NE 68321

SUBJECT: COOPER NUCLEAR STATION - ISSUANCE OF AMENDMENT NO. 266 RE: REQUEST FOR EXCEPTION FROM CERTAIN LEAK RATE TESTING INTERVAL REQUIREMENTS (EPID L-2019-LLA-0179)

Dear Mr. Dent:

The U.S. Nuclear Regulatory Commission (the Commission) has issued the enclosed Amendment No. 266 to Renewed Facility Operating License No. DPR-46 for Cooper Nuclear Station (Cooper). The amendment consists of changes to the Technical Specifications (TSs) in response to your application dated August 19, 2019.

The amendment revises Cooper TS 5.5.12, "Primary Containment Leakage Rate Testing Program," to allow for an exception from certain leak rate testing interval requirements of the program.

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

Sincerely,

/RA/

Thomas J. Wengert, Senior Project Manager
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-298

Enclosures:
1. Amendment No. 266 to DPR-46
2. Safety Evaluation

cc: Listserv

NEBRASKA PUBLIC POWER DISTRICT

DOCKET NO. 50-298

COOPER NUCLEAR STATION

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 266
Renewed License No. DPR-46

1. The U.S. Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Nebraska Public Power District (the licensee), dated August 19, 2019, 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, 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. DPR-46 is hereby amended to read as follows:

    (2) Technical Specifications

    The Technical Specifications contained in Appendix A as revised through Amendment No. 266, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. The license amendment is effective as of its date of issuance and shall be implemented within 60 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Jennifer L. Dixon-Herrity

Digitally signed by Jennifer L. Dixon-Herrity
Date: 2020.08.18 10:03:18 -04'00'

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 No. DPR-46 and the Technical Specifications

Date of Issuance: August 18, 2020

ATTACHMENT TO LICENSE AMENDMENT NO. 266

RENEWED FACILITY OPERATING LICENSE NO. DPR-46

COOPER NUCLEAR STATION

DOCKET NO. 50-298

Replace the following pages of the Renewed Facility Operating License No. DPR-46 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
-3-	-3-

Technical Specifications

REMOVE	INSERT
5.0-17	5.0-17
5.0-18	5.0-18

(5) 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 operation of the facility.

C. This license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations in 10 CFR Chapter I: Part 20, Section 30.34 of Part 30, Section 40.41 of Part 40, Sections 50.54 and 50.59 of Part 50, and Section 70.32 of Part 70; 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 licensee is authorized to operate the facility at steady state reactor core power levels not in excess of 2419 megawatts (thermal).

(2) Technical Specifications

The Technical Specifications contained in Appendix A as revised through Amendment No. 266, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

(3) Physical Protection

The licensee shall fully implement and maintain in effect all provisions of the Commission-approved physical security, training and qualification and safeguards contingency plans including amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR 27817 and 27822) and to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The combined set of plans, which contain Safeguards Information protected under 10 CFR 73.21, are entitled: "Cooper Nuclear Station Safeguards Plan," submitted by letter dated May 17, 2006.

NPPD shall fully implement and maintain in effect all provisions of the Commission-approved cyber security plan (CSP), including changes made pursuant to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The NPPD CSP was approved by License Amendment No. 238 as supplemented by changes approved by License Amendments 244 and 249.

(4) Fire Protection

NPPD shall implement and maintain in effect all provisions of the approved fire protection program that comply with 10 CFR 50.48(a) and 10 CFR 50.48(c), as specified in the license amendment request dated April 24, 2012 (and supplements dated July 12, 2012, January 14, 2013, February 12, 2013, March 13, 2013, June 13, 2013, December 12, 2013, January 17, 2014, February 18, 2014, and April 11, 2014), and as approved in the safety evaluation dated April 29, 2014. Except where NRC approval for changes or deviations is required by 10 CFR 50.48(c), and provided no other regulation, technical specification, license condition or requirement would require prior NRC approval, the licensee may make changes to the fire protection program without prior approval of the Commission if

5.5 Programs and Manuals

5.5.12 Primary Containment Leakage Rate Testing Program (continued)

5. Exemption from Section III.B of 10CFR Part 50, Appendix J, Option B, to allow the contribution from Main Steam Pathway (Main Steam lines and Main Steam inboard drain line) leakage to be excluded from the sum of the leakage rates from Type B and Type C tests (September 14, 2009).

