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December 28, 2021
L-21-259

10 CFR 50.55a

ATTN: Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

SUBJECT:
Perry Nuclear Power Plant, Unit No. 1
Docket No. 50-440, License No. NPF-58
10 CFR 50.55a Requests Number VR-10 and VR-11, Inservice Valve Testing

In accordance with the provisions of 10 CFR 50.55a(z)(1), Energy Harbor Nuclear Corp. hereby requests Nuclear Regulatory Commission (NRC) staff approval for two proposed alternatives that provide an acceptable level of quality and safety (Enclosure A and B).

The enclosures identify the proposed alternatives, the affected components, the applicable code requirements, the reason for the request, and the basis for use.

The requests are for use during the fourth 10-year IST interval, which began on May 18, 2019 and is scheduled to expire on May 17, 2029. Energy Harbor Nuclear Corp. is requesting approval of the proposed 10 CFR 50.55a request by January 31, 2023.

There are no regulatory commitments contained in this submittal. If there are any questions or if additional information is required, please contact Mr. Phil H. Lashley, Manager, Fleet Licensing, at (330) 696-7208.

Sincerely,

* Elliott

Rod L. Penfield

* Christopher M. Elliott, General Plant Manager, as designated by Rod L. Penfield

Enclosures:

A. Perry Nuclear Power Plant, 10 CFR 50.55a Request VR-10, Revision 0
B. Perry Nuclear Power Plant, 10 CFR 50.55a Request VR-11, Revision 0

cc: NRC Region III Administrator
NRC Resident Inspector
NRC Project Manager

Enclosure A
L-21-259

Perry Nuclear Power Plant
10 CFR 50.55a Request Number VR-10, Revision 0

(7 pages follow)

Proposed Alternative
In Accordance with 10 CFR 50.55a(z)(1)

--Alternative Provides Acceptable Level of Quality and Safety—

1. ASME Code Components Affected

Component ID	Component Description	Code Class	Valve Category
1C41-F006	Standby Liquid Control Injection Outboard Check Valve	1	C*
1C41-F007	Standby Liquid Control Injection Inboard Check Valve	1	C*
1E12-F008	Shutdown Cooling OTBD Suction Isolation Valve	1	A
1E12-F009	Shutdown Cooling INBD Suction Isolation Valve	1	A
1E12-F041A	Low Pressure Coolant Injection A Injection Check Valve	1	C*
1E12-F042A	Low Pressure Coolant Injection A Injection Valve	1	A
1E12-F041B	Low Pressure Coolant Injection B Injection Check Valve	1	C*
1E12-F042B	Low Pressure Coolant Injection B Injection Valve	1	A

*The four PIVs identified as Category C valves also have the requirement of a Category A valve classification “for which seat leakage is limited to a specific maximum amount in the closed position to fulfill their required function(s)” and, therefore, are subject to the requirements of ASME OM Code, Subsection ISTC-3630.

2. Applicable Code Edition and Addenda

American Society of Mechanical Engineers (ASME) Code for Operation and Maintenance of Nuclear Power Plants (OM Code), 2012 Edition.

3. Applicable Code Requirements

American Society of Mechanical Engineers, Operation and Maintenance of Nuclear Power Plants, 2012 edition, Subsection ISTC-3522, “Category C Check Valves,”(a) states in part:

> During operation at power, each check valve shall be exercised or examined in a manner that verifies obturator travel by using the methods in para. ISTC-5221. Each check valve exercise test shall include open and close tests.

American Society of Mechanical Engineers, Operation and Maintenance of Nuclear Power Plants, 2012 edition, Subsection ISTC-3522 “Category C Check Valves” (c) states:

If exercising is not practicable during operation at power and cold shutdowns, it shall be performed during refueling outages."

