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10 CFR 50.55a

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

Nine Mile Point Nuclear Station, Units 1 and 2
Renewed Facility Operating License Nos. DPR-63 and NPF-69
NRC Docket Nos. 50-220 and 50-410

Subject: Relief Requests Associated with Fifth and Fourth Ten-Year Inservice Testing Intervals

Attached for your review are relief requests for Nine Mile Point Nuclear Station (NMPNS), Units 1 and 2, associated with the fifth and fourth ten-year Inservice Testing (IST) intervals, respectively. The fifth and fourth intervals of the NMPNS, Units 1 and 2 IST program comply with the ASME OM Code, 2012 Edition. The fifth and fourth ten-year intervals for NMPNS, Units 1 and 2, both begin on January 1, 2019, and will conclude on December 31, 2028. We request your approval by December 20, 2018.

There are no regulatory commitments contained within this letter.

If you have any questions concerning this letter, please contact Mr. David Neff at (610) 765-5631.

Sincerely,

D. P. Helker

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

Attachment: Relief Requests Associated with the Fifth Ten-Year Interval for Nine Mile Point Nuclear Station, Unit 1 and the Fourth Ten-Year Interval for Nine Mile Point Nuclear Station, Unit 2

cc: USNRC Region I, Regional Administrator
NRC Senior Resident Inspector - NMP
NRC Project Manager, NRR - NMP
A. L. Peterson, NYSERDA

ATTACHMENT

Relief Requests Associated with the
Fifth Ten-Year Interval for Nine Mile Point Nuclear Station, Unit 1 and the
Fourth Ten-Year Interval for Nine Mile Point Nuclear Station, Unit 2

Relief Request No.	Description
Unit 1 and Unit 2	
GVRR-3	Elapsed Time Between Successive Openings of PIVs
Unit 1	
ADS-VR-01	ERV Power Operated Valves Tested Each Cycle
CRD-VR-01	Stroke Time Testing of SCRAM Discharge Volume Valves
CTNH202-VR-01	H2/O2 Sample and Return Valve Stroke Time Testing Group 1
CTNH202-VR-02	H2/O2 Sample and Return Valve Stroke Time Testing Group 2
MS-VR-01	Reactor Pressure Vessel Safety Valve Testing
RBCLC-PR-01	Alternate Frequency for Pump Testing
Unit 2	
MSS-VR-01	Reactor Pressure Vessel Safety Relief Valve Testing

1. ASME Code Component(s) Affected

Refer to GVRR-3 Tables 1 and 2, Affected Components

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 with no Addenda.

3. Applicable Code Requirement

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 to the requirement of ASME OM Code ISTC-3630(a) is requested. 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 and check valve PIVs are, in some cases, containment isolation valves (CIVs), but are not within the Appendix J scope since the Reactor Shutdown Cooling System valves are considered water-sealed.

The Nine Mile Point, Unit 1 (NMP1) leakage rate testing program is in accordance with the Nuclear Energy Institute (NEI) 94-01, *Industry Guideline for Implementing Performance-Based Option of 10 CFR 50, Appendix J*, Revision 0, dated July 21, 1995.

The Nine Mile Point, Unit 2 (NMP2) Technical Specifications (TS) contain a requirement to establish the leakage rate testing program in accordance with the guidelines contained in NEI 94-01, Revision 2-A, *Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J,* dated October 2008.

NEI 94-01, Paragraph 10.2.3.2, "Extended Test Interval," [as approved in the final safety evaluation for NEI 94-01, Revision 3, via letter dated June 8, 2012 (ADAMS Accession No. ML121030286)], 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 should be determined by a licensee in accordance with Section 11.0."

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 the successful completion of two consecutive leakage rate tests over two consecutive cycles, may increase their test frequencies. Further, it 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 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 pressures lower than function maximum pressure differential; however, the observed leakage is adjusted to the function maximum pressure differential value in accordance with ISTC-3630(b)(4). This proposed alternative is intended to provide for a performance-based scheduling of PIV tests at NMP1 and NMP2. The reason for requesting this alternative is dose reduction to conform with NRC and industry As Low As Reasonably Achievable (ALARA) radiation dose principles. The nominal fuel cycle lengths at NMP1 and NMP2 are 24 months. However, since refueling outages may be scheduled slightly beyond 24 months, a 4-1/2 year period is used to provide a bounding timeframe to encompass two refueling outages. The review of recent historical data identified that PIV testing each refueling outage results in a total personnel dose of approximately 1 Rem, assuming all of the PIVs remain classified as good performers. The proposed extended test intervals would provide for a savings of approximately 1 Rem over an approximate 4-year period (two 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-1985 historical failures of applicable valves industry-wide. These failures all involved human errors in either operations or maintenance. None of these failures involved inservice 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 valve's ability to reposition from open to closed.

For check valves, functional testing is accomplished in accordance with ASME OM Code paragraph ISTC-3522, "Category C Check Valves." For power-operated valves, full stroke functional testing is accomplished in accordance with the ASME OM Code paragraph ISTC-3521 "Category A and Category B Valves." 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

NMP1 and NMP2 propose 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. A conservative control will be established such that if any valve fails either PIV test, the test interval for both tests will be reduced consistent with Appendix J, Option B requirements until good performance is reestablished.

The functional capability of the check valves is demonstrated by the open and close exercising. This testing is separate and distinct from PIV testing and is performed at a refuel outage frequency in accordance with ASME OM Code, paragraph ISTC-3522.

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

If this proposed alternative is authorized and the PIVs continue to exhibit good performance, the PIV test frequency could be extended such that the leak test would not be required each 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."

Additional basis for this relief request is provided below:

- Separate functional testing of motor-operated valve (MOV) PIVs and Check Valve PIVs per the ASME OM Code will continue.
- The 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.

The primary basis for this alternative request is the historically good performance of the PIVs. Since approval of the previous request, (as described in precedent 1 in Section 7), the leakage test intervals are based on performance; therefore, additional leakage test data is not included in this interval alternative request.

Based on valve performance history, there is continued assurance of valve operational readiness, as required by ASME OM-2012 Code, paragraph ISTC-3630. Therefore, this proposed alternative to extend the testing frequency consistent with the testing frequencies discussed within NEI 94-01, Revision 3-A, Paragraph 10.2.3.2, "Extended Test Interval," will continue to provide assurance of the valves' operational readiness and provides an acceptable level of quality and safety pursuant to 10 CFR 50.55a(z)(1).

6. Duration of Proposed Alternative

This request, upon approval, will be applied to the NMP1 Fifth 10-Year and NMP2 Fourth10-year intervals, which are scheduled to begin January 1, 2019, and conclude on December 31, 2028.

7. Precedence

1. This relief request was previously approved for the fourth and third 10-year intervals for NMP1 and NMP2, respectively, in letter from NRC (J. G. Danna) to Exelon Generation Company, LLC (B. C. Hanson), “Nine Mile Point Nuclear Station, Units 1 and 2 – Re: Alternative [GVRR-3] to the Requirements of the American Society of Mechanical Engineers Code for Operation and Maintenance of Nuclear Power Plants (CAC Nos. MF9073 and MF9074),” dated May 30, 2017 (ML17136A112).

2. A similar relief request was approved for Fermi Power Station for the third IST Interval in a letter from NRC (R. J. Pascarelli) to Detroit Edison (J. M. Davis), "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).

3. A similar relief request was approved for Quad Cities Nuclear Power Station, Units 1 and 2 for the fifth IST interval in a letter from NRC (J. Wiebe) to Exelon (M. J. Pacilio), "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, ME7990, ME7991, ME7992, ME7993, ME7994, and ME7995),” dated February 14, 2013 (ML13042A348).

4. A similar relief request was approved for Dresden Nuclear Power Station, Units 2 and 3 for the fifth IST interval in a letter from NRC (T. L. Tate) to Exelon (B. C. Hanson), "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 andMF5090),” dated October 27, 2015 (ML15174A303).

5. A similar relief request was approved for Peach Bottom Atomic Power Station, Units 2 and 3 for the fourth interval in a letter from NRC (D. A. Broaddus) to Exelon (B. C. Hanson), "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).

8. References

1. NEI 94-01,“*Industry Guideline for Implementing Performance-Based Option of 10 CFR 50, Appendix J*,” Revision 3-A

2. NUREG 0933, “Resolution of Generic Safety Issues,” Section 3, Issue 105: "Interfacing Systems LOCA at LWRs (Rev. 4)”

3. Letter from NRC (S. Bahadur) to NEI (B. Bradley), 'Final Safety Evaluation of Nuclear Energy Institute (NEI) Report, 94-01, Revision 3, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J" (TAC No. ME2164),' dated June 8, 2012 (ADAMS Accession No. ML121030286)

GVRR-3 Table 1 – Affected Components- Unit 1

Unit 1 Components	**System**	**Code Class**	**Category**
CKV-40-03	CS	1	A/C
CKV-40-13	CS	1	A/C
CKV-40-20	CS	2	A/C
CKV-40-21	CS	1	A/C
CKV-40-22	CS	1	A/C
CKV-40-23	CS	2	A/C
CKV-38-165	SDC	2	A/C
CKV-38-166	SDC	2	A/C
CKV-38-167	SDC	2	A/C
CKV-38-168	SDC	2	A/C
CKV-38-169	SDC	1	A/C
CKV-38-170	SDC	1	A/C
CKV-38-171	SDC	1	A/C
CKV-38-172	SDC	1	A/C

GVRR-3 Table 2 – Affected Components - Unit 2

Components	System	Code Class	Category
2CSH*V108	CSH	1	A/C
2CSH*MOV107	CSH	1	A
2CSL*V101	CSL	1	A/C
2CSL*MOV104	CSL	1	A
2ICS*V156	ICS	1	A/C
2ICS*V157	ICS	1	A/C
2RHS*V16A	RHS	1	A/C
2RHS*V16B	RHS	1	A/C
2RHS*V16C	RHS	1	A/C
2RHS*V39A	RHS	1	A/C
2RHS*V39B	RHS	1	A/C
2RHS*MOV104	RHS	1	A
2RHS*MOV112	RHS	1	A
2RHS*MOV113	RHS	1	A
2RHS*MOV24A	RHS	1	A
2RHS*MOV24B	RHS	1	A
2RHS*MOV24C	RHS	1	A
2RHS*MOV40A	RHS	1	A
2RHS*MOV40B	RHS	1	A
2RHS*MOV67A	RHS	1	A
2RHS*MOV67B	RHS	1	A

1. ASME Code Component(s) Affected

The following Main Steam Electromatic Relief Valves (ERVs) are affected:

Component	Description	Class	Category
PSV-01-102A	Main Steam ERV	1	B
PSV-01-102B	Main Steam ERV	1	B
PSV-01-102C	Main Steam ERV	1	B
PSV-01-102D	Main Steam ERV	1	B
PSV-01-102E	Main Steam ERV	1	B
PSV-01-102F	Main Steam ERV	1	B

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 with no Addenda.

3. Applicable Code Requirement

ISTC-3510, Exercising Test Frequency, states, in part: "Power-operated relief valves shall be exercise tested once per fuel cycle."

ISTC-3700, Position Verification Testing, states, in part: "Valves with remote position indicators shall be observed locally at least once every 2 yr to verify that valve operation is accurately indicated."

ISTC-5111, Valve Testing Requirements, states, in part: "*(a)* Testing shall be performed in the following sequence or concurrently. If testing in the following sequence is impractical, it may be performed out of sequence, and a justification shall be documented in the record of tests for each test or in the test plan:
(1) leakage testing
(2) stroke testing
(3) position indication testing"

ISTC-5113, Valve Stroke Testing, states, in part: "*(a)* Active valves shall have their stroke times measured when exercised in accordance with para. ISTC-3500."

ISTC-5114, Stroke Test Acceptance Criteria, states, in part: "Test results shall be compared to the reference values established in accordance with para. ISTC-3300, ISTC-3310, or ISTC-3320."

