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10 CFR 50.55a

December 17, 2018

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

Limerick Generating Station, Units 1 and 2
Renewed Facility Operating License Nos. NPF-39 and NPF-85
NRC Docket Nos. 50-352 and 50-353

Subject: Relief Requests Associated with Fourth Ten-Year Inservice Testing Interval

In accordance with 10 CFR 50.55a, "Codes and standards," paragraph (z)(1), Exelon Generation Company, LLC (Exelon) requests your review and approval of the attached relief requests associated with the Inservice Testing (IST) Program for the Limerick Generating Station (LGS), Units 1 and 2. Exelon is requesting approval of these relief requests for the fourth ten-year IST interval which is currently scheduled to start on January 8, 2020.

We request approval of these relief requests by December 17, 2019.

There are no regulatory commitments contained within this submittal.

If you have any questions concerning this letter, please contact Mr. David Neff at (267) 533-1132.

Sincerely,

D. P. Helker

David P. Helker
Manager - Licensing & Regulatory Affairs
Exelon Generation Company, LLC

Attachment: Relief Requests Associated with the Fourth Ten-Year Interval for Limerick Generating Station, Units 1 and 2

cc:	USNRC Region I, Regional Administrator	w/attachment
	USNRC Project Manager, LGS	"
	USNRC Senior Resident Inspector, LGS	"
	R. R. Janati, Pennsylvania Bureau of Radiation Protection	"

ATTACHMENT

Relief Requests Associated with the Fourth Ten-Year Interval for Limerick Generating Station, Units 1 and 2

Relief Request No.	Description
GVRR-8, Revision 1	Pressure Isolation Valve Leakage Test Frequency
11-PRR-1, Revision 2	Use of Code Case OMN-16, ESW Pump Test Using Pump Curves
90-PRR-1, Revision 2	Installed Pump Flow Instrumentation Accuracy greater than 2%
47-VRR-2, Revision 1	Control Rod Drive Scram Valves

EXELON GENERATION COMPANY, LLC
IST PROGRAM – RELIEF REQUEST
Limerick Generating Station Units 1 and 2

Proposed Alternative in Accordance with 10 CFR 50.55a(z)(1)
GVRR-8, Revision 1 – Pressure Isolation Valve Leakage Test Frequency

1. ASME OM Code Component(s) Affected

Component	System	Code Class	Category	Type
HV-51-1(2)F041A-D	RHR	1	A/C	SA
HV-51-1(2)F017A-D	RHR	1	A	MO
HV-51-1(2)42A-D	RHR	1	A	AO
HV-51-1(2)F050A/B	RHR	1	A/C	SA
HV-51-1(2)F015A/B	RHR	1	A	MO
HV-51-1(2)51A/B	RHR	1	A	AO
51-1(2)200A/B	RHR	1	A/C	SA
HV-51-1(2)F008	RHR	1	A	MO
HV-51-1(2)F009	RHR	1	A	MO
HV-52-1(2)F005	CS	1	A	MO
HV-52-1(2)F006A/B	CS	1	A/C	SA
HV-52-1(2)F039A/B	CS	1	A	AO
HV-52-1(2)08	CS	1	A/C	SA

2. Applicable Code Edition

American Society of Mechanical Engineers (ASME) Code for Operation and Maintenance of Nuclear Power Plants (OM Code), 2012 Edition with no addenda.

3. Applicable Code Requirement(s)

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."

ISTC-3630(a), *Frequency*, states "Tests shall be conducted at least once every 2 yr."

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-3630(a). The basis of the request is that the proposed alternative would provide an acceptable level of quality and safety.

ISTC-3630 requires that leakage rate testing for Pressure Isolation Valves (PIVs) be performed at least once every two years. PIVs are not specifically included in the scope for performance-based testing as provided for in 10 CFR 50 Appendix J, *Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors*, Option B, *Performance Based Requirements*. These motor-operated valve (MOV), air-operated valve (AOV), and check valve (CV) PIVs are all containment isolation valves (CIVs) but are not all tested per Appendix J based on a justification of the penetration being a single CIV within a closed loop.

Limerick Generating Station (LGS) Technical Specification (TS) Section 6.8.4.g, *Primary Containment Leakage Rate Testing Program*, states, in part:

> "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 Regulatory Guide 1.163, "Performance-Based Containment Leakage Test program," dated September 1995 ..."

NRC Regulatory Guide (RG) 1.163, *Performance-Based Containment Leak-Test Program*, endorses Nuclear Energy Institute (NEI) 94-01, *Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J, Revision 0*, dated July 26, 1995, as an acceptable method for complying with the provisions of Option B to 10 CFR 50, Appendix J, with certain exceptions. Sections 10.1 and 11.3 of NEI 94-01 allow an extension of up to 25 percent of the test interval (not to exceed 15 months).

The concept behind the Option B alternative for CIVs is that licensees should be allowed to adopt cost effective methods for complying with regulatory requirements. Additionally, NEI 94-01 describes the risk-informed basis for the extended test intervals under Option B. That justification shows that for CIVs which have demonstrated good performance by passing their leak rate tests for two consecutive cycles, further failures would be governed by the random failure rate of the component. NEI 94-01 also presents the results of a comprehensive risk analysis, including the conclusion that "the risk impact associated with increasing [leak rate] test intervals is negligible (i.e., less than 0.1 percent of total risk)."