6. Exception to NEI 94-01, "Industry Guideline for Implementing Performance-Based Option of 10 CFR 50, Appendix J," to allow testing of Type C Residual Heat Removal heat exchanger relief valves and their associated Type B testable discharge flange tests at the same frequency as the visual examination, seat leakage testing, and set pressure testing performed for these valves under the requirements of the Inservice Testing Program per 10 CFR 50.55a(f).

b. The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 58.0 psig. The containment design pressure is 56.0 psig.

c. The maximum allowable containment leakage rate, L_a, at P_a, shall be 0.635% 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 C tests and $\leq 0.75\ L_a$ for Type A tests.

2. Air lock testing acceptance criteria are:

a. Overall air lock leakage rate is $\leq$ 12 scfh when tested at $\geq P_a$.

b. Overall air lock leakage rate is $\leq$ 0.23 scfh when tested at $\geq$ 3.0 psig.

e. The provisions of SR 3.0.2 do not apply to the test frequencies specified in the Primary Containment Leakage Rate Testing Program.

f. The provisions of SR 3.0.3 are applicable to the Primary Containment Leakage Rate Testing Program.

5.5.13 Control Room Envelope Habitability Program

A Control Room Envelope (CRE) Habitability Program shall be established and implemented to ensure that CRE habitability is maintained such that, with an OPERABLE Control Room Emergency Filter (CREF) System, CRE occupants can

(continued)

5.5 Programs and Manuals

5.5.13 Control Room Envelope Habitability Program (continued)

control the reactor safely under normal conditions and maintain it in a safe condition following a radiological event, hazardous chemical release, or a smoke challenge. The program shall ensure that adequate radiation protection is provided to permit access and occupancy of the CRE under design basis accident (DBA) conditions without personnel receiving radiation exposures in excess of either (a) 5 rem whole body or its equivalent to any part of the body for the duration of the loss-of-coolant accident, or (b) 5 rem total effective dose equivalent (TEDE) for the duration of the fuel handling accident. The program shall include the following elements:

a. The definition of the CRE and CRE boundary.

b. Requirements for maintaining the CRE boundary in its design condition including configuration control and preventive maintenance.

c. Requirements for (i) determining the unfiltered air inleakage past the CRE boundary into the CRE in accordance with the testing methods and at the Frequencies specified in Sections C.1 and C.2 of Regulatory Guide 1.197, "Demonstrating Control Room Envelope Integrity at Nuclear Power Reactors," Revision 0, May 2003, and (ii) assessing CRE habitability at the Frequencies specified in Sections C.1 and C.2 of Regulatory Guide 1.197, Revision 0. No exceptions to Sections C.1 and C.2 of Regulatory Guide 1.197, Revision 0, are proposed.

d. Measurement, at designated locations, of the CRE pressure relative to all external areas adjacent to the CRE boundary during the pressurization mode of operation by the CREF System, operating at the flow rate required by the Ventilation Filter Testing Program, at a Frequency of 24 months. The results shall be trended and used as part of the periodic assessment of the CRE boundary.

e. The quantitative limits on unfiltered air inleakage into the CRE. These limits shall be stated in a manner to allow direct comparison to the unfiltered air inleakage measured by the testing described in paragraph c. The unfiltered air inleakage limit for radiological challenges is the inleakage flow rate assumed in the licensing basis analyses of DBA consequences. Unfiltered air inleakage limits for hazardous chemicals must ensure that exposure of CRE occupants to these hazards will be within the assumptions in the licensing basis.

f. The provisions of SR 3.0.2 are applicable to the Frequencies for assessing CRE habitability, determining CRE unfiltered air inleakage, and measuring CRE pressure and assessing the CRE boundary as required by paragraphs c and d, respectively.