4. Reason for Request

Pursuant to 10 CFR 50.55a, *Codes and standards*, paragraph (z)(1), an alternative is proposed to the power-operated relief valve test requirements of the ASME OM-2012 Code. The basis of the request is that the proposed alternative would provide an acceptable level of quality and safety.

There are six ERVs installed on the main steam (MS) lines inside the drywell. Each ERV consists of a main valve, a pilot valve assembly, and a solenoid actuator (see Figure 1). The ERVs are opened by either signals from automatic actuation instrumentation or manually and, thus, do not rely on spring setpoints for valve actuation.

The ASME OM-2012 Code-required testing for the six ERVs would be satisfied by manually stroking open each ERV with the reactor at pressure once every operating cycle. It would be performed during plant startup following a refueling outage (RFO). Experience in the industry and at NMPNS, Unit No. 1 (NMP1), indicates that manually opening the ERVs during plant operation can increase the potential for main disc seat leakage and pilot valve seat leakage. NMP1 experienced main disc seat leakage in March 2001 and pilot valve seat leakage in December 2002, both of which were attributed to debris on the seats caused by testing the valves using steam. Leakage from the main valve disc can cause increases in suppression pool (torus) temperature and level, requiring more frequent suppression pool cooling and pump-down operations, and diverts steam from the power generation steam cycle. Excessive leakage from the pilot valve can cause inadvertent opening of the main valve and impair its ability to re-close.

The proposed alternative will allow testing of the ERVs that is appropriate to demonstrate functionality without cycling the valves in place using reactor steam pressure. This is consistent with NUREG-0737, "Clarification of TMI Action Item Requirements," Item II.K.3.16, "Reduction of Challenges and Failures of Relief Valves," which recommended that the number of relief valve openings be reduced as much as possible and that unnecessary challenges should be avoided.

5. Proposed Alternative and Basis for Use

This relief request proposes an alternative to performing in-situ ERV steam pressure testing every RFO. The proposed alternative consists of a combination of offsite steam testing of the main valves, actuator cycling, and other inspections and maintenance activities. The proposed alternative would provide an acceptable level of quality and safety, as further discussed below.

System Description

There are six Dresser model 1525VX solenoid-actuated, pilot-operated ERVs installed at NMP1. The ERVs are connected to the MS lines between the MS line flow restrictor and the inboard MS isolation valve. Each ERV has its own discharge pipe that is equipped with an acoustic monitor to detect flow noise and a thermocouple to sense discharge fluid temperature to monitor for valve actuation and/or leakage.

The ERVs have two functional modes of operation: the automatic depressurization system (ADS) mode and the overpressurization relief mode. In the ADS mode, the ERVs depressurize the reactor vessel in the event of a small break loss of coolant accident (SBLOCA) by relieving steam to the torus, allowing the core spray system to inject (spray) cooling water into the reactor vessel. The ADS mode actuates on concurrent “lo-lo-lo” reactor water level and high drywell pressure signals. The six ERVs, three primary valves, and three backup valves are required to be operable for the ADS mode. Operation of three ERVs is sufficient to depressurize the reactor coolant system (RCS) and permit full core spray system flow.

The ERVs also provide overpressure protection (relief mode) for the reactor and MS piping by limiting reactor pressure during transients that result in a pressure increase. In the overpressurization relief mode, pressure switches that monitor reactor vessel pressure actuate six ERVs at staggered setpoints to ensure sufficient margin between the analyzed peak transient pressure and the lowest setpoint for the reactor head safety valves to prevent safety valve actuation during anticipated transients.

Valve Operation

Steam under pressure from the reactor enters the main valve and passes upward around the disc guide. Steam enters the chamber below the main disc through a small orifice located in the disc retainer plate. Inlet steam pressure holds the main valve disc closed. A main disc spring is provided to keep the main valve disc in the closed position at low pressures or while depressurized. The pilot valve disc is held in the closed position by a pilot valve spring and steam pressure in the chamber below the pilot disc. When the solenoid actuator is energized, the actuator plunger depresses the pilot valve operating lever, thereby opening the pilot valve. When the pilot valve is opened, steam is released through the outlet port at a faster rate than supplied through the inlet orifice. This causes the chamber below the main disc to depressurize, causing the valve to open. To close the valve, the solenoid actuator is de-energized, thereby closing the pilot valve and allowing steam pressure to reseat the main valve.

1. Exercise Test Frequency Alternative to ISTC-3510

The six ERVs are currently tested in accordance with approved relief request ADS-VR-01. This includes the exercising of all six ERV solenoid actuators and the replacement of three (3) of the six (6) main valves each RFO. Inspections and precision preventative maintenance are performed each RFO for all six of the solenoid actuators and pilot valve assemblies.

For the proposed alternative, all six of the ERV solenoid actuators will be exercised each RFO, and two (2) of the six (6) main valves will be replaced with pretested spare valves each RFO such that all six valves will be replaced with pretested spare valves over a 6-year period with a six-month grace period. Inspections and precision preventive maintenance (described below) will be performed each RFO for all six of the solenoid actuators and pilot valve assemblies, with the IST requirements incorporated as part of the preventive maintenance activities. This combination of testing, inspections, and maintenance activities provides an acceptable level of quality and safety without requiring the six ERVs to be stroked with reactor steam during plant startup.

Solenoid Actuator

Each ERV solenoid actuator will be exercised each RFO. The closing stroke de-energizes the solenoid and allows the actuator to return to its fail-safe (closed) position. This test will be performed with the pilot valve and solenoid actuator mounted in their normal installed positions inside the drywell, which allows the solenoid actuator to be actuated electrically from the control room by placing the control switch in the Open position. The pilot valve operating lever and pilot valve stem will be secured in the Open position during this test to prevent damage to the pilot valve assembly, which could result from dry-stroking with no backpressure. The maintenance activities include detailed inspections of the electrical and mechanical components of the solenoid actuator.

NMP1 licensee event report (LER) 03-001, “Technical Specification Cooldown Rate Exceeded During Required Cooldown for a Failed Solenoid Actuated Pressure Relief Valve,” reported an event involving an ERV that failed to open due to high resistance in the solenoid actuator cutout switch contacts. The high resistance contacts limited the current through the solenoid operating coil, which reduced the force that the plunger applied to the pilot valve operating lever. Further investigation and examination showed that the high contact resistance was due to the tin coating having been worn off the cutout switch contacts, allowing excessive contact oxidation to occur. Preventive maintenance activities now include inspection and cleaning of the cutout switch contacts, as necessary, to assure that the contact surfaces are clean and free of oxidation, corrosion, and discoloration. The contact tin plating is verified to be intact and not worn off exposing the copper base material. Associated springs and mechanisms are inspected, and the as-left contact resistances are verified. Resistance checks are performed on both actuator coils, and actuator operating currents during electrical actuation are verified to be within acceptance limits. These steps provide substantial indication that the solenoid actuator is capable of functioning as designed and producing its full output force.

Stroke timing of the solenoid actuator is not performed since the actuator is a sub-component of the total ERV. Degradation is monitored through the preventive maintenance inspections in lieu of trending millisecond stroke time variations.

Pilot Valve

Each ERV pilot valve will be exercised each RFO when the new/refurbished pilot valve assembly is installed in the pilot housing. Note that the pilot valve housing is permanently welded to the outside of the ERV enclosure located in the drywell (see Figure 1). Removal and reinstallation of the pilot valve assembly does not affect the ERV main valve. The maintenance activities include inspections of the pilot valve assembly parts and the pilot valve housing interior to identify any damage or wear that could impair free movement of the stem or proper valve seating. Parts are refurbished or replaced as necessary. Cleanliness of parts and components and absence of foreign material are verified prior to reassembly.

NMP1 has experienced a stuck-open ERV event caused by improper maintenance. NMP1 LER 04-001, "Manual Reactor Scram and Cooldown Rate Exceeding Technical Specification Limits Due to Electromatic Relief Valve Failure to Close," reported an event involving an ERV that stuck open due to a maintenance error in which an extraneous gasket was installed in the pilot valve housing. This condition allowed steam to bypass the pilot valve seat, thereby preventing steam pressure from building up under the main valve disc to close the valve when

given the closure signal. Appropriate precautions and instructions have been incorporated into the ERV maintenance procedure to ensure that the correct gasket is used and sufficient torque is applied to prevent steam from bypassing the pilot valve seat.

Prior to re-installing the pilot valve assembly inside the pilot housing, pilot stem/disc leak testing and freedom of movement and reseat functionality are demonstrated. A complete cleanliness inspection must be performed prior to installing the pilot valve assembly back into the housing. The housing is thoroughly cleaned and vacuumed to remove moisture and debris to minimize the potential for debris blocking or hindering pilot valve performance. Following installation of the pilot valve assembly inside the housing, the pilot valve operating lever and pilot valve assembly freedom of movement and clearance adjustments are confirmed, followed by stroking the solenoid actuator plunger by hand to the full extent of travel. This ensures that the solenoid actuator plunger, pilot valve operating lever, and pilot valve assembly function as a unit, while eliminating the risk of damage resulting from electrically stroking the pilot valve in the absence of steam pressure (referred to as dry-stroking). The pilot valve freedom of movement check allows the pilot valve disc to return to its fail-safe (closed) position. NMP1 LER 00-04-01, "Manual Reactor Scram and Unusual Event Declaration Due to Stuck Open Electromatic Relief Valve and Failed Vacuum Breaker on Electromatic Relief Valve Discharge Line," reported an event involving an ERV that unexpectedly opened and would not reclose. The cause was attributed to a bent stem in the pilot valve assembly and partial disengagement of the pilot valve disc from the stem. It was determined that the pilot valve stem-disc separation had occurred as a result of dry-stroking the ERV pilot valve using the solenoid actuator. (Reference NRC Inspection Report 2000-008.)

Stroke timing of the pilot valve is not practical since the test is performed by hand and the pilot valve is a sub-component of the total ERV. Degradation of the pilot valve assemblies is monitored through the preventive maintenance inspections.

<u>Main Valve</u>

A sampling program is proposed that will remove and replace two of the six ERV main valves with pre-tested spare main valves during each RFO, such that all six ERV main valves are replaced every three RFOs (approximately 6 years). Each ERV main valve will be stroke tested at an offsite steam test facility once every 6 years (three RFOs). A 6-month grace period would be allowed to accommodate variations in fuel cycle length and extended shutdown periods. The main valve testing will capture the exercise and stroke time test data required by the ASME OM-2012 Code.

Main valve testing will be performed at an offsite steam test facility. As shown in Figure 1, the main valve is housed in a heavy steel enclosure that is attached to the main steam line inlet flange. The pilot valve assembly is installed inside the pilot valve housing, and the housing is welded onto the outside of the enclosure and physically separated from the ERV main valve body. Thus, only the main valve of the ERV can be sent to the test facility. A spare pilot valve assembly and a spare solenoid actuator, both representative of the components used at the plant, will be installed at the test facility to allow testing the main valve. The valve will be installed on a test steam header in the same orientation as the plant installation. The test conditions at the test facility will be similar to those in the plant, including ambient temperature and steam conditions. The main valve will receive an initial seat leak test, a functional test to

ensure it is capable of operating and closing, and a final seat leak test. Valve stroke time will be obtained during the exercise test. Valve seat tightness will be verified by a cold bar test, and if not free of fog, leakage will be measured and verified to be below specified acceptance criteria. This initial testing will be completed prior to plant startup from the RFO.

After initial testing, the main valves will be completely disassembled, inspected and refurbished, and then retested. The refurbished main valves will be stored at the offsite test facility and returned to the plant prior to the next scheduled use. The offsite test facility's storage requirements will ensure the valves are protected from physical damage. The valves will be stored in an area meeting ANSI/ ASME N45.2.2 Class B storage requirements, with the storage area temperature maintained between 50°F and 90°F. Maintaining the ERVs in a controlled environment during storage minimizes the potential for any valve degradation.