American Society of Mechanical Engineers, Operation and Maintenance of Nuclear Power Plants, 2012 edition, Subsection ISTC-3630, "Leakage Rate for Other Than Containment Isolation Valves," states:

Category A valves with a leakage requirement not based on an Owner's 10 CFR 50, Appendix J program, shall be tested to verify their seat leakages are within acceptable limits. Valve closure before seat leakage testing shall be by using the valve operator with no additional closing force applied.

American Society of Mechanical Engineers, Operation and Maintenance of Nuclear Power Plants, 2012 edition, Subsection ISTC-3630(a), "Frequency," states:

Tests shall be conducted at least once every 2 yr.

4. Reason for Request

In accordance with 10 CFR 50.55a, "Codes and Standards," paragraph (z)(1), "Alternatives to codes and standards requirements," an alternative is requested from the requirements of ASME OM Code ISTC-3630(a) for the 8 subject pressure isolation valves (PIVs) listed in Section 1. Additionally, for two of the subject PIVs (1C41-F006 and 1C41-F007), an alternative is requested from the requirements of ISTC-3522. The basis is that the proposed alternative would provide an acceptable level of quality and safety while substantially lowering cumulative radiation exposure (CRE).

ISTC-3630 requires leakage rate testing for PIVs be performed at least once every two years. PIVs are not directly included in the scope of performance-based testing as provided for in 10 CFR 50 Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Reactors," Option B, "Performance-Based Requirements." Some of these valves may be containment isolation valves (CIVs) but are out of the Appendix J scope considering they are water sealed. Approval of this alternative will allow PIV testing at Perry Nuclear Power Plant (PNPP) to be done on a performance-based frequency.

The PNPP Technical Specification 5.5.12, Primary Containment Leakage Rate Testing Program, contains a requirement to establish a leakage rate testing program in accordance with the guidelines contained in NEI Topical Report NEI 94-01, Revision 3-A, with conditions and limitations in NEI 94-01, Revision 2-A.

The Option B alternative for CIVs states that licensees should be allowed to adopt cost effective methods for complying with regulatory requirements. NEI 94-01, Revision 3-A, describes risk-informed basis for the extended test intervals under

Option B. The CIVs that have demonstrated good performance by the successful completion of two consecutive leakage rate tests over two consecutive cycles may increase their test frequencies. It also states that if the component does not fail within two operating cycles, further failures appear to be governed by the random failure rate of the component. NEI 94-01 presents the results of a comprehensive

risk analysis, with the conclusion that “the risk impact associated with increasing [leak rate] test interval is negligible (i.e., less than 0.1 percent of total risk).”

The eight PIV valves identified in this alternative are tested with water pressurized between 1040 and 1060 PSIG as required by Surveillance Requirement (SR) 3.4.6.1. This request is intended to provide for a performance-based scheduling of PIV tests at PNPP. The reason for requesting this alternative is to reduce required resources and over 600 mrem dose received for testing each outage.

NUREG-0933, “Resolution of Generic Safety Issues,” Issue 105, “Interfacing Systems LOCA at LWRs,” establishes the need for PIV leak rate testing based on three pre-1985 historical failures of applicable valves industry wide. All three failures were due to human error in operation or maintenance. None of the failures were due to in-service equipment degradation.

5. <u>Proposed Alternative and Basis for Use</u>

PNPP is requesting an alternative from ASME OM Code, Subsection ISTC-3630(a) for all eight subject valves, and an alternative from ISTC-3522, specifically, for the exercise close testing requirements for valves,1C41-F006 and 1C41-F007.

The proposed alternative from ASME OM Code, Subsection ISTC-3630(a) for the eight PIVs would establish the specific test interval for each PIV based on its historical performance and would be consistent with the containment isolation valve testing process under 10 CFR 50, Appendix J, Option B. Performance-based scheduling of the PIVs will be controlled like the methods described in NEI 94-01, Revision 3-A. The frequency of testing would range from every refueling outage to every third refueling outage dependent on valve performance. Valves that have demonstrated good performance for two consecutive cycles may have their test interval extended up to 75 months. If a valve fails the leak rate test, the interval will be reduced back to the two-year interval until the valve has re-established good performance.