(continued)

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 266 TO

RENEWED FACILITY OPERATING LICENSE NO. DPR-46

NEBRASKA PUBLIC POWER DISTRICT

COOPER NUCLEAR STATION

DOCKET NO. 50-298

1.0 INTRODUCTION

By application dated August 19, 2019 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML19238A065), Nebraska Public Power District (the licensee) requested changes to the Technical Specifications (TSs) for Cooper Nuclear Station (Cooper, CNS).

The proposed changes would revise Cooper TS 5.5.12, “Primary Containment Leakage Rate Testing Program,” to allow for an exception from certain leak rate testing interval requirements of the program. Specifically, the licensee requested that the Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, “Domestic Licensing of Production and Utilization Facilities,” Appendix J, “Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors,” Option B, “Performance-Based Requirements,” Type C leak rate testing of the residual heat removal (RHR) system heat exchanger relief valves (RV), RHR-RV-20RV and RHR-RV-21RV, and their associated Type B leak rate testing of the RV discharge flanges be performed at the same frequency as the visual examination, seat leakage testing, and set pressure testing performed for these valves under the requirements of the inservice testing (IST) program, as required by 10 CFR 50.55a, “Codes and standards,” paragraph (f), “Preservice and inservice testing requirements.”

2.0 REGULATORY EVALUATION

2.1 System Description

2.1.1 Primary Containment

The Cooper Updated Safety Analysis Report (USAR), Revision 29 (ADAMS Package Accession No. ML19137A206), describes the primary containment as a General Electric Mark I pressure suppression system design housing the reactor vessel, the reactor coolant recirculation loops, and other branch connections of the reactor primary system. The primary containment consists of a drywell, a suppression chamber (torus) that stores a large volume of water (suppression

pool), a connecting vent system between the drywell and the suppression pool, isolation valves, a primary containment isolation system, and a vacuum relief system. Additional equipment, including portions of the emergency core cooling system, is located within the primary containment and provides services to the primary containment. In the event of a process system piping failure within the drywell, reactor water and steam will be released into the drywell gas space. The resulting increased drywell pressure forces a mixture of air, steam, and water through the vent system into the suppression pool. The steam condenses rapidly in the suppression pool, resulting in rapid pressure reduction in the drywell. Air transferred during reactor blowdown to the suppression chamber pressurizes the chamber and subsequently is vented to the drywell through the vacuum relief system as the pressure in the drywell drops below that in the suppression chamber. Steam remaining in the drywell can be condensed by the containment spray system, as described in Cooper USAR Section IV-8.5.3, "Containment Cooling Subsystems."

Cooling systems are provided to remove heat from the suppression pool to provide for continuous cooling of the primary containment under the postulated design-basis accident conditions for which the primary containment is assumed to be functional. Isolation valves are provided to ensure containment of radioactive materials within the primary containment, which might be released from the reactor to the containment during an accident. Other service equipment is provided to maintain the containment within its design parameters during normal operation.

2.1.2 Low-Pressure Coolant Injection Subsystem

In the event of low-water level in the reactor or high pressure in the containment drywell, the low-pressure coolant injection mode of operation of the RHR will pump water into the reactor vessel in time to flood the reactor core and limit fuel-clad temperature.

The low-pressure coolant injection operation consists of initiation of both subsystems (loops), each of which consists of two RHR alternating current motor-driven centrifugal pumps, taking water from the suppression pool and pumping it into their corresponding reactor recirculation loops, as described in Cooper USAR Sections VI-5.2.7, "Emergency Core Cooling Systems," and VII-4.5.5, "Low Pressure Coolant Injection Control and Instrumentation." The water enters the reactor vessel through the associated recirculation system loops to the jet pumps to restore the water level inside the core shroud to the height of the jet pump nozzle. The low-pressure coolant injection operation includes using associated valves, controls, instrumentation, and pump accessories.