Prior to installation at the plant, the spare main valves will be inspected for foreign material and damage. The steam line and ERV discharge line openings will also be inspected to verify cleanliness and absence of foreign material. Procedural requirements ensure that the proper ERV inlet flange gasket separating ring thickness is provided so proper crush of the flexitallic gasket is achieved when the valve is installed. The valves are then installed and necessary connections completed, including connecting the vent tube and installing the enclosure cover and bellows assembly. Proper connections will be verified per procedure.

The four main valve discs that are not exercised during each RFO will have inspections and maintenance performed on their solenoid actuators and pilot valve assemblies as described above. Review of past surveillance testing and preventive maintenance history indicates that the ERV main valves are highly reliable. During the second 10-year IST interval (1986 to 1999), the ERVs were inspected and refurbished at 48-month intervals (every two RFOs). From 1999 to 2004, the preventive maintenance interval for the ERV main valves was extended to 6 years, and since 2004, the preventive maintenance interval for the ERV main valves has been 10 years. These preventive maintenance activities have found the ERV main valves in excellent condition with no significant degradation noted. The table below lists the most recent preventive maintenance performed for each ERV main valve:

Table 1 ADS-VR-01: ERV Surveillance Test	
Valve Number	Date of Last Preventive Maintenance
PSV-01-102A (ERV-111)	2/17/2017
PSV-01-102B (ERV-112)	2/19/2015
PSV-01-102C (ERV-121)	9/10/2015
PSV-01-102D (ERV-122)	2/25/2015
PSV-01-102E (ERV-113)	2/20/2015
PSV-01-102F (ERV-123)	2/10/2017

Prior to 2011 the plant Technical Specifications (TS) required each ERV to be manually stroked open, once per operating cycle, until the downstream acoustic monitors or

thermocouples indicated the valve was open and passing steam. However, following approval of the license amendment in a Safety Evaluation Report (SER) dated September 28, 2011, Nine Mile Point Nuclear Station, Unit No. 1 – Issuance of Amendment Regarding Changes to Modify Surveillance Requirements for Testing of the Main Steam Electromatic Relief Valves (TAC No. ME4849), the TS only requires manual operation of each ERV solenoid actuator each operating cycle. Therefore, no TS surveillance test data exists to show how the main valve will perform if not stroked, fully open, for a period beyond approximately 48 months, which is the present test interval. The only failure of an ERV to open during the last 20 years is the event reported in NMP1 LER 03-001, which was caused by a problem with the solenoid actuator for ERV-111, not with the main valve.

The proposed sampling program whereby two of the six ERV main valves will be removed and replaced with pre-tested spare main valves during each RFO is consistent with the requirements of ASME OM Code Case OMN-17, Alternative Rules for Testing ASME Class 1 Pressure Relief/Safety Valves. OMN-17 specifies that Class 1 pressure relief valves shall be tested at least once every 72 months (6 years) with a 6-month grace period and that a minimum of 20% of the valves from each valve group shall be tested within any 24-month interval. It also allows that the testing requirements may be satisfied by installing pretested valves to replace valves that have been in service provided the valves removed from service are tested prior to resumption of electric power generation and have been subjected to the specified maintenance.

Additionally, the NRC has previously authorized extensions of the Mandatory Appendix I, 5-year test interval, for testing ASME Class 1 pressure relief valves. For Nine Mile Point Unit 2 (NMP2), the NRC authorized the alternative described in Relief Request MSS-VR-01, Revision 1, to extend the test interval for Class 1 MS safety relief valves to 3 refueling cycles (approximately 6 years, plus 6 months grace), as documented in SER dated January 29, 2016. Also, for the James A. Fitzpatrick plant, the NRC authorized the alternative described in Relief Request VRR-06 Revision 1, to extend the test interval for Class 1 MS safety relief valves to 72 months (6 years) with a 6-month grace period, as documented in SER dated October 1, 2009, and associated license amendment dated July 21, 2010.

Based on the above discussion, extending the main valve exercising interval from every RFO (approximately 2 years) to every 6 years, plus a 6-month grace period, is reasonable and will not adversely impact the ability of the valves to perform their safety functions or result in additional valve failures. The testing and refurbishment activities performed at the off-site test facility on the partial compliment sample (two valves each RFO) will ensure that main valve degradation mechanisms are detected in a timely manner. Monitoring of the ERV discharge line temperatures during plant operation also provides an indication of degradation of the installed main valves.

2. <u>Position Indication Verification Alternative to ISTC-3700 and ISTC-5111</u>

This proposed alternative performs position indication verification for the six ERVs by observing the control room position indicating lights during the solenoid actuator test. Each ERV is equipped with red and green indicating lights, which provide control room open and closed indication, respectively, by monitoring the solenoid actuator plunger position. A blue indicating light is also provided in the control room, which monitors power to the solenoid actuator. The blue light is "On" when the solenoid is deenergized (valve closed) and "Off”

when the solenoid is energized (valve open). As previously noted, the pilot valve operating lever and pilot valve stem will be secured in the open position during this test to prevent damage to the pilot valve assembly, which could result from dry-stroking with no backpressure. Solenoid actuator plunger movement will be observed locally in the drywell and compared to the control room indication to verify that solenoid actuator operation is accurately indicated. The proposed position indication verification alternative provides indirect pilot valve position, which ultimately represents the position of the main valve disc when steam is present, without cycling the ERVs in place with reactor steam pressure. This test is performed every RFO for each of the six ERVs.

The proposed position indication verification alternative provides an acceptable level of quality and safety without requiring indication of main valve obturator movement.

3. <u>Stroke Time Testing Alternative to ISTC-5113 and ISTC-5114</u>

Since the ERVs are not being in-situ tested, and since only the main valve is being tested at the offsite test facility (as previously noted), ERV full stroke time from initiating signal to indication of the end of the operating stroke cannot be obtained. Instead, main valve stroke times will be measured at the test facility. Stroke time acceptance criteria will use a pre-established reference value that represents good performance for this valve type. Since the whole valve assembly is not being tested and the test facility cannot duplicate the control circuitry, a simplified valve actuation circuit will be used. Although these differences may result in minor differences in measured stroke time compared to previous test data for in-situ testing of the complete ERV, the stroke times measured at the test facility will be comparable to each other and, thus, can be used to detect abnormalities in valve performance.

The proposed alternatives described above will maintain acceptable power-operated relief valve test accuracy and continue to provide an acceptable level of quality and safety; therefore, this proposed alternative is being requested pursuant to 10 CFR 50.55a(z)(1).

6. Duration of Proposed Alternative

This request, upon approval, will be applied to the NMP1 fifth 10-year interval, which begins on January 1, 2019, and is scheduled to end on December 31, 2028.

7. Precedence

1. A similar relief request was previously approved for the fourth 10-year interval at NMP1, and all ERVs were removed and tested every four years, as documented in NRC safety evaluation, “Nine Mile Point Nuclear Station, Unit No. 1 – Request for Alternative: Automatic Depressurization System (ADS)-VR-01 for the Testing of Main Steam Electromatic Relief Valves Associated with the Fourth 10-Year Inservice Testing Interval (TAC No. ME4848),” dated September 28, 2011 (ML112660001).

2. The NRC has authorized similar alternatives to the current method of ERV in-situ steam pressure testing, via the following NRC safety evaluations:

- Oyster Creek Nuclear Generating Station – Relief Request [RR-RV-53] Re: Reactor Inservice Testing of Main Steam Electromatic Relief Valves (TAC No. MC8672), dated August 31, 2006 - ADAMS Accession No. ML062220410.
- Dresden Nuclear Power Station, Units 2 and 3 and Quad Cities Nuclear Power Station, Units 1 and 2 – Issuance of Amendments for Main Steam Line Relief Valves and Associated Relief Requests [RV-021 and RV-030E, respectively] (TAC Nos. MC1792 through MC1795), dated October 19, 2004 - ADAMS Accession No. ML042600571 (ML042600563).

NMP1, Dresden, Quad Cities, and Oyster Creek all use the same Dresser model 1525VX solenoid actuated, pilot-operated ERVs. Similarities and differences between the proposed NMP1, alternative and the authorized alternatives for Dresden, Quad Cities, and Oyster Creek are summarized below:

<u>Similarities</u>

- A partial compliment of the ERV main valves will be removed and replaced with pre-tested spare main valves during each RFO.
- Preventive maintenance is performed on all of the ERV solenoid actuators and their associated cutout switches during each RFO.

<u>Differences</u>

- NMP1 classifies the ERVs as OM Category B power-operated valves and tests the ERVs in accordance with the OM Code-2012, Subsection ISTC. The Dresden and Quad Cities relief requests cite both Subsection ISTC and Appendix I of the OM Code for ERV testing requirements, and the Oyster Creek relief request cites only Appendix I of the OM Code.
- The proposed change for NMP1, includes a 6-month grace period on the 6-year interval for steam testing the ERVs at the offsite test facility. For Dresden, Quad Cities, and Oyster Creek, the ERV main valve testing is in accordance with Appendix I of the OM Code, which specifies a test interval of five years.
- For Dresden, Quad Cities, and Oyster Creek, the offsite testing includes both the main valve and the pilot valve. For NMP1, the pilot valve cannot be tested with the main valve because of the unique heavy steel enclosure that houses the main valve. The pilot valve housing is welded onto the outside of the enclosure and is physically separated from the ERV main valve body.
- For Dresden, Quad Cities, and Oyster Creek, the pilot valve assemblies for the ERVs that are not scheduled for removal and offsite testing are replaced with new or refurbished assemblies each RFO. For NMP1, all six of the ERV pilot valve assemblies will be replaced with new or refurbished assemblies each RFO.
- For Dresden, Quad Cities, and Oyster Creek, the pilot valve is dry stroked using the solenoid actuator. For NMP1, dry stroking will not be performed due to a past event where dry stroking caused pilot valve damage. Instead, separate testing, inspections, and maintenance will be performed for the solenoid actuators and the pilot valve assemblies.

8. References

1. Letter from L.A. Hopkins (Niagara Mohawk Power Corporation) to Document Control Desk (NRC), LER 03-001, "Technical Specification Cooldown Rate Exceeded During Required Cooldown for a Failed Solenoid Actuated Pressure Relief Valve," dated June 23, 2003 (ML031880016)

2. Letter from L.A. Hopkins (NMPC) to Document Control Desk (NRC), LER 04-001, "Manual Reactor Scram and Cooldown Rate Exceeding Technical Specification Limits Due to Electromatic Relief Valve Failure to Close," dated July 1, 2004 (ML041950181)

3. Letter from L. A. Hopkins (NMPC) to Document Control Desk (NRC), LER 00-[0]04, Supplement 1, "Manual Reactor Scram and Unusual Event Declaration Due to Stuck Open Electromatic Relief Valve and Failed Vacuum Breaker on Electromatic Relief Valve Discharge Line," dated December 6, 2000 (ML003777259)

4. Letter from M. G. Evans (NRC) to J. H. Mueller (NMPC), "NRC's Nine Mile Point Inspection Report 05000220/2000-008, 05000410/2000-008," dated December 22, 2000 (ML003780274)

5. Letter from T. L. Tate (NRC) to B. Hanson (Exelon Generation Company, LLC), "Nine Mile Point Nuclear Station, Unit 2 – Safety Evaluation by the Office of Nuclear Reactor Regulation for Relief Request MSS-VR-01, Revision 1 (CAC No. MF5773)," dated January 29, 2016 (ML15345A006)

6. Letter from N. L. Salgado (NRC) to Vice President, Operations (Entergy Nuclear Operations, Inc.), "James A. Fitzpatrick Nuclear Power Plant – Relief Request VRR-06, Revision 1 from the Requirements of the OM Code Re: Inservice Testing of Safety Relief Valves (TAC No. ME1818)," dated October 1, 2009 (ML092730032)