The primary basis for this request is the historical performance of the PIVs. The historical performance of the eight subject valves over the past six consecutive refueling outages is compiled in the attached table, Table 1 - Historical Leak Rate Test Performance. The historical test performance for the eight PIVs consistently demonstrate successful leakage test results.

Valve functional capability is demonstrated by the open and close exercise test for 1E12-F041A and 1E12-F041B, and the requirements of Mandatory Appendix III,

Paragraph III-3300, Inservice Test, and Section III-3600, MOV [Motor-Operated Valve] Exercising Requirements, for 1E12-F008, 1E12-F009, 1E12-F042A, and 1E12-F042B. These tests are separate and distinct from the PIV testing and are performed in accordance with the requirements of the ASME OM Code. For valves 1C41-F006 and 1C41-F007, the proposed alternative from ISTC-3522 exercise close testing would credit the PIV leak rate testing and will be on the same frequency as the PIV leak rate testing. The exercise open testing of 1C41-F006 and 1C41-F007 will be retained on a refueling outage interval and will be demonstrated by the ability to flow 41.9 gpm from the standby liquid control system to the reactor pressure vessel. Considering the historical performance of the PIVs, it is clear the eight subject valves are exhibiting the required obturator movement to close and remain closed.

NEI 94-01, Revision 3-A, is not the sole basis for this request, given that it does not address seat leakage testing with water. The NEI document is being cited as an approach analogous to the requested alternative method. If the proposed alternative is authorized and the valves exhibit good performance, the PIV test frequency will be altered as described in NEI 94-01, Revision 3-A, so that testing is not required every refueling outage.

The extension of test frequencies will be consistent with the guidance provided for Appendix J, Type C leak rate tests as detailed in NEI 94-01, Revision 3-A, Paragraph 10.2.3.2, “Extended Test Interval,” which states:

> Test intervals for Type C valves may be increased based upon completion of two consecutive periodic as-found Type C tests where the result of each test is within a licensee’s allowable administrative limits. Elapsed time between the first and last tests in a series of consecutive passing tests used to determine performance shall be 24 months or the nominal test interval, for example refueling cycle, for the valve prior to implementing Option B to Appendix J. Intervals for Type C testing may be increased to a specific value in a range of frequencies from 30 months up to a maximum of 75 months. Test intervals for Type C valves should be determined by a licensee in accordance with Section 11.0.

6. Duration of Proposed Alternative

The proposed alternative is requested for use during the PNPP fourth 10-year in-service test interval, scheduled to end May 17, 2029.

7. Precedent

A similar alternative to allow PIV testing under a performance-based testing approach like that established under 10 CFR 50, Appendix J, Option B has been approved as specified in the following:

NRC letter to Exelon Generation Company, LLC (Exelon), "LaSalle County Station, Units 1 and 2 – Request from the Requirements of the ASME Code Related to Pressure Isolation Valve Testing Frequency (EPID L-2019-LLR 0062)," (ML19217A306), dated September 10, 2019.