The valves identified in this request are all in water applications. Testing is performed with water pressurized to the functional maximum pressure differential. This request is intended to provide for a performance-based scheduling of PIV tests at LGS. The reason for requesting this relief is dose reduction to conform with NRC and industry As-Low-As-Reasonably Achievable (ALARA) radiation dose principles. The review of historical data identified that PIV testing each refueling outage results in a total personnel dose of approximately 700 milli-roentgen equivalent man (rem). The proposed extended test interval (assuming all PIVs are on extended frequency) would provide for a savings of approximately 1.4 rem over three refuel outages.

NUREG-0933, *Resolution of Generic Safety Issues*, Issue 105, *Interfacing Systems LOCA at LWRs*, discussed the need for PIV leak rate testing based primarily on three pre-1980 historical failures of applicable valves industry-wide. These failures all involved human errors in either operations or maintenance. None of these failures involved in-service equipment degradation. The performance of PIV leak rate testing provides assurance of acceptable seat leakage with the valve in a closed condition.

Typical PIV testing does not identify functional problems which may inhibit the valves' ability to reposition from open to closed. For check valves, functional testing is accomplished in accordance with ASME OM Code sections ISTC-3520, *Exercising Requirements*, and ISTC-3522, *Category C Check Valves.*

For power-operated valves, full stroke testing is performed in accordance with the ASME OM Code Section ISTC-5100, Power Operated Valves (POVs) to ensure their functional capabilities. Performance of the separate two-year PIV leak rate testing does not contribute any additional assurance of functional capability; it only determines the seat tightness of the closed valves.

5. Proposed Alternative and Basis for Use

LGS proposes to perform PIV testing at intervals ranging from every refueling outage to every third refueling outage. The specific interval for each valve would be a function of its performance and would be established in a manner consistent with the CIV process under 10 CFR 50 Appendix J, Option B. For those valves that are also Appendix J leak tested, a conservative control will be established such that if any valve fails either the Appendix J or PIV test, the test interval for both tests will be reduced consistent with Appendix J, Option B requirements, until good performance is reestablished.

The primary basis for this relief request is the historically good performance of the PIVs with the exceptions of the HV-51-2F050A/B check valves. HV-51-2F050A/B are the shutdown cooling injection header check valves. Several modifications have been implemented to improve the leak tightness of these valves. Based on the test data presented in Table 2, certain valves demonstrating unsatisfactory performance will remain on a two-year test frequency until satisfactory performance is achieved.

The functional capability of the active check valves is demonstrated by the opening and closing of the valves each refueling outage. These tests are separate and distinct from the PIV seat leakage testing and are performed in accordance with ASME OM Code, Section ISTC-3522.

Note that NEI 94-01 is not the sole basis for this relief request, given that NEI 94-01 does not address seat leakage testing with water. This document was cited as an approach similar to the requested alternative method.

Tables 1 through 4 below present historical test data that documents PIV performance for the Residual Heat Removal (RHR) and Core Spray (CS) systems.

Table 1: Historical Leak Rate Test Performance for RHR MOV PIVs

Component	Date of Test	Measured Valve (gpm)	Required Action Limit (gpm)	Comments
HV-51-1F008	03/29/2016	0.07	1	
	03/24/2014	0.01	1	
	03/01/2012	0.09	1	
HV-51-1F009	03/29/2016	0.07	1	
	03/24/2014	0.01	1	
	03/01/2012	0.08	1	
HV-51-1F015A	03/28/2018	0.00	1	
	03/25/2016	0.01	1	
	03/27/2014	0.01	1	
HV-51-1F015B	04/03/2016	0.1	1	
	03/18/2014	0.0	1	
	02/23/2012	0.012	1	
HV-51-1F017A	04/02/2018	0.00	1	
	03/22/2016	0.01	1	
	03/28/2014	0.0	1	
HV-51-1F017B	04/04/2016	0.0	1	
	03/19/2014	0.0	1	
	02/22/2012	0.06	1	
HV-51-1F017C	03/30/2018	0.0	1	
	03/25/2016	0.0	1	
	03/29/2014	0.01	1	
HV-51-1F017D	03/31/2016	0.0	1	
	03/18/2014	0.0	1	
	02/22/2012	0.0	1	
HV-51-2F008	04/25/2015	0.0	1	
	03/31/2013	0.0	1	
	04/09/2011	0.04	1	
HV-51-2F009	04/25/2015	0.0	1	
	03/31/2013	0.0	1	
	04/09/2011	0.0	1	

Table 1: Historical Leak Rate Test Performance for RHR MOV PIVs (continued)				
HV-51-2F015A	05/01/2017	0.0	1	
	04/22/2015	0.0	1	
	03/29/2013	0.0	1	
HV-51-2F015B	04/16/2015	0.02	1	
	04/05/2013	0.0	1	
	04/11/2011	0.0	1	
HV-51-2F017A	05/02/2017	0.0	1	
	04/26/2015	0.0	1	
	03/28/2013	0.0	1	
HV-51-2F017B	04/17/2015	0.0	1	
	04/06/2013	0.0	1	
	04/11/2011	0.0	1	
HV-51-2F017C	05/02/2017	0.07	1	
	04/27/2015	0.03	1	
	03/30/2013	0.06	1	
HV-51-2F017D	04/18/2015	0.01	1	
	04/04/2013	0.059	1	
	04/13/2011	0.02	1	