7. Letter from B. K. Vaidya (NRC) to Vice President, Operations (Entergy Nuclear Operations, Inc.), "James A. Fitzpatrick Nuclear Power Plant – Issuance of Amendment Regarding Testing of Safety/Relief Valves (TAC No. ME2810)," dated July 21, 2010 (ML101750325)

8. NUREG-0737, "Clarification of TMI Action Item Requirements," Item II.K.3.16, "Reduction of Challenges and Failures of Relief Valves," November 1980 (ML051400209)

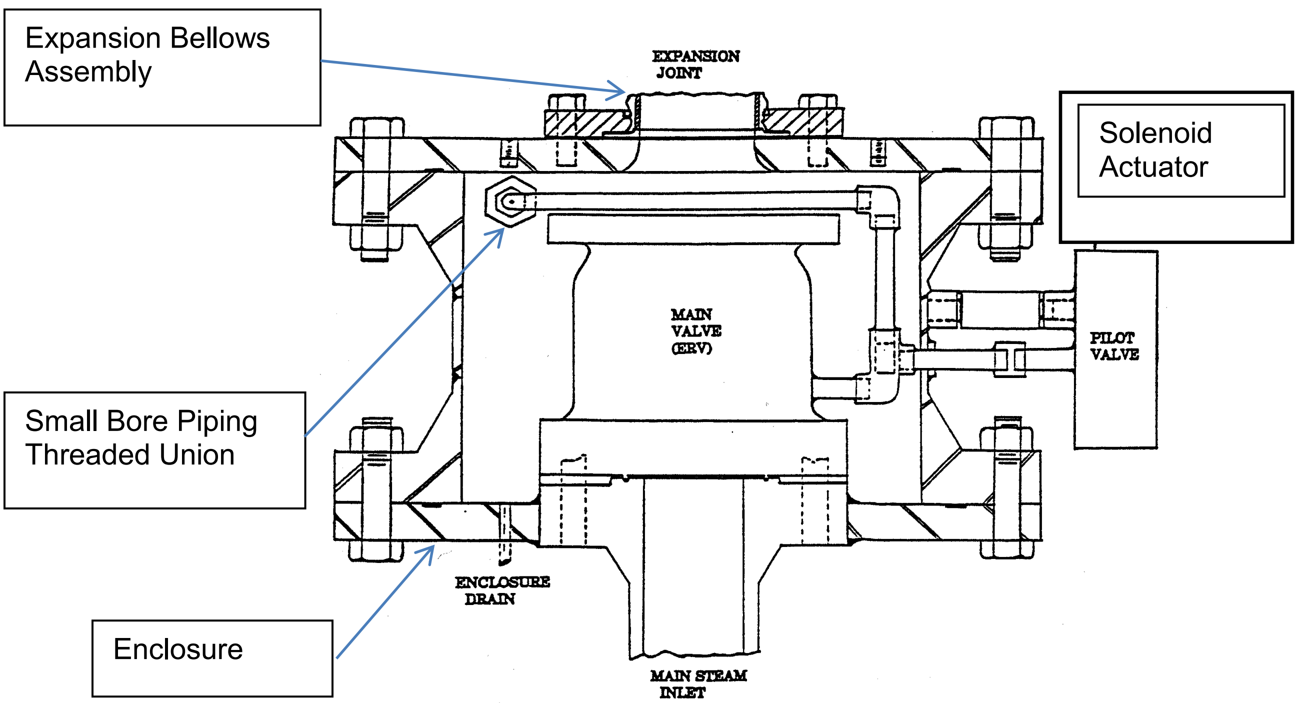
9. ASME OM Code Case OMN-17, Alternative Rules for Testing ASME Class 1 Pressure Relief/Safety Valves

10. ANSI/ ASME N45.2.2, "Packaging, Receiving, Storage, and Handling of Items for Nuclear Power Plants; QA Cases – December 1978"

11. Letter from NRC (R. V. Guzman) to NMPNS (T. A. Lynch), "Nine Mile Point Nuclear Station, Unit No. 1 – Issuance of Amendment [No. 210] Regarding Changes to Modify Surveillance Requirements for Testing of the Main Steam Electromatic Relief Valves (TAC No. ME4849)," dated September 28, 2011 (ML112500067)

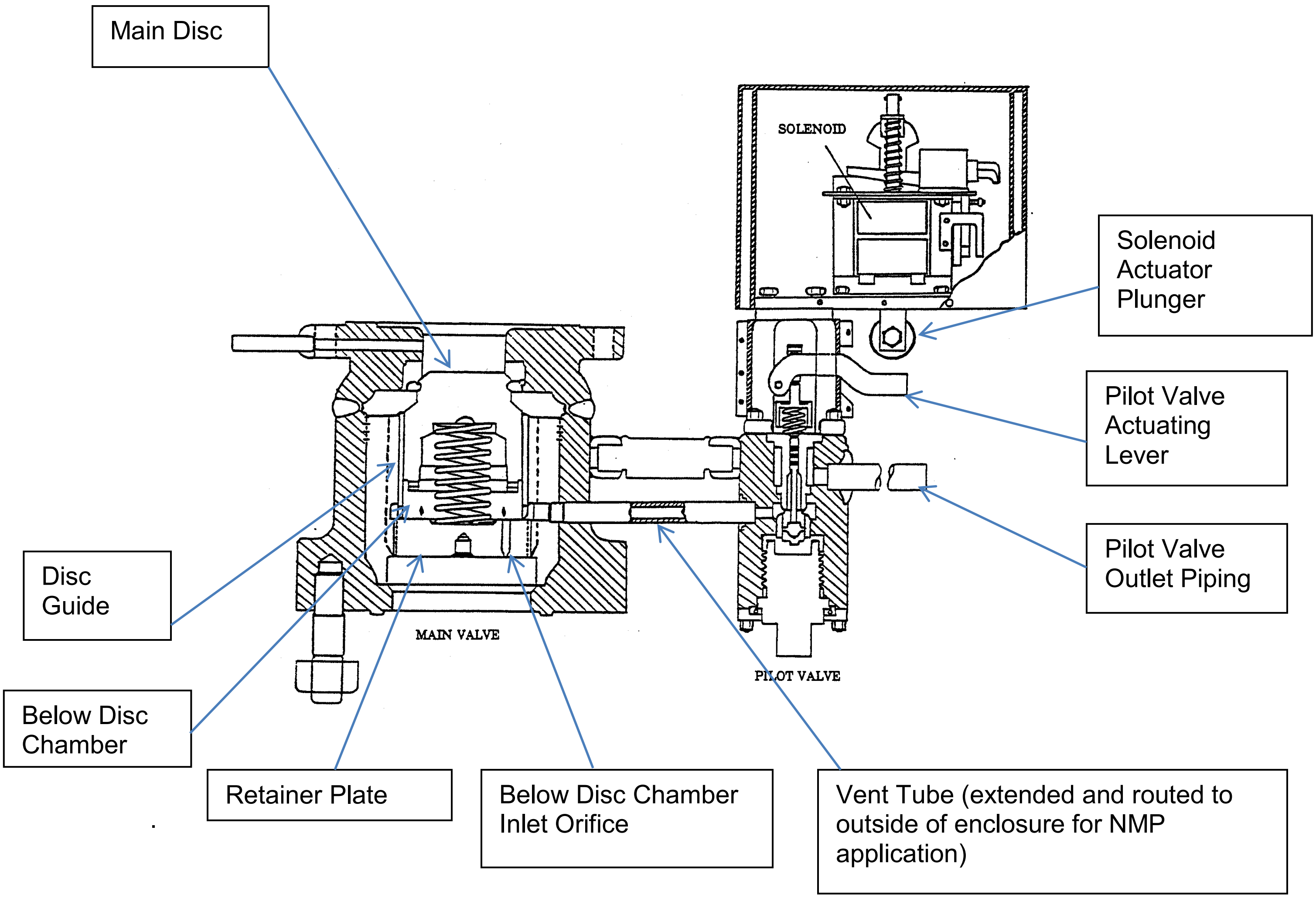
ADS-VR-01 Figure 1

ERV with Enclosure as Installed at NMPNS, Unit 1



ADS-VR-01 Figure 1 (Continued)

Standard ERV similar to other Plants
(Cutaway view of internals is applicable to NMP1)



EXELON GENERATION COMPANY, LLC
IST PROGRAM - RELIEF REQUEST
Nine Mile Point Nuclear Station, Unit 1
Proposed Alternative in Accordance with 10 CFR 50.55a(z)(1),
CRD-VR-01 – Stroke Time Testing of SCRAM Discharge Volume Valves

1. ASME Code Component(s) Affected

The following Scram Discharge Volume (SDV) containment isolation valves (CIVs) are affected:

Component	Description	Class	Category
IV-44.2-15	SDV VENT INBOARD IV	2	A
IV-44.2-16	SDV VENT OUTBOARD IV	2	A
IV-44.2-17	SDV DRAIN OUTBOARD IV	2	A
IV-44.2-18	SDV DRAIN INBOARD IV	2	A

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 with no Addenda.

3. Applicable Code Requirement

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

4. Reason for Request

Pursuant to 10 CFR 50.55a, *Codes and standards*, paragraph (z)(1), an alternative is proposed to the power-operated valve test requirements of the ASME OM-2012 Code. The basis of the request is that the proposed alternative would provide an acceptable level of quality and safety.

The SDV CIVs are normally open valves. These valves close on the loss of air or the de-energizing of the solenoid valves (SOV-113-275 and SOV-113-276 for IV-44.2-16 and IV-44.2-17; and SOV-113-273 and SOV-113-274 for IV-44.2-15 and IV-44.2-18). The SDV air header and valve arrangement are single failure proof. The solenoid valves are powered from either reactor trip bus (RTB) 131 or 141 through fuses. Removing the fuses to fail-safe test these valves causes a scram in approximately six (6) seconds due to the de-energizing of SOV-113-271 and SOV-113-272. Venting the scram air header due to exercising of the valves by pulling fuses subjects the control rod drives to higher differential pressures than observed during a scram at normal operating conditions. The high differential pressure applied to control rods fully inserted has resulted in equipment damage.

Testing via the safety-related scram exhaust path cannot be performed during power operation since this could result in a plant trip. The safety-related exhaust path (scram path) is through SOV-113-275 and SOV-113-276 or SOV-113-273 and SOV-113-274 exhaust ports. A test solenoid valve (SOV-113-277) was installed as a result of Information Bulletin (IEB) No. 80-17 dated July 3, 1980, to permit fail-safe and stroke time

testing without causing a scram. The test solenoid exhaust path (test path) adds a restriction that is not present in the scram path. When the test solenoid is energized, the SDV air header and valve actuators are vented through SOV-113-277.

The restriction is due to exhausting air through the SOV-113-274 and SOV-113-276 air inlet supply port, since the solenoids are energized. The solenoid valve employs an internal pilot in the inlet port. Air can exhaust through the inlet port; however, the flow path is not a fixed resistance path. The variable resistance can cause variations in the quarterly stroke time measurements of the valves. These variations can result in inaccurate stroke times and mask the true valve performance. This limits the ability to accurately monitor for and detect degradation. Additionally, the test path is not the safety-related exhaust path (scram path) for the CIVs.

Stroke time testing through the scram path can be performed during refueling outages. Stroke times obtained during refueling outage tests (using the scram vent path) have provided consistent accurate results. This testing method provides an accurate indication of valve performance and provides the ability to monitor for and detect degradation.

5. Proposed Alternative and Basis for Use

The SDV CIVs will be full stroke exercised and fail safe tested quarterly using the test solenoid valve. These valves will be stroke-time tested through the scram path during refueling outages.

The basis of the request is that the proposed alternative would provide an acceptable level of quality and safety; therefore, the alternative testing is proposed pursuant to 10 CFR 50.55a(z)(1).