2.2 The Licensee's Proposed TS Changes

In the license amendment request (LAR), the licensee proposed to change Cooper TS 5.5.12 by adding a new exception "6" to the description of the guidance used. TS 5.5.12 currently states, in part:

> a. A program shall establish 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 Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September, 1995, as modified by the following exceptions:

The licensee's proposed change would add a sixth exception to TS 5.5.12, which would state:

> 6. Exception to NEI [Nuclear Energy Institute] 94-01, "Industry Guideline for Implementing Performance-Based Option of 10 CFR 50, Appendix J," to allow testing of Type C Residual Heat Removal heat exchanger relief valves and their associated Type B testable discharge flange tests at the same frequency as the visual examination, seat leakage testing, and set pressure testing performed for these valves under the requirements of the Inservice Testing Program per 10 CFR 50.55a(f).

This proposed exception to the adopted testing program guidance is intended to allow containment isolation valves RHR-RV-20RV and RHR-RV-21RV and their associated discharge flanges to have their 10 CFR Part 50, Appendix J required leakage rate testing be performed at the same interval as the IST program required lift setpoint and seat leakage tests or, effectively, the test interval could be extended to 8 years.

2.3 Regulatory Requirements

In the LAR, the licensee requested a change to the Renewed Facility Operating License for Cooper in accordance with 10 CFR 50.90, "Application for amendment of license, construction permit, or early site permit."

The regulation in 10 CFR 50.36(c)(5), "Administrative controls," requires, in part, the inclusion of administrative controls in TSs that are necessary to ensure operation of the facility in a safe manner. The licensee requested a change to the "Administrative Controls" section of the Cooper TSs.

In accordance with 10 CFR 50.54(o), 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 includes two options: "Option A – Prescriptive Requirements" and "Option B – Performance-Based Requirements," either of which can be used to meet 10 CFR Part 50, Appendix J requirements.

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 TSs and (b) integrity of the containment structure is maintained during the service life of the containment.

Section V.B.3 of 10 CFR Part 50, Appendix J, Option B, requires a licensee to develop a performance-based leakage-testing program using a U.S. Nuclear Regulatory Commission (NRC, the Commission) regulatory guide (RG) or other implementation document and referencing this document in the plant TSs. A submittal for TS revisions must also contain justification, including supporting analyses, if the licensee proposed to deviate from methods approved by the NRC and endorsed in an RG, such as RG 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995 (ADAMS Accession No. ML003740058).

Option B of 10 CFR Part 50, Appendix J specifies performance-based requirements and criteria for preoperational and subsequent leakage rate testing. These requirements are met by

- Type A tests to measure the containment system overall integrated leakage rate;
- Type B pneumatic tests to detect and measure local leakage rates across pressure retaining, leakage-limiting boundaries such as penetrations; and
- Type C pneumatic tests to measure containment isolation valves leakage rates.

Type B and Type C tests are performed based on the safety significance and historical performance of each boundary and isolation valve to ensure the integrity of the overall containment system as a barrier to fission product release.

The regulation in 10 CFR 50.55a contains the component IST requirements that, in conjunction with the requirements of 10 CFR Part 50, Appendix J, ensure the continued leak-tight and structural integrity of the containment during its service life.

The regulation in 10 CFR 50.65, "Requirements for monitoring the effectiveness of maintenance at nuclear power plants," paragraph (a)(1), states, 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."

2.4 Regulatory Guidance

RG 1.163, Revision 0, endorses NEI 94-01, Revision 0, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J" (ADAMS Accession No. ML11327A025), which provides methods for complying with 10 CFR Part 50, Appendix J, Option B, and includes provisions that address the extension of the performance-based Type A test interval for up to 10 years, based upon two consecutive successful tests. NEI 94-01 also allows for establishing Type B and Type C testing performance-based intervals out to a maximum of 120 months. However, RG 1.163, Regulatory Position C.2. did not endorse Type C testing intervals beyond 60 months at that time due to "uncertainties (particularly unquantified leakage rates for test failures, repetitive/common mode failures, and aging effects) in historical Type C component performance data…."