Table 2: Historical Leak Rate Test Performance for RHR CV/AOV PIVs				
Component	Date of Test	Measured Valve (gpm)	Required Action Limit (gpm)	Comments
HV-51-1F050A HV-51-151A	03/28/2018	0.0	2	
	03/25/2016	0.2	2	
	03/27/2014	0.4	2	
HV-51-1F050B HV-51-151B	04/03/2016	0.6	2	
	03/18/2014	0.5	2	
	02/23/2012	0.38	2	
51-1200A	03/28/2018	0.0	1	
	03/25/2016	0.7	1	
	03/27/2014	0.1	1	

Table 2: Historical Leak Rate Test Performance for RHR CV/AOV PIVs (continued)

51-1200B	04/03/2016 03/18/2014 02/23/2012	0.2 0.05 0.32	1 1 1	
HV-51-1F041A HV-51-142A	03/27/2018 03/22/2016 03/28/2014	0.0 UNSAT 0.14	2 2 2	EXCESSIVE LEAKAGE
HV-51-1F041B HV-51-142B	04/07/2018 04/04/2016 03/19/2014	0.0 0.08 0.05	2 2 2	
HV-51-1F041C HV-51-142C	03/30/2018 03/25/2016 03/29/2014	0.0 0.4 0.236	2 2 2	
HV-51-1F041D HV-51-142D	03/31/2016 03/18/2014 02/22/2012	0.2 0.1 0.07	2 2 2	
HV-51-2F050A HV-51-251A	05/04/2017 04/22/2015 03/29/2013	UNSAT UNSAT 0.0	2 2 2	OFF SCALE
HV-51-2F050B HV-51-251B	04/25/2017 04/16/2015 04/05/2013	UNSAT UNSAT UNSAT	2 2 2	OFF SCALE UNABLE TO PRESS
51-2200A	05/01/2017 04/22/2015 03/29/2013	0.0 0.0 0.0	1 1 1	
51-2200B	04/19/2017 04/16/2015 04/05/2013	0.0 0.0 0.0	1 1 1	
HV-51-2F041A HV-51-242A	05/02/2017 04/26/2015 03/28/2013	0.02 0.0 0.0	2 2 2	

Table 2: Historical Leak Rate Test Performance for RHR CV/AOV PIVs (continued)

HV-51-2F041B HV-51-242B	04/17/2015 04/06/2013 04/11/2011	0.0 0.0 0.0	2 2 2	
HV-51-2F041C HV-51-242C	05/02/2017 04/27/2015 03/30/2013	0.07 0.06 0.06	2 2 2	
HV-51-2F041D HV-51-242D	04/25/2017 04/18/2015 04/04/2013	UNSAT UNSAT 0.0	2 2 2	OFF SCALE

Table 3: Historical Leak Rate Test Performance of CS MOV PIVs

Component	Date of Test	Measured Valve (gpm)	Required Action Limit (gpm)	Comments
HV-52-1F005	03/31/2018 03/23/2016 03/30/2014	0.0 0.0 0.0	1 1 1	
HV-52-2F005	05/03/2017 04/24/2015 03/27/2013	0.0 0.0 0.035	1 1 1	

Table 4: Historical Performance of CS CV/AOV PIVs

Component	Date of Test	Measured Valve (gpm)	Required Action Limit (gpm)	Comments
HV-52-1F006A HV-52-1F039A	03/31/2018	0.09	2	
	03/23/2016	0.06	2	
	03/30/2014	0.1	2	
HV-52-1F006B HV-52-1F039B	04/05/2016	0.0	2	
	03/15/2014	0.118	2	
	02/23/2012	0.06	2	
HV-52-108	04/05/2016	0.1	1	
	03/15/2014	0.0	1	
	02/23/2012	0.0	1	
HV-52-2F006A HV-52-2F039A	05/03/2017	0.09	2	
	04/24/2015	0.1	2	
	03/27/2013	0.035	2	
HV-52-2F006B HV-52-2F039B	04/15/2015	0.0	2	
	04/08/2013	0.05	2	
	04/10/2011	0.09	2	
HV-52-208	04/15/2015	0.1	1	
	04/08/2013	0.0	1	
	04/10/2011	0.0	1	

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 (e.g., 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 are determined in accordance with NEI 94-01, Revision 3-A, Section 11.0, *Bases for Performance and Risk-Based Testing Frequencies for Type A, Type B, and Type C Tests.*"

Additional basis for this proposed alternative is provided below:

- Separate functional testing of MOV PIVs, AOV PIVs and CV PIVs per the ASME OM Code.
- Low likelihood of valve mis-positioning during power operations (e.g., procedures, interlocks).
- Relief valves in the low pressure (LP) piping - these relief valves may not provide Inter-System Loss of Coolant Accident (ISLOCA) mitigation for inadvertent PIV mis-positioning but their relief capacity can accommodate conservative PIV seat leakage rates.
- Alarms that identify high pressure (HP) to LP leakage - Operators are highly trained to recognize symptoms of a present ISLOCA and to take appropriate actions.