6. Duration of Proposed Alternative

This request, upon approval, will be applied to the NMPNS, Unit No. 1, fifth 10-year interval, which begins on January 1, 2019, and is scheduled to end on December 31, 2028.

7. Precedent

This relief request was previously approved for the fourth 10-year interval at NMPNS, Unit No. 1, as documented in NRC safety evaluation, “Nine Mile Point Nuclear Station – Safety Evaluation of Relief Requests [CRD-VR-01] for the Unit No. 1 Fourth 10-Year and Unit No. 2 Third 10-Year Pump and Valve Inservice Testing Program (TAC Nos. MD9202 and MD9203),” dated December 29, 2008 (ML083500039).

8. Reference

NRC Information Bulletin (IEB) No. 80-17, Failure of 76 of 185 Control Rods to Fully Insert During Scram at a BWR,” dated July 3, 1980 (ML8005050076)

1. ASME Code Component(s) Affected

The following Hydrogen and Oxygen sample and return containment isolation valves (CIVs) are affected:

Component	Description	Class	Category
IV-201.2-109	#11 TORUS RETURN INBOARD IV	2	A
IV-201.2-110	#11 TORUS SAMPLE INBOARD IV	2	A
IV-201.2-111	#11 TORUS SAMPLE OUTBOARD IV	2	A
IV-201.2-112	#11 TORUS RETURN OUTBOARD IV	2	A
IV-201.7-01	#11 SAMPLE STREAM B INBOARD IV	2	A
IV-201.7-02	#11 SAMPLE STREAM B OUTBOARD IV	2	A
IV-201.7-08	DW CAM SAMPLE INBOARD IV	2	A
IV-201.7-09	DW CAM SAMPLE OUTBOARD IV	2	A
IV-201.7-10	#11 DW RETURN INBOARD IV	2	A
IV-201.7-11	#11 DW RETURN OUTBOARD IV	2	A

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 with no Addenda.

3. Applicable Code Requirement

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

ISTC-5132, Stroke Test Acceptance Criteria, states, "Test results shall be compared to the reference values established in accordance with para. ISTC-3300, ISTC-3310, or ISTC-3320."

4. Reason for Request

Pursuant to 10 CFR 50.55a, *Codes and standards*, paragraph (z)(1), an alternative is proposed to the power-operated valve test requirements of the ASME OM-2012 Code. The basis of the request is that the proposed alternative would provide an acceptable level of quality and safety.

These pneumatically operated valves are grouped together on common control switches. The groups are:

- IV-201.7-08, IV-201.7-09, IV-201.7-10, & IV-201.7-11
- IV-201.2-109, IV-201.2-112, IV-201.2-110, IV-201.2-111, IV-201.7-01, & IV-201.7-02

These arrangements have a common closed light (green) for a group of valves and individual open lights (red) for each valve. Reference values are established for each group by timing the valves for at least three exercises. The exercising is conducted over a sufficient interval to prevent erroneous data due to pre-conditioning. An individual reference value is developed for each valve in a group. A composite (group) reference value is developed by averaging the individual reference values. Typically, the individual valve's reference values are within ± 0.5 second of the group reference value.

As needed, primarily after rework or repair, the individual reference values and the group reference value are re-established. This group reference value is used as a common reference value for each valve in the group. The valve stroke-time test uses switch-actuation-to-red-light-out (closed indication) for open-to-close stroke time. The stroke-time of the slowest valve is observed and recorded. Typically, the slowest valve is not always the same component within the group. If the slowest valve exceeds the acceptance criterion (i.e., ± 50% of the group reference value), the group is declared inoperable. Corrective action is then taken, per ISTC-5133, Stroke Test Corrective Action.

The group reference values are less than 10 seconds, significantly below the Updated Final Safety Analysis Report (UFSAR), Table VI-3b, Primary Containment Isolation Valves Lines Entering Free Space of the Containment, maximum operating time of 60 seconds. While some performance degradation is masked by this testing methodology, nuclear safety will not be compromised. Prior to any valve degrading and exceeding the UFSAR maximum operating time of 60 seconds, the acceptance criterion would be significantly exceeded, and corrective action would be taken. The proposed alternate testing method provides an adequate capability to monitor and detect individual valve degradation prior to exceeding the UFSAR maximum operating time. This method provides an equivalent level of quality and safety compared to the Code required individual valve stroke-timing.

5. Proposed Alternative and Basis for Use

NMPNS proposes to establish individual reference values, group reference values, and group acceptance criteria. Stroke-timing of the valve groups will record the slowest operating valves' corresponding stroke-time. NMPNS will then compare the slowest valve stroke-time to the acceptance criterion to determine the valve group operability status. Corrective actions will be taken, as required, for exceeding the acceptance criterion.

Therefore, relief is requested pursuant to 10 CFR 50.55a(z)(1) based on the proposed alternative providing an acceptable level of quality and safety.

6. Duration of Proposed Alternative

This request, upon approval, will be applied to the NMPNS, Unit No. 1, fifth 10-year interval, which begins on January 1, 2019, and is scheduled to end on December 31, 2028.

7. Precedent

This relief request was previously approved for the fourth 10-year interval at NMPNS, Unit No. 1, as documented in NRC safety evaluation, "Nine Mile Point Nuclear Station – Safety Evaluation of Relief Requests [CTNH202-VR-01] for the Unit No. 1 Fourth 10-Year and Unit No. 2 Third 10-year Pump and Valve Inservice Testing Program (TAC Nos. MD9202 and MD9203)," dated December 29, 2008 (ML083500039).

EXELON GENERATION COMPANY, LLC
IST PROGRAM - RELIEF REQUEST
Nine Mile Point Nuclear Station, Unit 1
Proposed Alternative in Accordance with 10 CFR 50.55a(z)(1),
CTNH202-VR-02 - H2/O2 Sample and Return Valve Stroke Time Testing Group 2

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

The following Hydrogen and Oxygen sample and return containment isolation valves (CIVs) are affected:

Component	Description	Class	Category
IV-201.2-23	#12 TORUS SAMPLE INBOARD IV	2	A
IV-201.2-24	#12 TORUS SAMPLE OUTBOARD IV	2	A
IV-201.2-29	#12 DRYWELL SAMPLE INBOARD IV	2	A
IV-201.2-30	#12 DRYWELL SAMPLE OUTBOARD IV	2	A

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 with no Addenda.

3. **Applicable Code Requirement**

ISTC-5151, Valve Stroke Testing, states, in part, "*(a)* Active valves shall have their stroke times measured when exercised in accordance with para. ISTC-3500."

ISTC-5152, Stroke Test Acceptance Criteria, states, "Test results shall be compared to the reference values established in accordance with para. ISTC-3300, ISTC-3310, or ISTC-3320."

4. **Reason for Request**

Pursuant to 10 CFR 50.55a, *Codes and standards*, paragraph (z)(1), an alternative is proposed to the power-operated valve test requirements of the ASME OM-2012 Code. The basis of the request is that the proposed alternative would provide an acceptable level of quality and safety.

These solenoid operated valves are grouped together on a common control switch. The group is:

- IV-201.2-23, IV-201.2-24, IV-201.2-29, & IV-201.2-30

This arrangement has a common closed light (green) for each pair of valves and individual open lights (red) for each valve. A reference value is established for each pair by timing the valves for at least three exercises. The exercising is conducted over a sufficient interval to prevent erroneous data due to pre-conditioning. A composite (group) reference value is developed by averaging the valve pair reference values. Individual reference values are not established. These valves stroke in less than 2 seconds and are all

designated as “rapid acting” valves. A limiting value of 2 seconds is assigned to the group.

As needed, primarily after rework or repair, the individual reference values and the group reference value are re-established. This group reference value is used as a common reference value for each valve in the group. The valve stroke-time test uses switch-actuation-to-red-light-out (closed indication) for open-to-close stroke time. The stroke-time of the slowest valve is observed and recorded. Typically, the slowest valve is not always the same component within the group. If the slowest valve exceeds the acceptance criterion (i.e., ± 50% of the group reference value), the group is declared inoperable. Corrective action is then taken, per ISTC-5133, Stroke Test Corrective Action.

The group limiting value of 2 seconds is significantly below the Updated Final Safety Analysis Report (UFSAR), Table VI-3b, Primary Containment Isolation Valves Lines Entering Free Space of the Containment, maximum operating time of 60 seconds. While some performance degradation is masked by this testing methodology, nuclear safety will not be compromised. Prior to any valve degrading and exceeding the UFSAR maximum operating time of 60 seconds, the acceptance criterion would be significantly exceeded, and corrective action would be taken. The proposed alternate testing method provides an adequate capability to monitor and detect individual valve degradation prior to exceeding the UFSAR maximum operating time of 60 seconds. This method provides an equivalent level of quality and safety compared to the Code required individual valve stroke-timing.

5. Proposed Alternative and Basis for Use

NMPNS proposes to establish valve pair reference values, group reference values, and group acceptance criteria. Stroke-timing of the valve groups will record the slowest operating valve’s corresponding stroke-time. NMPNS will then compare the slowest valve stroke-time to the acceptance criterion to determine the valve group operability status. Corrective actions will be taken, as required, for exceeding the acceptance criterion.

Therefore, relief is requested pursuant to 10 CFR 50.55a(z)(1) based on the proposed alternative providing an acceptable level of quality and safety.

6. Duration of Proposed Alternative

This request, upon approval, will be applied to the NMPNS, Unit No. 1, fifth 10-year interval, which begins on January 1, 2019, and is scheduled to end on December 31, 2028.

7. Precedent

This relief request was previously approved for the fourth 10-year interval at NMPNS, Unit No. 1, as documented in NRC safety evaluation, “Nine Mile Point Nuclear Station – Safety Evaluation of Relief Requests [CTNH202-VR-02] for the Unit No. 1 Fourth 10-Year and Unit No. 2 Third 10-Year Pump and Valve Inservice Testing Program (TAC Nos. MD9202 and MD9203),” dated December 29, 2008 (ML083500039).

1. ASME Code Component(s) Affected

The following Reactor Pressure Vessel Safety Valves are affected:

Component	Description	Class	Category
PSV-01-119A	Reactor Pressure Vessel Safety Valve	1	C
PSV-01-119B	Reactor Pressure Vessel Safety Valve	1	C
PSV-01-119C	Reactor Pressure Vessel Safety Valve	1	C
PSV-01-119D	Reactor Pressure Vessel Safety Valve	1	C
PSV-01-119F	Reactor Pressure Vessel Safety Valve	1	C
PSV-01-119G	Reactor Pressure Vessel Safety Valve	1	C
PSV-01-119H	Reactor Pressure Vessel Safety Valve	1	C
PSV-01-119J	Reactor Pressure Vessel Safety Valve	1	C
PSV-01-119M	Reactor Pressure Vessel Safety Valve	1	C

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 with no Addenda.

3. Applicable Code Requirement

Division 1, Mandatory Appendix I, Inservice Testing of Pressure Relief Devices in Light-Water Reactor Nuclear Power Plants, paragraph I-1320, Test Frequencies, Class 1 Pressure Relief Valves, subparagraph *(a) 5-Yr Test Interval*, states:

"Class 1 pressure relief valves shall be tested at least once every 5 yr, starting with initial electric power generation. No maximum limit is specified for the number of valves to be tested within each interval; however, a minimum of 20% of the valves from each valve group shall be tested within any 24-mo interval. This 20% shall consist of valves that have not been tested during the current 5-yr interval, if they exist. The test interval for any installed valve shall not exceed 5 yr. The 5-yr test interval shall begin from the date of the as-left set pressure test for each valve."

4. Reason for Request

Pursuant to 10 CFR 50.55a, *Codes and standards*, paragraph (z)(1), an alternative is proposed to the Class 1 pressure relief valve test requirements of the ASME OM Code Mandatory Appendix I, I-1320(a). The basis of the request is that the proposed alternative would provide an acceptable level of quality and safety.