Table 1 – Historical Leak Rate Test Performance

Historical Leak Rate Test Performance				
Component	**Date of Test**	**Measured Value (gpm)**	**Required Action Limit (gpm)**	**Comments**
1C41-F006	04/28/2011	0.00	0.75	Sat
	03/21/2013	0.0053	0.75	Sat
	03/19/2015	0.003	0.75	Sat
	03/12/2017	0.0066	0.75	Sat
	03/18/2019	0.0010	0.75	Sat
	03/13/2021	0.00	0.75	Sat
1C41-F007	04/28/2011	0.00	0.75	Sat
	03/21/2013	0.00	0.75	Sat
	03/19/2015	0.00	0.75	Sat
	03/12/2017	0.00	0.75	Sat
	03/18/2019	0.00	0.75	Sat
	03/13/2021	0.007	0.75	Sat
1E12-F008	05/07/2011	1.069	5.0	Sat
	03/29/2013	1.0	5.0	Sat
	03/21/2015	1.0	5.0	Sat
	03/19/2017	1.25	5.0	Sat
	03/22/2019	1.23	5.0	Sat
	03/25/2021	1.02	5.0	Sat
1E12-F009	05/07/2011	0.417	5.0	Sat
	03/29/2013	0.24	5.0	Sat
	03/21/2015	0.315	5.0	Sat
	03/19/2017	0.00	5.0	Sat
	03/22/2019	0.64	5.0	Sat
	03/25/2021	0.69	5.0	Sat
1E12-F041A	05/19/2011	0.00	5.0	Sat
	04/10/2013	0.00132	5.0	Sat
	04/14/2015	0.00	5.0	Sat
	03/18/2017	0.00	5.0	Sat
	03/25/2019	0.0105	5.0	Sat
	03/19/2021	0.001	5.0	Sat

Table 1 – Historical Leak Rate Test Performance (continued)

Historical Leak Rate Test Performance				
Component	**Date of Test**	**Measured Value (gpm)**	**Required Action Limit (gpm)**	**Comments**
1E12-F042A	05/19/2011	0.00	5.0	Sat
	04/10/2013	0.00	5.0	Sat
	04/14/2015	0.00	5.0	Sat
	03/18/2017	0.008	5.0	Sat
	03/25/2019	0.00	5.0	Sat
	03/19/2021	0.003	5.0	Sat
1E12-F041B	05/08/2011	0.00	5.0	Sat
	04/22/2013	0.002	5.0	Sat
	03/31/2015	0.007	5.0	Sat
	03/22/2017	0.00	5.0	Sat
	03/22/2019	0.00	5.0	Sat
	03/27/2021	0.00	5.0	Sat
1E12-F042B	05/08/2011	0.00066	5.0	Sat
	04/22/2013	0.0045	5.0	Sat
	03/31/2015	0.070	5.0	Sat
	03/22/2017	0.00	5.0	Sat
	03/22/2019	0.143	5.0	Sat
	03/27/2021	0.291	5.0	Sat

Enclosure B
L-21-259

Perry Nuclear Power Plant
10 CFR 50.55a Request Number VR-11, Revision 0

(3 pages follow)

Proposed Alternative Request
In Accordance with 10 CFR 50.55a(z)(1)

--Alternative Provides Acceptable Level of Quality and Safety—

1. **ASME Code Component(s) Affected**

The valves covered by this proposed alternative are those stem-disk separation non-susceptible valves with remote position indication. These valves are within the scope of the American Society of Mechanical Engineers (ASME) Code for Operation and Maintenance of Nuclear Power Plants (OM Code), Subsection ISTC (Inservice Testing of Valves in Light-Water Reactor Nuclear Power Plants) including its mandatory appendices and their verification methods and frequencies, in accordance with regulatory requirements.

A listing of the valves requiring position indication and testing in accordance with ISTC-3700 (Position Verification Testing) was submitted by correspondence to the Commission dated February 13, 2020 (ADAMS Accession No. ML20045E972) as part of the fourth Inservice Testing Program update and is maintained up to date as required by ASME Code.

2. **Applicable Code Edition and Addenda**

American Society of Mechanical Engineers (ASME) Code for Operation and Maintenance of Nuclear Power Plants (OM Code), 2012 Edition

3. **Applicable Code Requirement**

American Society of Mechanical Engineers, Operation and Maintenance of Nuclear Power Plants, 2012 edition, Subsection ISTC-3700, Position Verification Testing states:

> Valves with remote position indicators shall be observed locally at least once every 2-yr to verify that valve operation is accurately indicated. Where practicable, this local observation should be supplemented by other indications such as use of flow meters or other suitable instrumentation to verify obturator position. These observations need not be concurrent. Where local observation is not possible, other indications shall be used for verification of valve operation.