6. Duration of Proposed Alternative

The proposed alternative, upon approval, shall be utilized for LGS, Units 1 and 2, during the entire 4th 10-year interval which is scheduled to begin on January 8, 2020 and end on January 7, 2030.

7. Precedent

1. Limerick Nuclear Generating Station – Letter from Stephen S. Koenick (NRC) to Bryan C. Hanson (Exelon Nuclear), *Limerick Generating Station, Units 1 and 2 – Proposed Relief Request GVRR-8 Regarding Inservice Testing Program Third 10-Year Interval (CAC Nos. MF8787 and MF8788)* dated February 7, 2017 (ML17004A063).

2. Peach Bottom Atomic Power Station - Letter from D. A. Broaddus (NRC) to B. C. Hanson (Exelon Nuclear), *Peach Bottom Atomic Power Station, Units 2 and 3 Safety Evaluation of Relief Request GVRR-2 Regarding the Fourth 10-Year Interval of the Inservice Testing Program (CAC Nos. MF7630 and MF7631)* dated September 21, 2016 (ML16235A340).

3. Fermi Power Station - Letter from R. J. Pascarelli (NRC) to J. M. Davis (Detroit Edison), *Fermi 2 Evaluation of In-Service Testing Program Relief Requests VRR-011, VRR-012, and VRR-013 (TAC Nos. ME2558, ME2557, and ME2556)*, dated September 28, 2010 (ML102360570).

4. Quad Cities Nuclear Power Station, Units 1 and 2 - Letter from J. Wiebe (NRC) to M.J. Pacilio (EGC), *Quad Cities Nuclear Power Station, Units 1 and 2 - Safety Evaluation in Support of Request for Relief Associated with the Fifth 10 Year Interval Inservice Testing Program (TAC Nos. ME7981, ME7982, ME7983, ME7984, ME7985, ME7986, ME7987, ME7988, ME7989, ME7990, ME7991, ME7992, ME7993, ME7994, ME7995)*, dated February 14, 2013 (ML13042A348).

5. Dresden Nuclear Power Station, Units 2 and 3 - Letter from T. L. Tate (NRC) to B. Hanson (EGC), *Dresden Nuclear Power Station, Units 2 and 3 - Relief Request to Use An Alternative from the American Society of Mechanical Engineers Code Requirements (CAC Nos. MF5089 and MF5090)* dated October 27, 2015 (ML15174A303).

8. References

1. NRC Regulatory Guide (RG) 1.163, *Performance-Based Containment Leak-Test Program,* dated September 1995

2. Limerick Generating Station (LGS) Technical Specification (TS) Section 6.8.4.g, *Primary Containment Leakage Rate Testing Program*

3. NUREG-0933, *Resolution of Generic Safety Issues:* Issue 105: *Interfacing Systems LOCA at LWRs*

4. Nuclear Energy Institute (NEI) 94-01, *Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J, Revision 0*, dated July 26, 1995

5. 10 CFR 50 Appendix J, *Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors*, Option B, *Performance Based Requirements*

EXELON GENERATION COMPANY, LLC
IST PROGRAM – RELIEF REQUEST
Limerick Generating Station Units 1 and 2

Proposed Alternative in Accordance with 10 CFR 50.55a(z)(1)
11-PRR-1, Revision 2 – Use of Code Case OMN-16, ESW Pump Test Using Pump Curves

1. **ASME Code Components Affected:**

Component ID	Description	Code Class	Group
0A-P548	Emergency Service Water Pump A	3	A
0B-P548	Emergency Service Water Pump B	3	A
0C-P548	Emergency Service Water Pump C	3	A
0D-P548	Emergency Service Water Pump D	3	A

The emergency service water (ESW) pumps are motor driven, vertical line shaft pumps.

2. **Applicable Code Edition**

American Society of Mechanical Engineers (ASME) Code for Operation and Maintenance of Nuclear Power Plants (OM Code), 2012 Edition with no addenda.

3. **Applicable Code Requirement(s)**

ISTA-3130, *Application of Code Cases*, subparagraph *(b)* states "Code Cases shall be applicable to the edition and addenda specified in the test plan."

4. **Reason For Request**

Pursuant to 10 CFR 50.55a, *Codes and standards*, paragraph (z)(1), an alternative is proposed to ISTA-3130(b) requirements for implementing Code Case OMN-16, *Use of a Pump Curve for Testing*, Revision 1. The basis of this request is that the proposed alternative would provide an acceptable level of quality and safety.

ISTA-3130(b) states, "Code Cases shall be applicable to the edition and addenda specified in the test plan." ASME has approved Code Case (CC) OMN-16, Revision 1. This CC is conditionally approved for use by the Nuclear Regulatory Commission (NRC) in Table 2 of Regulatory Guide (RG) 1.192, *Operation and Maintenance Code Case Acceptability, ASME OM Code*, Revision 2. However, CC OMN-16, Revision 1, *Applicability*, states that it is applicable to the "1998 Edition and subsequent editions and addenda through the OMa-2011 Addenda." During the 4th ten-year IST interval, LGS, Units 1 and 2, will be implementing the ASME OM Code 2012 Edition and also proposes to implement CC OMN-16, Revision 1 for testing the ESW pumps.