The nine (9) reactor pressure vessel safety valves provide Code-required overpressure protection to the reactor pressure vessel and the Class 1 reactor recirculation system and are located on the reactor vessel head inside the primary containment. In the event of main steam isolation valve (MSIV) closure, the safety valves are designed and sized to limit the pressure rise to 110% of the design pressure.

The Dresser Model 3777QA, spring-loaded safety valves have shown exemplary test history at NMPNS, Unit 1 (NMP1), as shown in MS-VR-01 Table 1, Reactor Head Safety Valve Test Results. However, given the current 24-month operating cycle, NMP1 is required to remove and test approximately fifty (50) percent of the safety valves every refueling outage (i.e., alternating between either four or five of the nine each outage), so that all valves are removed and tested every two refueling outages. This ensures compliance with the ASME OM Code requirements for testing Class 1 pressure relief valves within a 5-year interval. Approval of extending the test interval to 6 years with a grace period of six months would reduce the number of safety valves tested at NMP1 over three (3) refueling outages by at least four.

Code Case OMN-17, Alternative Rules for Testing ASME Class 1 Pressure Relief/Safety Valves, was published in the ASME OM-2009 Code Edition. This Code Case has not been approved for use in Regulatory Guide (RG) 1.192, *Operations and Maintenance Code Case Acceptability, ASME OM Code*, nor incorporated in 10 CFR 50.55a by reference; however, the NRC has allowed licensees to use ASME Code Case OMN-17, provided all requirements are met. Code Case OMN-17 allows the Owner to extend the test frequencies for Class 1 pressure relief valves to a 72-month (6-year) test interval, with a 6-month grace period, providing the requirements of the Code Case are satisfied. The Code Inquiry from Code Case OMN-17, references, in part, ASME OM Code 2001 Edition through the 2006 Addenda of Mandatory Appendix I, paragraph I-1320. NMP1 is preparing to implement the ASME OM-2012 Code Edition and has verified no revisions were made to Mandatory Appendix I in this edition, since the OMa-2006 Addenda of the ASME OM Code, that would affect the implementation of Code Case OMN-17. NMP1 currently meets or exceeds all the requirements specified in Code Case OMN-17.

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

As an alternative to the Code-required 5-year test interval per Mandatory Appendix I, paragraph I-1320(a), Exelon proposes that the subject Class 1 Reactor Pressure Vessel Safety Valves be tested at least once every three (3) refueling cycles (approximately 6 years/72 months) with a minimum of 20% of the valves tested within any 24-month interval. This 20% would consist of valves that have not been tested during the current 72-month interval, if they exist. The test interval for any individual valve would not exceed the 72 months, except that a 6-month grace period is allowed to coincide with refueling outages to accommodate extended shutdown periods and certification of the valve prior to installation.

After as-found set pressure testing, the valves shall be disassembled and inspected to verify that parts are free of defects resulting from time-related degradation or service induced wear. As-left set pressure testing shall be performed following maintenance and

prior to returning the valve(s) to service. Each valve shall be disassembled and inspected prior to the start of the 72-month interval. Disassembly and inspection prior to the implementation of Code Case OMN-17 may be used.

The safety valve testing and maintenance cycle at NMP1 consists of removal of the safety valve complement requiring testing and transportation to an off-site test facility. Upon receipt at the off-site facility, the valves are subject to an as-found inspection, seat leakage and set pressure testing. Prior to the return of a complement of safety valves for installation in the plant, the valves are disassembled and inspected to verify the internal surfaces and parts are free from defects resulting from time related degradation or service induced wear prior to the start of the next test interval. During this process, any identified adverse conditions are corrected; damaged or worn parts, springs, gaskets and seals are replaced and the valve seats are lapped, if necessary; and the valve is reassembled. Following reassembly, the valve's set pressure is recertified with an acceptance criterion of ±1%. This existing process is in accordance with ASME OM Code Case OMN-17, paragraphs (d) *Maintenance* and (e) *Disassembly and Inspection*.

After recertification testing, the safety valves are stored at the test facility for future use. The storage area is inspected and maintained to the requirements of ANSI/ASME N45.2.2, *Packing, Handling, Shipping, Storage and Handling of Items for Nuclear Power Plants*, which will minimize the potential for any valve degradation.

NMP1 has reviewed the as-found set pressure test results for all of the safety valves tested since March 2011 as shown in Attachment 1. No safety valve tested exceeded the as-found set pressure test acceptance criterion of ±3%. However, as required by Code Case OMN-17 paragraph (c), *Requirements for Testing Additional Valves (same as* Appendix I, I-1320(c)), NMP1 will expand the scope of safety valve testing upon an as-found set-pressure (first test actuation) failure. Specifically, for each valve tested for which the as-found set-pressure (first test actuation) exceeds the greater of either the plus/minus tolerance limit of the Owner established set-pressure acceptance criteria or ±3% of valve nameplate set-pressure, two additional valves shall be tested from the same valve group.

Based on the valve performance history and safety valve maintenance practices, there is continued assurance of valve operational readiness, consistent with ASME OM-2012 Code, Mandatory Appendix I. Therefore, the proposed alternative for testing the subject Class 1 safety valves at least once every three (3) refueling cycles (approximately 6 years/72 months) with a minimum of 20% of the valves tested within any 24-month interval, including a 6-month grace period, would maintain acceptable valve operational readiness and provide an acceptable level of quality and safety; pursuant to 10 CFR 50.55a(z)(1).

6. Duration of Proposed Alternative

This request, upon approval, will be applied to the NMP1 Fifth 10-Year Interval, which begins on January 1, 2019, and is scheduled to end on December 31, 2028.

7. Precedence

1. A similar relief request was authorized for use during the third 10-year interval at NMP2, as documented in letter from NRC (T. L. Tate) to Exelon Generation Company, LLC (B. Hanson), "Nine Mile Point Nuclear Station, Unit 2 – Safety Evaluation by the Office of the Nuclear Reactor Regulation for Relief Request MSS-VR-01, Revision 1 (CAC No. MF5773)," dated January 29, 2016 (ML15345A006). *[Authorized the 6-month grace afforded by Code Case OMN-17.]*

2. A similar relief request was authorized for use via MSS-VR-01 during the third 10-year interval for NMP2, as documented in letter from NRC (M. G. Kowal) to Nine Mile Point Nuclear Station, LLC (K. J. Polson), "Nine Mile Point Nuclear Station – Safety Evaluation of Relief Requests [MSS-VR-01] for the Unit No. 1 Fourth 10-Year and Unit No. 2 Third 10-Year Pump and Valve Inservice Testing Program (TAC Nos. MD9202 and MD9203)," dated December 29, 2008 (ML083500039). *[Authorized the Extended Test Interval to 6 years for Safety Valve testing.]*

3. Peach Bottom Atomic Power Station, Units 2 and 3 – Safety Evaluation of Relief Request 01A-VRR-3 Regarding the Fourth 10-Year Interval of the Inservice Testing Program (TAC Nos. MF2509 and MF2510), dated April 30, 2014 (ML14094A051).

8. References

1. ASME OM Code Case OMN-17, Alternative Rules for Testing ASME Class 1 Pressure Relief/Safety Valves

2. ANSI/ASME N45.2.2, Packing, Handling, Shipping, Storage and Handling of Items for Nuclear Power Plants

3. RG 1.192, *Operations and Maintenance Code Case Acceptability, ASME OM Code*, Revision 1, dated August 2014

MS-VR-01 Table 1
Reactor Pressure Vessel Safety Valve Test Results

Safety Valve Tested	**Serial No.**	**Setpoint (psig)**	**As-Found Setpoint Test Results (psig)**	**Max Set Pressure +3% (psig)**	**Min Set Pressure-3% (psig)**	**Accept/ Reject**	**Time from last test (Years)**
Refueling Outage 24, March 2017							
PSV-01-119C	BK6524	1218	1200	1254.5	1181.4	Accept	4
PSV-01-119F	BK6254	1245	1239	1282.3	1201.8	Accept	4
PSV-01-119H	BR08508	1218	1198	1254.5	1181.4	Accept	4
PSV-01-119J	BK6325	1227	1226	1263.8	1190.1	Accept	4
Refueling Outage 23, March 2015							
PSV-01-119A	BK6267	1236	1214	1273.0	1198.9	Accept	2
PSV-01-119B	BK6317	1227	1220	1263.8	1190.1	Accept	2
PSV-01-119D	BL6303	1254	1233	1291.6	1219.3	Accept	2
PSV-01-119G	BL6280	1236	1225	1273.0	1198.9	Accept	2
PSV-01-119M	BK6250	1218	1214	1254.5	1181.4	Accept	2
Refueling Outage 22, March 2013							
PSV-01-119A	BK6535	1236	1239	1273.0	1198.9	Accept	2
PSV-01-119B	BK6253	1227	1195	1263.8	1190.1	Accept	2
PSV-01-119C	BK6520	1218	1236	1254.5	1181.4	Accept	2
PSV-01-119D	BK6292	1254	1248	1291.6	1219.3	Accept	2
PSV-01-119F	BK6297	1245	1230	1282.3	1201.8	Accept	2
PSV-01-119G	BK6256	1236	1203	1273.0	1198.9	Accept	2
PSV-01-119H	BK6521	1218	1235	1254.5	1181.4	Accept	2
PSV-01-119J	BK6319	1227	1196	1263.8	1190.1	Accept	2
PSV-01-119M	BK6522	1218	1194	1254.5	1181.4	Accept	2
Refueling Outage 21, March 2011							
PSV-01-119A	BK6267	1236	1273	1273.0	1198.9	Accept	2
PSV-01-119B	BK6317	1227	1259	1263.8	1190.1	Accept	2
PSV-01-119C	BK6524	1218	1223	1254.5	1181.4	Accept	2
PSV-01-119D	BK6303	1254	1268	1291.6	1219.3	Accept	2
PSV-01-119F	BK6254	1245	1251	1282.3	1201.8	Accept	2
PSV-01-119G	BK6280	1236	1235	1273.0	1198.9	Accept	2
PSV-01-119H	BR08508	1218	1245	1254.5	1181.4	Accept	2
PSV-01-119J	BK6325	1227	1239	1263.8	1190.1	Accept	2
PSV-01-119M	BK6250	1218	1246	1254.5	1181.4	Accept	2

Comments for MS-VR-01 Table 1:

Maintenance History: Prior to installation, each valve was fully disassembled, inspected, reassembled and as-left recertified by NWS Technologies, Spartanburg, South Carolina.

- Number of as-found setpoint tests results acceptable: 27
- Number of as-found setpoint tests results unacceptable: 0

EXELON GENERATION COMPANY, LLC
IST PROGRAM - RELIEF REQUEST
Nine Mile Point Nuclear Station, Unit 1
Proposed Alternative in Accordance with 10 CFR 50.55a(z)(2),
RBCLC-PR-01 – Alternate Frequency for Pump Testing

1. ASME Code Component(s) Affected

The following Reactor Building Closed Loop Cooling (RBCLC) Pumps are affected:

Component	Description	Class	Group
PMP-70-01	Reactor Building Closed Loop Cooling Water #11	3	A
PMP-70-02	Reactor Building Closed Loop Cooling Water #12	3	A
PMP-70-03	Reactor Building Closed Loop Cooling Water #13	3	A

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 with no Addenda.

3. Applicable Code Requirement

ISTB-3400, Frequency of Inservice Tests, states "An inservice test shall be run on each pump as specified in Table ISTB-3400-1."

Table ISTB-3000-1, Inservice Test Parameters, provides the parameters for Flow Rate (Q) and Differential Pressure (ΔP) for Group A pump testing.

Table ISTB-3400-1, Inservice Test Frequency, provides Group A Test frequency for Group A pumps as "Quarterly".