3.0 TECHNICAL EVALUATION

3.1 Existing Testing Regime for RHR-RV-20RV and RHR-RV-21RV

In the LAR, the licensee provided a diagram depiction and narrative description of the valve location and physical installation details. These are 1-inch (nominal) diameter RVs protecting the RHR heat exchanger pressure vessel shell side (RHR) from over-pressurization. Each relief valve discharges to a common header routed to connect with the turbine-driven high-pressure coolant injection pump turbine exhaust line that terminates in a submerged sparger in the suppression pool. The high-pressure coolant injection pump turbine exhaust line has vacuum breakers that open to let suppression chamber atmosphere into the exhaust line to prevent suppression pool water from being drawn into the exhaust line when steam exhaust is terminated. As such, these lines, including the RVs, present a potential containment

atmosphere leakage pathway into the secondary containment and are thus subject to 10 CFR Part 50, Appendix J leakage rate testing.

In the LAR, the licensee described the current testing of these relief valves, as follows:

> in order to perform the IST or Appendix J testing on the relief valves, they must be removed from the system. This is typically done by removing the relief valve assembly from the flange on the inlet side to the testable flange on the discharge side (reference Enclosure 1 [of the LAR]). In general, prior to the removal of each RV assembly and after re-installation of each RV assembly, a Type B Appendix J leak test is performed at the respective testable discharge flange for each valve. Each testable discharge flange utilizes a two O-ring design in which the air space between these two O-rings is pressurized through a 1/4" [inch] test fitting (reference Enclosure 3 [of the LAR]). The completion of this Type B test will ensure that external leakage from this Type B flange joint is within the acceptance criteria of the Appendix J Program. It has also been a challenge to establish Option B extended frequencies on the discharge flange tests due to the required IST relief valve testing, often resulting in Appendix J Type B testing being performed more frequently than is required to ensure reliable performance of the testable discharge flanges.

3.2 Historical Testing Results for RHR-RV-20RV and RHR-RV-21RV

In the LAR, the licensee provided the following information regarding the historical 10 CFR Part 50, Appendix J, testing results for the RVs and discharge flanges:

> Table 1 represents the Appendix J as-found test results for RHR-RV-20RV and its associated discharge test flange for the past seven outages and Table 2 represents the Appendix J as-found test results for RHR-RV-21RV and its associated discharge test flange for the past seven outages. The current CNS NEI 94-01, Revision 0, administrative limit (current CNS operability limit) for each discharge flange is 1.0 standard cubic feet per hour (scfh) and for each relief valve is 5.0 scfh. To put these values in perspective, the total minimum path leakage (L_a) allowed at CNS is 317.40 scfh during power operations.

Table 1: As-Found (AF) Appendix J Test History for RHR-RV-20RV and Associated Discharge Flange

Date of Test	Outage	AF test of RHR-RV-20RV Discharge Flange (scfh)	AF test of RHR-RV-20RV Valve (scfh)
4/25/2008	RE24	0.005	
4/27/2008	RE24		0.0267
9/30/2009	RE25	0.0042	
10/4/2009	RE25		0.08
4/3/2011	RE26	0.2405	
4/6/2011	RE26		0.1725
10/15/2012	RE27	0.0042	
10/20/2012	RE27		0.0355
10/17/2014	RE28	0.0042	
10/18/2014	RE28		0.0055

10/1/2016	RE29	0.42	
10/3/2016	RE29		0.005
10/25/2018	RE30	0.0042	
11/2/2018	RE30		0.0042

Note: The administrative limit (current CNS operability limit) for the discharge flange is 1.0 scfh and for the relief valve is 5.0 scfh. The allowed CNS L_a value is 317.40 scfh.