The ESW System includes a large number of variable heat loads. In addition, the temperature of the system is seasonally dependent and can vary significantly. Therefore, it is extremely difficult to vary the resistance of the system to establish flow or differential pressure conditions at any fixed reference point. Operations personnel would need to assume local manual control of automatic room cooler valves and equipment modulating valves. This requires access to Emergency Core Cooling System (ECCS) room coolers and other safety related equipment causing numerous entries into Radiological Controlled Areas (RCAs) to adjust flow to a fixed reference point in order to perform this quarterly test, which would also result in additional dose. Establishment of multiple sets of reference values would not improve the capability to set either variable at a fixed point.

LGS was authorized to use CC OMN-16 during the previous 3rd 10-year IST interval and proposes to continue with the use of CC OMN-16, Revision 1, in the subsequent 4th IST interval, for using pump reference curves during IST of the ESW pumps. Circumstances and basis for previous NRC approval in the safety evaluation for the 3rd 10-year IST interval for use of CC OMN-16 have not changed.

5. Proposed Alternative and Basis for Use

An alternative to ISTA-3130(b) is proposed to implement CC OMN-16, Revision 1, since the CC Applicability statement only covers through the OMa-2011 Addenda, and ISTA-3130(b) requires applicability to the OM 2012 Edition for LGS, Units 1 and 2.

RG 1.192, Revision 2, Table 2, *Conditionally Acceptable OM Code Cases*, approves use of CC OMN-16, Revision 1 with the following condition: "Figure 1 was inadvertently omitted from OMN-16, Revision 1, in the 2012 Edition of the OM Code. The Code Case is approved for use provided it is supplemented with Figure 1 of OMN-16 that is in the 2006 Addendum of the OM Code. (Note: CC OMN-16, 2006 Addenda, was unconditionally approved in Revision 1 of RG 1.192.)"

In order to monitor the ESW pumps for degradation and assure their operational readiness, reference curves as described in Code Case OMN-16, Revision 1, will be used for inservice testing. This revision of the Code Case will be supplemented with Figure 1, *Examples of Graphical Evaluation of Tests Using Reference Curves*, from the version of OMN-16 that is in the OMb-2006 Addenda of the OM Code, which will address the condition stated in Table 2 of RG 1.192, Revision 2.

Pump testing is performed quarterly using these pump curves. Flow, normally in the range of 3000 to 4100 gpm, is measured and total dynamic head is calculated from the pump discharge pressure and the level of the Spray Pond (i.e., suction). The test point is then compared to the pump curve and determined to be within the acceptance range of Table ISTB-5221-1 (0.95 to 1.10 P_r for the Group A test or 0.95 to 1.06 P_r for the Comprehensive Test), which is also plotted on the pump curve. Corrective action, if required, shall meet the requirements of ISTB-6200.

The original pump curves were prepared during flow balancing activities before commercial operation of LGS, Unit 2, and include many empirical data points taken over the entire operating range of the pumps, essentially from shutoff head to approximately 1.5 times the maximum flow required for safe shutdown or accident mitigation. As the pumps have been replaced, new curves have been generated based on the preservice test requirements of ISTB-5210. These curves exceed the requirements of OMN-16 for a minimum of 3 data points and at least one data point for each 20% of the maximum pump curve range for the portion of the maximum pump curve established by the reference curve.

Vibration readings are taken in accordance with ISTB-3540. In addition to the Code-required vibration readings, several additional readings are taken and analyzed in accordance with the LGS Predictive Maintenance Program. Since these pumps show little variation in vibration over their normal operating range, the acceptance criteria for vibration testing complies with the requirements of Table ISTB-5221-1.

Using the provisions of this request as an alternative to the requirements of ISTA-3130(b), will provide adequate detection of observable ESW pump degradation, and along with the pump testing per CC OMN-16, Revision 1, will continue to provide reasonable assurance of the operational readiness of the LGS, Units 1 and 2, ESW pumps. Therefore, the proposed alternative provides an acceptable level of quality and safety pursuant to 10 CFR 50.55a(z)(1).

6. Duration of Proposed Alternative

The proposed alternative, upon approval, shall be utilized for LGS, Units 1 and 2, during the entire 4^{th} ten-year IST interval, which is scheduled to begin on January 8, 2020 and end on January 7, 2030.

7. Precedent

1. Letter from the U.S. NRC (H. K Chernoff) to Exelon Nuclear (C. G. Pardee), *Limerick Generating Station, Units 1 and 2 – Evaluation of Relief Requests [11-PRR-1, Revision 1] Associated with the Third Inservice Testing Interval (TAC Nos. ME0742-ME0751),* dated November 17, 2009 (ML093080382)

8. References

1. ASME OM Code Case OMN-16, *Use of a Pump Curve for Testing*, Revision 1
2. RG 1.192, *Operation and Maintenance Code Case Acceptability, ASME OM Code*, Revision 2, dated March 2017, published January 2018 (ML 16321A337)

EXELON GENERATION COMPANY, LLC
IST PROGRAM - RELIEF REQUEST
Limerick Generating Station Units 1 and 2

Proposed Alternative in Accordance with 10 CFR 50.55a(z)(1),
90-PRR-1, Revision 2 – Installed Pump Flow Instrumentation Accuracy greater than 2%

1. ASME Code Components Affected

Component ID	Description	Code Class	Group
0AP162	Main Control Room Chilled Water Pump A	3	A
0BP162	Main Control Room Chilled Water Pump B	3	A

2. Applicable Code Edition

American Society of Mechanical Engineers (ASME) Code for Operation and Maintenance of Nuclear Power Plants (OM Code), 2012 Edition with no addenda.