ISTB-5121, Group A Test Procedure, states, in part, that "Group A tests shall be conducted with the pump operating as close as practical to a specified reference point and within the variances from the reference point as described in this paragraph. The test parameters shown in Table ISTB-3000-1 shall be determined and recorded as required by this paragraph.

4. Reason for Request

Pursuant to 10 CFR 50.55a, *Codes and standards*, paragraph (z)(2), an alternative is proposed to the pump test requirements of the ASME OM-2012 Code. The basis of the request is that compliance with the Code requirements results in hardship or unusual difficulty with no compensating increase in the level of quality and safety.

The RBCLC system is not a fixed resistance system. For the RBCLC system, no pump test loops or individual pump flow instrumentation is installed. Individual pump flow can only be determined by measuring system flow rate. The system flow rate and differential pressure are a function of the number of pumps running and system heat loads. During normal plant operations, system heat loads prevent removing the RBCLC system from service. Operating conditions do not permit single pump operation at repeatable test

conditions to allow individual pump parameters (i.e., flow rate and differential pressure) to be measured.

Therefore, during normal plant operation, operating a single RBCLC pump at a fixed reference condition (per ISTB-5121) to perform a Group A test (per ISTB-3400) would require reducing system heat loads and may result in a plant shutdown to cold shutdown conditions. Complying with the Code would require NMP1, to enter cold shutdown conditions every quarter where RBCLC system operating conditions allow single pump operation. Cold shutdown reduces system heat loads sufficiently to allow single RBCLC pump operation at a fixed reference condition and thus allows measurement of individual pump parameters (i.e., flow rate and differential pressure).

Obtaining flow rate and differential pressure measurements (parameters required by Table ISTB-3000-1 for an individual RBCLC pump on a quarterly basis poses a significant hardship (plant shutdown).

Alternatively, compliance could be achieved by a major system redesign and modification such as installation of individual pump test loops with flow instrumentation. This would allow a single pump to be removed from the system flow path and operated on a test flow path at Code required fixed reference conditions. Such a major system modification would be costly and burdensome with no compensating increase in the level of quality or safety.

5. Proposed Alternative and Basis for Use

Quarterly, during normal system operation, vibration (V) shall be measured for each RBCLC pump. During cold shutdowns, all the applicable parameters for a Group A test from Table ISTB-3000-1 (flow rate (Q), vibration (V), and differential pressure (ΔP)) shall be measured for each RBCLC pump. The comprehensive test specified in Table ISTB-3400-1 will also be performed biennially. The testing alternative described above will still allow an adequate determination of pump operational readiness and permit detection of component degradation.

Therefore, relief is requested pursuant to 10 CFR 50.55a(z)(2) based on the determination that compliance with the Code required Group A pump test requirements cannot be achieved without major system modifications resulting in a hardship or unusual difficulty without a compensating increase in the level of quality and safety and the proposed alternative provides reasonable assurance of pump operational readiness.

6. Duration of Proposed Alternative

This request, upon approval, will be applied to the NMP1, fifth 10-year interval, which begins on January 1, 2019, and is scheduled to end on December 31, 2028.

7. Precedent

This relief request was previously approved for the fourth 10-year interval at NMP1, as documented in NRC safety evaluation, "Nine Mile Point Nuclear Station – Safety Evaluation of Relief Requests [RBCLC-PR-01] for the Unit No. 1 Fourth 10-Year and Unit No. 2 Third 10-Year Pump and Valve Inservice Testing Program (TAC Nos. MD9202 and MD9203)," dated December 29, 2008 (ML083500039).

1. ASME Code Component(s) Affected

The following main steam safety relief valves (MSSRVs) are affected:

Component	Description	Class	Category
2MSS*PSV120	MAIN STEAM SRV	1	C
2MSS*PSV121	MAIN STEAM SRV (ADS)*	1	C
2MSS*PSV122	MAIN STEAM SRV	1	C
2MSS*PSV123	MAIN STEAM SRV	1	C
2MSS*PSV124	MAIN STEAM SRV	1	C
2MSS*PSV125	MAIN STEAM SRV	1	C
2MSS*PSV126	MAIN STEAM SRV (ADS)	1	C
2MSS*PSV127	MAIN STEAM SRV (ADS)	1	C
2MSS*PSV128	MAIN STEAM SRV	1	C
2MSS*PSV129	MAIN STEAM SRV (ADS)	1	C
2MSS*PSV130	MAIN STEAM SRV (ADS)	1	C
2MSS*PSV131	MAIN STEAM SRV	1	C
2MSS*PSV132	MAIN STEAM SRV	1	C
2MSS*PSV133	MAIN STEAM SRV	1	C
2MSS*PSV134	MAIN STEAM SRV (ADS)	1	C
2MSS*PSV135	MAIN STEAM SRV	1	C
2MSS*PSV136	MAIN STEAM SRV	1	C
2MSS*PSV137	MAIN STEAM SRV (ADS)	1	C

*ADS = Automatic Depressurization System

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 with no Addenda.

3. Applicable Code Requirement

Mandatory Appendix I, Inservice Testing of Pressure Relief Devices in Light-Water Reactor Nuclear Power Plants, paragraph I-1320, Test Frequencies, Class 1 Pressure Relief Valves, (a) *5-Yr Test Interval*, states, “Class 1 pressure relief valves shall be tested at least once every 5 yr, starting with initial electric power generation. No maximum limit is

specified for the number of valves to be tested within each interval; however, a minimum of 20% of the valves from each valve group shall be tested within any 24-mo interval. This 20% shall consist of valves that have not been tested during the current 5-yr interval, if they exist. The test interval for any installed valve shall not exceed 5 yr. The 5-yr test interval shall begin from the date of the as-left set pressure test for each valve."

4. Reason for Request

Pursuant to 10 CFR 50.55a, *Codes and standards*, paragraph (z)(1), an alternative is proposed to the relief valve requirements of the ASME OM Code listed above. The basis of the request is that the proposed alternative would provide an acceptable level of quality and safety.

Appendix I, Section I-1320(a) of the ASME OM-2012 Code states, in part, that Class 1 pressure relief valves shall be tested at least once every 5 years, starting with initial electric power generation. This section also states a minimum of 20% of the pressure relief valves are tested within any 24-month interval and that the test interval for any individual valve shall not exceed 5 years. The required tests ensure that the Safety Relief Valves (SRVs), which are located on each of the Main Steam (MS) lines between the reactor vessel and the first isolation valve within the drywell, will open at the pressures assumed in the safety analysis.

The SRVs are Dikkers / Model G471-6/125.04 (18 SRVs installed) in four (4) steam lines. The SRVs are designed to actuate by either of two modes - the safety mode or the ADS mode. In the safety mode, the valve will open when reactor pressure exceeds a specific spring set-pressure. In the ADS mode, the valve will automatically open upon receipt of an overpressure signal (seven of the eighteen valves support this ADS function). The proposed changes do not impact either the safety mode of operation or the ADS mode of operation.

The SRVs have shown acceptable test history at NMPNS Unit 2 (NMP2) as described in Section 5 below. However, given the current 24-month operating cycle, NMP2 would be required to remove and test approximately half of the SRVs every refueling outage in order to ensure that all valves are removed and tested in compliance with the ASME OM-2012 Code requirements for testing Class 1 pressure relief valves within a 5-year interval. With a 5-year interval, NMP2 would be required to remove all 18 SRVs over 2 refuel cycles (i.e., 4 years). However, consistent with the previously approved alternative MSS-VR-01, Revision 1 (ML15345A006), approval of extending the test interval to 6.5 years will reduce the number of SRVs removed during an individual outage, such that the full scope of 18 SRVs are replaced over 3 refuel cycles (i.e., 6 years, plus 6 months grace). This is consistent with the test interval and grace period described in ASME Code Case OMN-17, *Alternate Rules Class 1 Pressure/Safety Valves*, and continues to provide an acceptable level of quality and safety while restoring the operational and maintenance flexibility that was lost when the 24-month fuel cycle produced the unintended consequence of additional testing burden. Without Code relief, the incremental outage work due to the inclusion of the additional 2 - 3 SRVs per outage would be contrary to the principle of maintaining radiation dose As Low As Reasonably Achievable (ALARA). The removal and

replacement of the additional 2 - 3 SRVs per outage without Code relief results in an additional exposure of approximately 2 - 4 Rem each outage. Additionally, the grace period allows for flexibility in the scheduling of as-left and as-found set-pressure testing, which is based on a test-to-test frequency.

In accordance with 10 CFR 50.55a(z)(1), Exelon requests approval of an alternative to the 5-year test interval requirements of ASME OM Code, Appendix I, Section I-1320(a) for the SRVs at NMP2. Exelon requests that the test interval be increased from 5 years to 6.5 years. All other requirements of the applicable ASME OM Code would be met.

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

As an alternative to the Code required 5-year test interval per Appendix I, paragraph I-1320(a), Exelon proposes that the subject Class 1 pressure relief valves be tested at least once every three (3) refueling cycles (approximately 6 years/72 months) with a minimum of 20% of the valves tested within any 24-month interval. This 20% would consist of valves that have not been tested during the current 72-month interval, if they exist. The test interval for any individual valve would not exceed 72 months except that a 6-month grace period is allowed to coincide with refueling outages to accommodate extended shutdown periods and certification of the valve prior to installation.

As-found testing using steam and subsequent valve maintenance are currently performed at an off-site test facility. Subsequent to completion of as-found testing, each SRV in the removed complement is disassembled to perform inspections and a complete valve overhaul. Any SRV that failed the as-found set-pressure test is inspected to determine the cause of the test failure. Valve overhaul is performed to ensure that parts are free of defects resulting from time related degradation or service induced wear. All identified adverse conditions are corrected, the disc and seats are lapped, and the valve is reassembled. Each SRV is then recertified for service through inspection and testing consistent with ASME OM Code requirements, including set-pressure, seat tightness, stroke time and disc lift verifications, solenoid coil pick up/drop out, and air actuator integrity tests.

After recertification testing, the SRVs are stored at the test facility for future use. The storage area is inspected and maintained to the requirements of ANSI/ASME N45.2.2, *Packing, Handling, Shipping, Storage and Handling of Items for Nuclear Power Plants*, which will minimize the potential for any valve degradation.

The SRV as-found set-pressure test data in MSS-VR-01 Tables 1 and 2 demonstrates that the maintenance practices previously employed by NMP2 were effective. In the Spring 2010, prior to the RFO12 refueling outage, testing of the SRVs began at an offsite testing facility. These results are reflected in MSS-VR-01 Table 2.

Only one as-found set-pressure test failure (2002) has been experienced during the time period encompassed by the data in MSS-VR-01 Table 1. Note that testing performed on SRVs removed during these RFOs utilized nitrogen, with a correlated set-pressure. The data in MSS-VR-01 Table 1 also illustrates that SRVs that have exceeded 6 years

between tests still demonstrated acceptable as-found set-pressure test results. Additionally, there were a total of four (4) as-found set-pressure failures experienced in 2010 (MSS-VR-01 Table 2), which were determined not to be a result of set-pressure drift or a hardware degradation issue. The cause, which was determined to be common among the four (4) SRVs, was minor inaccuracies associated with the correlation used to establish the as-left set-pressure using nitrogen, and the as-found set-pressure using saturated steam.

During RFO13 (Spring of 2012), RFO14 (Spring of 2014), and RFO15 (Spring of 2016), there were no as-found set-pressure failures identified (MSS-VR-01 Table 3).

Based on the above valve performance history and SRV maintenance practices, there is continued assurance of valve operational readiness, as required by ASME OM-2012 Code, Mandatory Appendix I, paragraph I-1310(b). Therefore, this proposed alternative for testing the subject Class 1 pressure relief valves at least once every three (3) refueling cycles (approximately 6 years/72 months) with a minimum of 20% of the valves tested within any 24-month interval, including a 6-month grace period, will continue to provide assurance of the valves' operational readiness and provides an acceptable level of quality and safety pursuant to 10 CFR 50.55a(z)(1).