The supplemental indication portion of ISTC-3700 is made mandatory by 10 CFR 50.55a(b)(3)(xi),”OM condition: Valve Position Indication,” states:

> When implementing paragraph ISTC-3700, “Position Verification Testing,” in the ASME OM Code, 2012 Edition through the latest edition and

addenda of the ASME OM Code incorporated by reference in paragraph (a)(1)(iv) of this section[10 CFR 50.55a], licensees shall verify that valve operation is accurately indicated by supplementing valve position indicating lights with other indications, such as flow meters or other suitable instrumentation to provide assurance of proper obturator position for valves with remote position indication within the scope of Subsection ISTC including its mandatory appendices and their verification methods and frequencies.

4. **Reason for Request**

Pursuant to 10 CFR 50.55a, *Codes and standards,* paragraph (z)(1), an alternative is proposed to the requirement of ASME OM Code ISTC-3700. The position verification with supplemental position indication (SPI) requires the valves to be exercised in the open and closed direction and the valve's position verified by other indications such as the use of flow meters or other suitable instrumentation to verify obturator position.

Code Case OMN-28, "Alternative Valve Position Verification Approach to Satisfy ISTC-3700 for Valves Not Susceptible to Stem-Disk Separation," has been determined to satisfy the valve position verification requirements in ASME OM Code, Subsection ISTC, paragraph ISTC-3700, for valves that are not susceptible to stem-disk separation.

5. Proposed Alternative and Basis for Use

In lieu of compliance with ISTC-3700, Perry Nuclear Power Plant (PNPP) proposes to implement Code Case OMN-28 on the basis that it provides an acceptable level of quality and safety in accordance with 10 CFR 50.55a, "Codes and standards," paragraph (z)(1).

The valves covered by this code case are those stem-disk separation non-susceptible valves with remote position indication within the scope of Subsection ISTC, including its mandatory appendices and their verifications methods and frequencies, in accordance with regulatory requirements. Valves with remote position indication within the scope of ASME OM Code, Subsection ISTA, paragraph ISTA-1100, not satisfying the scope and provisions of this code case shall meet the valve position verification requirements in ASME OM Code, Paragraph ISTC-3700, in accordance with regulatory requirements.

To categorize a valve as not susceptible to stem-disk separation, the valve shall have a documented justification that the stem-disk connection is not susceptible to separation based on the internal design, service conditions, applications, and evaluation of the stem-disk connection using plant-specific and industry operating experience, and vendor recommendations.

Valves with remote position indicators that are not susceptible to stem-disk separation shall be verified to accurately represent valve operation as discussed in paragraph 1.4, “Position Verification Testing Requirements for Valves Not Susceptible to Stem-Disk Separation,” of the code case.

Code Case OMN-28 was approved for use by the ASME on March 4, 2021 and is listed on the ASME OM Code Case index as being applicable to the 2012 Edition. As stated in Section 2, the ASME OM Code, 2012 Edition is the applicable edition used at PNPP.

6. <u>Duration of Proposed Alternative</u>

The proposed alternative is requested for use during the PNPP fourth 10-year in-service test interval, scheduled to end May 17, 2029.

7. <u>Precedent</u>

This alternative to implement Code Case OMN-28 to satisfy ISTC-3700 for valves not susceptible to stem-disk separation has been approved as specified in the following:

> Letter from Scott P. Wall, US NRC to David Rhoades, Exelon Generation Company, LLC. Subject: Braidwood Station, Units 1 And 2; Calvert Cliffs Nuclear Power Plant, Units 1 And 2; Clinton Power Station, Unit No. 1; Limerick Generating Station, Units 1 And 2; Nine Mile Point Nuclear Station, Units 1 And 2; Peach Bottom Atomic Power Station, Units 2 And 3; And R. E. Ginna Nuclear Power Plant — Proposed Alternative To Use ASME OM Code Case OMN-28 (EPID L-2021-LLR-0056),(ML21230A206), dated September 3, 2021.