3. Applicable Code Requirement(s)

ISTB-3500, *Data Collection*, paragraph ISTB-3510, *General,* subparagraph (a), *Accuracy*, states, in part, "Instrument accuracy shall be within the limits of Table ISTB-3510-1. . . .For individual analog instruments, the required accuracy is percent of full-scale."

Table ISTB-3510-1, *Required Instrument Accuracy*, specifies an accuracy requirement of ±2% for flow rate instruments.

Paragraph ISTB-3510(b), *Range*, subparagraph (1) states, "The full-scale range of each analog instrument shall be not greater than three times the reference value."

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 ISTB-3510(a). The basis of this request is that the proposed alternative would provide an acceptable level of quality and safety.

For instruments to be in compliance with the Code, both requirements stated above must be met, individually, for each instrument. The combination of the two requirements (i.e., accuracy equal to ±2% of full-scale and full scale being up to 3 times the reference value) yields a permissible inaccuracy of ±6% of the reference value.

The permanently installed flow instruments, FI-90-34A and FI-90-34B, as shown in Table 1 below are calibrated to an accuracy that does not meet the ±2% of full-scale requirement.

5. Proposed Alternative and Basis for Use

As a proposed alternative, LGS proposes to use the currently installed analog instruments, FI-90-034A and FI-90-034B, for measurement of discharge flow for the Main Control Room Chilled Water pumps. Although these instruments do not meet Code requirements, they provide better indication accuracy at the reference value than that which is permitted by the Code.

NUREG 1482, Revision 2, Section 5.5.1, *Range and Accuracy of Analog Instruments*, states, in part, "When the range of a permanently installed analog instrument is greater than three times the reference value, but the accuracy of the instrument is more conservative than that required by the Code, the staff may grant relief when the combination of the range and accuracy yields a reading that is at least equivalent to that achieved using instruments that meet the Code requirements (i.e., up to ±6 percent for Group A and B tests…)."

Table 1 shows the instrument accuracy and full-scale range of the flow instruments used to conduct inservice testing of the Main Control Room Chilled Water pumps listed above. The resulting instrument tolerance and indicated accuracy are calculated and also listed in Table 1. The full-scale range of the installed flow measuring instruments is within the required three times the reference value and meets the OM Code requirement specified in ISTB-3510(b)(1). However, the instrument accuracy is greater than the required ±2 percent of full scale. The indicated accuracy at the reference value is shown to be within the required ±6 percent.

Table 1 – Main Control Room Chilled Water Pumps' Discharge Flow Measuring Instrument Accuracies

Instrument Number	Reference Value (gpm)	Instrument Range (Full-Scale)	Instrument Accuracy	Instrument Tolerance	Indicated Accuracy
FI-90-034A	600	0 - 800	3.08%	24.64	4.11%
FI-90-034B	600	0 - 800	3.04%	24.32	4.05%

Based on Section 5.5.1 of NUREG 1482, Revision 2, and the information provided herein, the existing permanently installed pump instrumentation is considered acceptable in meeting the intent of the ASME OM Code-2012, paragraphs ISTB 3510(a) and 3510(b)(1). Thus, utilizing the permanently installed instrumentation for measuring the Main Control Room Chilled Water pumps' discharge flow provides an acceptable level of quality and safety; therefore, this alternative is proposed in accordance with 10 CFR 50.55a(z)(1).

Circumstances and basis for previous NRC approval of Relief Request 90-PRR-1, Revision 1 for use during the 3rd IST interval have not changed. This request updates the Code reference for use during the 4th IST interval.

6. **Duration of Proposed Alternative**

The proposed alternative, upon approval, shall be utilized for LGS, Units 1 and 2, during the entire 4th 10-year IST interval, which is scheduled to begin on January 8, 2020 and end on January 7, 2030.

7. Precedent

1. Letter from the U.S. NRC (H. K Chernoff) to Exelon Nuclear (C. G. Pardee), *Limerick Generating Station, Units 1 and 2 – Evaluation of Relief Requests [90-PRR-1, Revision 1] Associated with the Third Inservice Testing Interval (TAC Nos. ME0742-ME0751)*, dated November 17, 2009 (ML093080382)

2. Letter from T. Boyce (U. S. Nuclear Regulatory Commission) to R. Duncan II (Carolina Power & Light Company), *Shearon Harris Nuclear Plant, Unit 1 – Relief Request AF-PR-1 for the Third 10-Year Inservice Inspection Interval (TAC No. MD3894)*, dated July 16, 2007 (Relief Request AF-PR-1)

3. Letter from H. Chernoff (U. S. Nuclear Regulatory Commission) to W. Levis (PSEG Nuclear LLC), *Safety Evaluation of Relief Requests for the Third 10-Year Interval of the Inservice Testing Program for Hope Creek Generating Station (TAC Nos. MD3300, MD3301, MD3337, MD3338, MD3353, and MD3354)*, dated April 5, 2007 (P-01)

8. Reference

1. NUREG 1482, *Guidelines for Inservice Testing at Nuclear Power Plants, Inservice Testing of Pumps and Valves and Inservice Examination and Testing of Dynamic Restraints (Snubbers) at Nuclear Power Plants*, Revision 2, dated September 2013 (published October 2013).