Table 2: As-Found (AF) Appendix J Test History for RHR-RV-21RV and Associated Discharge Flange			
Date of Test	Outage	AF test of RHR-RV-21RV Discharge Flange (scfh)	AF test of RHR-RV-21RV Valve (scfh)
4/12/2008	RE24	0.0042	
4/18/2008	RE24		0.005
10/16/2009	RE25	0.0042	
10/18/2009	RE25		0.51
3/15/2011	RE26	0.0042	
3/30/2011	RE26		0.55
11/7/2012	RE27	0.0042	
11/17/2012	RE27		0.07
9/28/2014	RE28	0.0042	
10/05/2014	RE28		0.2092
10/21/2016	RE29	0.0042	
10/22/2016	RE29		0.0386
10/5/2018	RE30	0.01	
10/12/2018	RE30		0.013

Note: The administrative limit (current CNS operability limit) for the discharge flange is 1.0 scfh and for the relief valve is 5.0 scfh. The allowed CNS L_a value is 317.40 scfh.

3.3 Summary of Technical Evaluation

The AF testing results from the past seven refueling outages presented in the LAR and summarized above show the leakage attributed to RHR-RV-20RV and RHR-RV-21RV and their associated discharge flanges to at all times be less than half of the administrative limits and, usually, significantly less than that. Additionally, in combination, the leakage attributed to these valves and flanges contributes to much less than 1 percent of the allowable leakage of the primary containment, L_a. Based on these results, the NRC staff determined that these valves and flanges have made a negligible contribution to primary containment leakage potential over the past seven refueling outages.

The 10 CFR Part 50, Appendix J testing results over the past seven refueling outages show that a negligible leakage potential develops over an operating cycle with respect to RHR-RV-20RV and RHR-RV-21RV and their associated discharge flanges. The RVs are likely not actuating or cycling other than during the periodic testing and, therefore, are likely not experiencing significant wear or degradation. As described in the LAR, the testing program guidance requires that, should a valve fail the AF 10 CFR Part 50, Appendix J test, the cause would be determined, and the valve would have to pass two subsequent tests on a base interval not to exceed 30 months for the valve to again be placed on the 10 CFR Part 50, Appendix J extended test interval. Therefore, the NRC staff concludes that the proposed change to TS 5.5.12 will still

provide reasonable assurance that the subject components remain capable of fulfilling their intended functions. The change to TS 5.5.12 is therefore consistent with the requirements of 10 CFR 50.65(a)(1).

Based on its regulatory and technical evaluations, summarized above, the NRC staff finds that the licensee has demonstrated that the proposed exception to Cooper TS 5.5.12 establishes intervals for the 10 CFR Part 50, Appendix J tests of RHR-RV-20RV and RHR-RV-21RV and their associated discharge flanges that, based on the safety significance and historical performance of these valves and flanges, ensure the integrity of the overall containment system as a barrier to fission product release. This change only affects the licensee's TS Administrative Controls requirements and the TSs will continue to specify the remedial measures to be taken if one of its requirements is not satisfied. In addition, the change to add the exception to RG 1.163, allows extending the maximum Type C test interval of RHR-RV-20RV and RHR-RV-21RV to 8 years to allow closer coordination of the 10 CFR Part 50, Appendix J testing with the IST program requirements. The NRC staff also notes that a later revision of NEI 94-01 (i.e., Revision 3-A), dated July 2012 (ADAMS Accession No. ML12221A202), allows for extending containment isolation valve testing intervals to 75 months based upon completion of two consecutive periodic AF tests where the results of each test are within a licensee's allowable administrative limits. Therefore, the NRC staff concludes that, with the addition of the proposed exception to Cooper TS 5.5.12 regarding the primary containment leakage rate testing program, the Cooper TSs will continue to include administrative controls necessary to assure operation of the facility in a safe manner, and thus, the proposed change is acceptable.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Nebraska State official was notified of the proposed issuance of the amendment on July 9, 2020. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes requirements with respect to the 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 published in the *Federal Register* on December 3, 2019 (84 FR 66231), and there has been no public comment on such finding. 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.

Principal Contributor: J. Bettle

Date: August 18, 2020

SUBJECT: COOPER NUCLEAR STATION - ISSUANCE OF AMENDMENT NO. 266 RE: REQUEST FOR EXCEPTION FROM CERTAIN LEAK RATE TESTING INTERVAL REQUIREMENTS (EPID L-2019-LLA-0179) DATED AUGUST 18, 2020

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