6. Duration of Proposed Alternative

This request, upon approval, will be applied to the NMP2 Fourth 10-Year Interval, which is scheduled to begin January 1, 2019, and conclude on December 31, 2028.

7. Precedence

1. This relief request was previously authorized for use during the third 10-year interval at NMP2, as documented in letter from NRC (T. L. Tate) to Exelon Generation Company, LLC (B. Hanson), "Nine Mile Point Nuclear Station, Unit 2 – Safety Evaluation by the Office of the Nuclear Reactor Regulation for Relief Request MSS-VR-01, Revision 1 (CAC No. MF5773)," dated January 29, 2016 (ML15345A006).

2. This relief request was previously authorized for use via MSS-VR-01 during the third 10-year interval for NMP2, as documented in letter from NRC (M. G. Kowal) to Nine Mile Point Nuclear Station, LLC (K. J. Polson), "Nine Mile Point Nuclear Station – Safety Evaluation of Relief Requests for the Unit No. 1 Fourth 10-Year and Unit No. 2 Third 10-Year Pump and Valve Inservice Testing Program (TAC Nos. MD9202 and MD9203)," dated December 29, 2008 (ML083500039).

3. Peach Bottom Atomic Power Station, Units 2 and 3–Safety Evaluation of Relief Request 01A-VRR-3 Regarding the Fourth 10-Year Interval of the Inservice Testing Program (TAC Nos. MF2509 and MF2510), dated April 30, 2014 (ML14094A051).

4. This alternative was authorized for use at NMP2 for the remainder of the second 10-year interval via letter from NRC (M. Banerjee) to Niagara Mohawk Power Corporation (J. H. Mueller), "Nine Mile Point Nuclear Station, Unit 2 – Alternative [GV-RR-07] to

American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code Regarding Inservice Testing of Main Steam Safety/Relief Valves (TAC No. MB0290)," dated April 17, 2001 (ML010880286).

8. References

1. ASME OM Code Case OMN-17, *Alternate Rules Class 1 Pressure/Safety Valves*

2. ANSI/ASME N45.2.2, *Packing, Handling, Shipping, Storage and Handling of Items for Nuclear Power Plants*

MSS-VR-01 Table 1
Main Steam SRVs Test Results for NMP2 Refueling Outages RFO7 (2000) through RFO11 (2008) [(1)]

SRV Tested	Serial No.	Set-Pressure (psig)	As-Found Set-Pressure Test Results (psig)	Correlated Set-Pressure (psig)	Correlated Max Set-Pressure +3% (psig)	Correlated Min Set-Pressure -3% (psig)	Accept/ Reject	Time from Last Test (Years)
Refueling Outage 11, April 2008								
2MSS*PSV126	160965	1195	1212	1215	1248	1181	Accept	6.77
2MSS*PSV127	160956	1205	1220	1224.9	1258.7	1190.9	Accept	6.79
2MSS*PSV128	160972	1165	1162	1184	1217	1151	Accept	8.42
2MSS*PSV131	160961	1175	1200	1194	1227	1161	Accept	6.77
2MSS*PSV132	160915	1185	1182	1205	1238	1171.4	Accept	6.77
2MSS*PSV135	160964	1195	1197	1215	1248	1181	Accept	10.1
Refueling Outage 10, April 2006								
2MSS*PSV120	160935	1185	1203	1205	1238	1171	Accept	6.44
2MSS*PSV121	160966	1195	1224	1215	1248	1181	Accept	6.40
2MSS*PSV122	160951	1185	1222	1204	1238	1171	Accept	8.11
2MSS*PSV125	160968	1185	1194	1205	1238	1171	Accept	8.12
2MSS*PSV129	160971	1205	1225	1225	1258	1191	Accept	6.42
2MSS*PSV133	160958	1165	1176	1184	1217	1151	Accept	4.78
Refueling Outage 9, April 2004								
2MSS*PSV123	160960	1175	1191	1195.2	1228	1162	Accept	7.19
2MSS*PSV124	160974	1175	1193	1195.2	1228	1162	Accept	4.43
2MSS*PSV130	160936	1195	1193	1215.5	1249	1181	Accept	7.21
2MSS*PSV134	160954	1205	1225	1225	1259	1191.6	Accept	7.92
2MSS*PSV136	160973	1175	1189	1195.2	1228	1162	Accept	6.11
2MSS*PSV137	160905	1205	1239	1225.7	1259.7	1191.6	Accept	6.12
Refueling Outage 8, March 2002								
2MSS*PSV121	160939	1195	1219	1214	1248	1180	Accept	4.09
2MSS*PSV126	160967	1195	1189	1214	1247	1180	Accept	5.88
2MSS*PSV127	160955	1205	1201	1224	1258	1190	Accept	5.90
2MSS*PSV128	160903	1165	1176	1184	1216	1151	Accept	4.08
2MSS*PSV129	160904	1205	1220	1224	1258	1190	Accept	4.08
2MSS*PSV132	160953	1185	1181	1204	1237	1171	Accept	5.16
2MSS*PSV134	160970	1205	1192	1224	1258	1190	Accept	5.18
2MSS*PSV135[(2)]	160976	1195	1170	1214	1247	1180	Reject	5.16
2MSS*PSV131[(2)]	160962	1175	1186	1194	1227	1161	Accept	5.17
2MSS*PSV133[(2)]	160959	1165	1169	1184	1216	1151	Accept	5.18

MSS-VR-01 Table 1 (Continued)
Main Steam SRVs Test Results for NMP2 Refueling
Outages RFO7 (2000) through RFO11 (2008) (1)

SRV Tested	Serial No.	Set-Pressure (psig)	As-Found Set-Pressure Test Results (psig)	Correlated Set-Pressure (psig)	Correlated Max Set-Pressure +3% (psig)	Correlated Min Set-Pressure -3% (psig)	Accept/ Reject	Time from Last Test (Years)
Refueling Outage 7, March 2000								
2MSS*PSV120	160915	1185	1219	1204	1238	1171	Accept	3.90
2MSS*PSV121	160965	1195	1231	1215	1248	1181	Accept	3.90
2MSS*PSV122	160950	1185	1222	1204	1238	1171	Accept	3.94
2MSS*PSV123	160963	1175	1208	1194	1227	1161	Accept	3.92
2MSS*PSV124	160906	1175	1189	1194	1227	1161	Accept	3.92
2MSS*PSV125	160952	1185	1220	1204	1238	1171	Accept	3.90
2MSS*PSV128	160958	1165	1193	1184	1217	1151	Accept	3.94
2MSS*PSV129	160956	1205	1214	1225	1258	1191	Accept	3.16
2MSS*PSV135	160975	1195	1244	1215	1248	1181	Accept	3.88
2MSS*PSV136	160961	1175	1221	1194	1227	1161	Accept	3.93
2MSS*PSV137	160954	1205	1222	1225	1259	1191	Accept	3.88

Notes: (1) Testing was performed at the NMP2 onsite test facility using nitrogen as the test medium.

(2) SRV 2MSS*PSV135 (SN 160976) failed the as-found set-pressure test (relieved early) during Refueling Outage 8. Two additional valves (2MSS*PSV131 (SN 160962) and 2MSS*PSV133 (SN 160959)) were tested per Code requirements, and both passed. The cause for this failure was determined to be set-pressure drift. Minor adjustments were made to restore the set-pressure to the acceptance range. No additional causes for the set-pressure drift were found during valve maintenance. The valve was refurbished and re-certified.

MSS-VR-01 Table 2
Main Steam SRVs Test Results for NMP2 Refueling Outage RFO12 (2010) (1)

SRV Tested	Serial No.	Set-Pressure (psig)	As-Found Set-Pressure Test Results (psig)	Max Set-Pressure +3%(psig)	Min Set-Pressure -3% (psig)	Accept/ Reject	Time from Last Test (Years)
Refueling Outage 12, April 2010(1)							
2MSS*PSV120	160953	1185	1229	1220.5	1149.4	Reject	7.38
2MSS*PSV121	160967	1195	1226	1230.9	1159	Accept	7.29
2MSS*PSV122	160952	1185	1217	1220.5	1149.4	Accept	9.76
2MSS*PSV123	160914	1175	1193	1210	1139.7	Accept	13.03
2MSS*PSV124	160906	1175	1212	1210	1139.7	Reject	9.70
2MSS*PSV125	160950	1185	1220	1220.5	1149.4	Accept	9.75
2MSS*PSV126	160939	1195	1229	1230.9	1159	Accept	7.44
2MSS*PSV127	160905	1205	1234	1241	1174.4	Accept	5.30
2MSS*PSV128	160903	1165	1196	1199.9	1130	Accept	7.46
2MSS*PSV129	160904	1205	1234	1241	1174.4	Accept	7.46
2MSS*PSV130	160976	1195	1226	1230.9	1159	Accept	7.4
2MSS*PSV131	160974	1175	1192	1210	1139.7	Accept	5.29
2MSS*PSV132	160969	1185	1197	1220.5	1149.4	Accept	3.31
2MSS*PSV133	160959	1165	1214	1199.9	1130	Reject	7.44
2MSS*PSV134	160955	1205	1211	1241	1174.4	Accept	7.39
2MSS*PSV135	160936	1195	1222	1230.9	1159	Accept	5.30
2MSS*PSV136	160962	1175	1204	1210	1139.7	Accept	7.46
2MSS*PSV137	160970	1205	1245	1241	1174.4	Reject	7.36

Notes: (1) All 18 SRVs were removed and replaced with pre-tested valves in Refueling Outage 12 (2010). The testing was performed at an offsite test facility using saturated steam as the test medium.

MSS-VR-01 Table 3
Main Steam SRVs Test Results for NMP2 Refueling
Outages RFO13 (2012), RFO14 (2014), and RFO15 (2016)

SRV Tested	**Serial No.**	**Set-Pressure (psig)**	**As- Found Set-pressure Test Results (psig)**	**Max Set-Pressure +3% (psig)**	**Min Set-Pressure -3% (psig)**	**Accept/ Reject**	**Time from Last Test (Years)**
Refueling Outage 15, March 2016							
2MSS*PSV120	160968	1185	1150	1220.55	1149.45	Accept	6
2MSS*PSV121	160966	1195	1222	1230.85	1159.15	Accept	6
2MSS*PSV124	160973	1175	1174	1210.25	1139.75	Accept	6
2MSS*PSV125	160951	1185	1175	1220.55	1149.45	Accept	6
2MSS*PSV127	160971	1205	1188	1241.15	1168.85	Accept	6
2MSS*PSV128	160958	1165	1167	1199.95	1130.05	Accept	6
Refueling Outage 14, March 2014							
2MSS*PSV122	160935	1185	1181	1220.55	1149.45	Accept	4
2MSS*PSV123	160963	1175	1170	1210.25	1139.75	Accept	4
2MSS*PSV126	160964	1195	1164	1230.85	1159.15	Accept	4
2MSS*PSV134	160954	1205	1201	1241.15	1168.85	Accept	4
2MSS*PSV135	160975	1195	1199	1230.85	1159.15	Accept	4
2MSS*PSV137	160957	1205	1190	1241.15	1168.85	Accept	4
Refueling Outage 13, April 2012							
2MSS*PSV129	160956	1205	1207	1241.15	1168.85	Accept	2
2MSS*PSV130	160965	1195	1196	1230.85	1159.15	Accept	2
2MSS*PSV131	160961	1175	1166	1210.25	1139.75	Accept	2
2MSS*PSV132	160915	1185	1182	1220.55	1149.45	Accept	2
2MSS*PSV133	160972	1165	1139	1199.95	1130.05	Accept	2
2MSS*PSV136	160960	1175	1168	1210.25	1139.75	Accept	2