EXELON GENERATION COMPANY, LLC
IST PROGRAM - RELIEF REQUEST
Limerick Generating Station Units 1 and 2

Proposed Alternative in Accordance with 10 CFR 50.55a(z)(1),
47-VRR-2, Revision 1 - Control Rod Drive Scram Valves

1. **ASME Code Components Affected**

Component ID	Description	Code Class	Category
XV-47-1-26 (all 185 HCUs)	U1 – Inlet Scram Valves (AOVs)	2	B
XV-47-1-27 (all 185 HCUs)	U1 – Outlet Scram Valves (AOVs)	2	B
XV-47-2-26 (all 185 HCUs)	U2 – Inlet Scram Valves (AOVs)	2	B
XV-47-2-27 (all 185 HCUs)	U2 – Outlet Scram Valves (AOVs)	2	B
47-1-14 (all 185 HCUs)	U1 – Scram Discharge Riser Check Valve	2	C
47-2-14 (all 185 HCUs)	U2 – Scram Discharge Riser Check Valve	2	C

Each of the valves listed above represents 1 of 185 Control Rod Drive (CRD) Hydraulic Control Units (HCU)

2. **Applicable Code Edition**
American Society of Mechanical Engineers (ASME) Code for Operation and Maintenance of Nuclear Power Plants (OM Code), 2012 Edition with no addenda.

3. **Applicable Code Requirement(s)**

ISTC-3510, *Exercising Test Frequency*, states, in part that active "...Category B and Category C check valves shall be exercised nominally every 3 mo, except as provided by paras. ISTC-3520, ISTC- 3540, ISTC-3550, ISTC-3570, ISTC-5221, and ISTC-5222."

ISTC-3560, *Fail-Safe Valves*, states, "Valves with fail safe actuators shall be tested by observing the operation of the actuator upon loss of valve actuating power in accordance with the exercising frequency of para. ISTC-3510."

ISTC-5131, *Valve Stroke Testing*, subparagraph (a), states "Active valves shall have their stroke time measured when exercised in accordance with para. ISTC-3500."

ISTC-5221, *Valve Obturator Movement*, subparagraph (a)(2), states, in part, that "Check valves that have a safety function in only the open direction shall be exercised by initiating flow and observing that the obturator has traveled [to] either the full open position or to the position required to perform its intended function(s)..., and verify closure."

4. **Reason For Request**

Pursuant to 10 CFR 50.55a, *Codes and standards*, paragraph (z)(1), an alternative is proposed to use the guidance provided in Generic Letter (GL) 89-04, Position 7, in lieu of the Code-required exercise frequency, actuator fail-safe testing, and stroke time testing for the inlet and outlet scram valves and the scram discharge riser check valves located on each CRD HCU. The basis of the request is that the proposed alternative to the specified testing requirements would provide an acceptable level of quality and safety.

GL 89-04, Position 7 discusses alternative testing of individual scram valves for control rods in Boiling Water Reactors (BWRs). Position 7 states, in part, that "for those control rod drive system valves where testing could result in the rapid insertion of one or more control rods, the rod scram test frequency identified in the facility TS [Technical Specifications] may be used as the valve testing frequency to minimize rapid reactivity transients and wear of the control rod drive mechanisms. This alternative test frequency should be clearly stated and documented in the IST [Inservice Testing] program."

NUREG-1482, Revision 2, Section 1.3, states, "The recommendations herein replace the guidance and technical positions in GL 89-04. Note that specific relief is required to implement the guidance derived from GL 89-04. However, relief justification may refer to the positions in the GL with clarifying information to clearly show how it would apply to a licensee's situation."

5. Proposed Alternative and Basis for Use

In order to exercise the Category B valves in accordance with ISTC-3510, and test the fail-safe actuators as required by ISTC-3560, the air operated inlet and outlet scram valves would need to be exercise tested at power nominally every three (3) months. The air operated inlet and outlet scram valves, XV-47-1(2)-26 and XV-47-1(2)-27, open on a signal from the Reactor Protection System (RPS) to permit rapid insertion of the control rods (scram). These valves can only be tested by scramming the CRD.

ISTC-5131(a) applies to the Category B air operated inlet and outlet scram valves (XV-47-1(2)-26 and XV-47-1(2)-27). Stroke timing of the air-operated valves is impractical since they are not equipped with indication of the open and closed positions. Control room panel lights verify insertion of the control rod, not valve position. Accordingly, Code-compliant stroke time testing cannot be performed to meet the ISTC-5131(a) requirement. As a proposed alternative, scram time testing as described in GL 89-04, Position 7 will be performed in accordance with LGS, Units 1 and 2, TS SR 4.1.3.2.

ISTC-3510 and ISTC-5221(a)(2) apply to the scram discharge riser check valves (47-1(2)-14). The scram discharge riser check valve 47-1(2)-14 is flow actuated as a result of the outlet scram valves XV-47-1(2)-27 opening. In order to demonstrate that the safety function is exercised, these valves can only be tested by scramming the CRD.

For all listed components, exercise testing at power will result in rapid insertion of control rods causing potential reactivity transients and wear of the CRD mechanisms.

Accordingly, an alternative is proposed to test the valves in accordance with LGS, Units 1 and 2 Technical Specification (TS) Surveillance Requirement (SR) 4.1.3.2, *Control Rod Maximum Scram Insertion Times*, and in conformance with GL 89-04 Position 7.

As a proposed alternative, the valve and scram time testing would be performed in accordance with the LGS, Units 1 and 2 TS SR 4.1.3.2, which states:

> "The maximum scram insertion time of the control rods shall be demonstrated through measurement and, during single control rod scram time tests, the control rod drive pumps shall be isolated from the accumulators:
>
> a. For all control rods prior to THERMAL POWER exceeding 40% of RATED THERMAL POWER with reactor coolant pressure greater than or equal to 950 psig, following CORE ALTERATIONS or after reactor shutdown that is greater than 120 days.
>
> b. For specifically affected individual control rods following maintenance on or modification to the control rod or control rod drive system which could affect the scram insertion time of those specific control rods in accordance with either "1" or "2" as follows:
>
> 1.a Specifically affected individual control rods shall be scram time tested at zero reactor coolant pressure and the scram insertion time from the fully withdrawn position to notch position 05 shall not exceed 2.0 seconds, and
>
> 1.b Specifically affected individual control rods shall be scram time tested at greater than or equal to 950 psig reactor coolant pressure prior to exceeding 40% of rated thermal power.
>
> 2. Specifically affected individual control rods shall be scram time tested at greater than or equal to 950 psig reactor coolant pressure.
>
> c. For at least 10% of the control rods, with reactor coolant pressure greater than or equal to 950 psig, on a rotating basis, and in accordance with the Surveillance Frequency Control Program."

Scram time testing of the control rods demonstrates that the above listed valves will perform their safety function. These valves are required to open to provide drive water and create an exhaust path for insertion of the control rods. Failure of a valve to open would result in the control rod not scramming in accordance with the TS requirements. It is noted that TS SR 4.1.3.2.a is performed at least once per refueling cycle; TS SR 4.1.3.2.b is performed following maintenance or modification to the control rod or control rod drive system; and, TS SR 4.1.3.2.c is performed at a frequency controlled by the Surveillance Frequency Control Program (SFCP). The LGS SFCP and any changes to the surveillance frequencies in the SFCP are implemented in accordance with LGS TS Section 6.8.4.j. GL 89-04, Position 7 states that the rod scram test frequency identified in the TS may be used as the valve testing frequency to minimize rapid reactivity transients and wear of the control rod drive mechanisms. Therefore, testing of the inlet and outlet scram valves and the scram discharge riser check valves would be performed in accordance with LGS, Units 1 and 2 TS SR 4.1.3.2 in conformance with the Surveillance Frequency Control Program.

Monitoring and trending the stroke times of the inlet and outlet scram valves is not necessary because they are indirectly stroke timed and no meaningful correlation between the scram time and valve stroke time can be obtained. The proposed alternative of verifying that the associated control rod meets the scram insertion time limits defined in the LGS TS allows for detecting degradation of these valves, thus ensuring valve operational readiness. Therefore, this alternative to the Code-required exercise frequency, actuator fail-safe testing, and stroke time testing for the inlet and outlet scram valves and the scram discharge riser check valves located on each CRD HCU provides an acceptable level of quality and safety, and thus, is proposed in accordance with 10 CFR 50.55a(z)(1).

6. Duration of Proposed Alternative

The proposed alternative, upon approval, shall be utilized during the entire 4th 10-year IST interval for LGS, Units 1 and 2, which is scheduled to begin on January 8, 2020 and conclude on January 7, 2030.

7. Precedent

1. Letter from the U.S. NRC (H. K Chernoff) to Exelon Nuclear (C. G. Pardee), *Limerick Generating Station, Units 1 and 2 – Evaluation of Relief Requests [47-VRR-1, Revision 0] Associated with the Third Inservice Testing Interval (TAC Nos. ME0742-ME0751),* dated November 17, 2009 (ML093080382)

8. References

1. LGS, Units 1 and 2 Technical Specifications, Surveillance Requirement 4.1.3.2, *Control Rod Maximum Scram Insertion Times*

2. LGS Units 1 and 2 TS Surveillance Frequency Control Program

3. GL 89-04, *Guidance on Developing Acceptable Inservice Testing Programs, Position 7, Testing Individual Control Rod Scram Valves in Boiling Water Reactors (BWRs)*, dated April 3, 1989.

4. NUREG 1482, *Guidelines for Inservice Testing at Nuclear Power Plants, Inservice Testing of Pumps and Valves and Inservice Examination and Testing of Dynamic Restraints (Snubbers) at Nuclear Power Plants*, Revision 2, dated September 2013 (published October 2013).