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10 CFR 50.90
10 CFR 50.55a

NMP1L3348

August 20, 2020

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

Nine Mile Point Nuclear Station, Unit 1
Renewed Facility Operating License No. DPR-63
NRC Docket No. 50-220

Subject: License Amendment Request – Revise Technical Specifications to Apply TSTF-334, "Relaxed Surveillance Frequency for Excess Flow Check Valve Testing," Revision 2

In accordance with 10 CFR 50.90, "Application for amendment of license, construction permit, or early site permit," Exelon Generation Company, LLC (Exelon) requests an amendment to the Technical Specifications (TS), Appendix A, of Renewed Facility Operating License No. DPR-63 for Nine Mile Point Nuclear Station, Unit 1 (NMP1). The proposed amendment would revise Surveillance Requirement (SR) 4.3.4c in NMP1 TS 3.3.4, "Primary Containment Isolation Valves," to state that a representative sample (i.e., approximately 20 percent) of instrument-line flow check valves will be tested in accordance with the Surveillance Frequency Control Program (SFCP) such that each instrument-line flow check valve will be tested at least once every 10 years (nominal). This proposed change is consistent with NRC-approved Technical Specification Task Force (TSTF) 334, "Relaxed Surveillance Frequency for Excess Flow Check Valve Testing," Revision 2. Currently, SR 4.3.4c requires each instrument-line flow check valve to be tested. The NMP1 SFCP requires testing each valve on a 24-month frequency for SR 4.3.4c.

Additionally, in accordance with 10 CFR 50.55a, "Codes and standards," paragraph (z)(1), Exelon requests NRC approval of the attached relief request (RR) GV-RR-09 associated with the fifth 10-year interval Inservice Testing (IST) Program for NMP1. The IST RR is being submitted to request relief from the 10 CFR 50.55a requirements for testing the instrument-line flow check valves in accordance with the American Society of Mechanical Engineers Operation and Maintenance (ASME OM) Code requirements. The IST RR is being submitted to modify the IST Program to be consistent with the proposed TS change.

The sampling methodology was approved by the NRC in a Safety Evaluation (SE) dated March 14, 2000, associated with the approval of the Boiling Water Reactors (BWR) Owners Group Licensing Topical Report NEDO 32977-A Excess Flow Check Valve Testing Relaxation." This SE also addressed the approval of Technical Specification Task Force (TSTF) Improved Standard Technical Specifications Change Traveler TSTF-334, "Relaxed Surveillance Frequency for Excess Flow Check Valve Testing," Revision 2, dated October 31, 2000. The license amendment request (LAR) and the associated RR are consistent with the referenced SE, the associated BWR Owners Group Topical Report, and TSTF-334, Revision 2.

The reduced testing associated with this proposed change will result in an increase in the availability of the associated instrumentation during outages and will result in dose savings without significantly impacting the health and safety of the public while continuing to provide an acceptable level of quality and safety.

Attachment 1 provides the Evaluation of Proposed Changes. Attachment 2 is the failure rate analysis prepared by GE Hitachi Nuclear Energy for NMP1. Attachment 3 provides the Proposed TS Marked-Up Page. Attachment 4 provides the Bases Marked-Up Page. Attachment 5 provides Relief Request GV-RR-09.

The proposed changes have been reviewed by the NMP Plant Operations Review Committee in accordance with the requirements of the Exelon Quality Assurance Program.

Exelon requests approval of the proposed amendment by February 28, 2021. Once approved, the amendment shall be implemented no later than startup from the NMP1 Spring 2021 refueling outage.

There are no regulatory commitments contained in this request.

Exelon has concluded that the proposed change presents no significant hazards consideration under the standards set forth in 10 CFR 50.92.

In accordance with 10 CFR 50.91, "Notice for public comment; State consultation," paragraph (b), Exelon is transmitting a copy of this application and its attachments to the designated State Official.

Should you have any questions concerning this submittal, please contact Ron Reynolds at (610) 765-5247.

I declare under penalty of perjury that the foregoing is true and correct. Executed on the 20th day of August 2020.

Respectfully,

David T. Gudger

David T. Gudger
Senior Manager - Licensing
Exelon Generation Company, LLC

Attachments:

1) Evaluation of Proposed Change
2) GEH 006N1767 R0 - Nine Mile Point Unit 1 (NMP1) Excess Flow Check Valve (EFCV) Failure Rate Analysis
3) Proposed Markup of Technical Specification Page
4) Proposed Markup of Technical Specification Bases Page
5) 10CFR 50.55a Relief Request GV-RR-09 Related to Excess Flow Check Valve Testing Frequency

cc:	USNRC Region I, Regional Administrator	w/attachments
	USNRC Senior Resident Inspector, NMP	w/attachments
	USNRC Project Manager, NMP	w/attachments
	A. L. Peterson, NYSERDA	w/attachments

ATTACHMENT 1

License Amendment Request

Nine Mile Point Nuclear Station, Unit 1

Docket No. 50-220

EVALUATION OF PROPOSED CHANGE

CONTENTS

SUBJECT: Revise Technical Specifications to Adopt TSTF-334, Revision 2, "Relaxed Surveillance Frequency for Excess Flow Check Valve Testing"

1.0 SUMMARY DESCRIPTION

Pursuant to 10 CFR 50.90, “Application for amendment of license, construction permit, or early site permit,” Exelon Generation Company, LLC (Exelon) requests an amendment to the Technical Specifications (TS), Appendix A, of Renewed Facility Operating License No. DPR-63 for Nine Mile Point Nuclear Station, Unit 1 (NMP1).

NMP1 Technical Specifications (TS) 3.3.4, "Primary Containment Isolation Valves," currently requires performance of Surveillance Requirement (SR) 4.3.4c on each instrument-line flow check valve in accordance with the Surveillance Frequency Control Program (SFCP). The proposed changes would revise the number of instrument-line flow check valves tested by TS SR 4.3.4c from "each" to "a representative sample." The representative sample will test approximately 20 percent of the instrument-line flow check valves each operating cycle such that each instrument-line flow check valve will be tested at least once every 10 years (nominal).

The reduced testing associated with the proposed change will result in an increase in the availability of the associated instrumentation and dose savings during outages. The proposed change will not significantly impact the health and safety of the public and will continue to provide an acceptable level of quality and safety.

2.0 DETAILED DESCRIPTION

TS SR 4.3.4c currently requires that each instrument-line flow check valve be tested for operability in accordance with the SFCP. The NMP1 SFCP currently requires each instrument-line flow check valve to be tested on a 24-month frequency. This SR is currently implemented for the instrument-line flow check valves performed during the Reactor Pressure Vessel System Leakage Test (also known as hydrostatic testing) each refuel outage.

The proposed change revises SR 4.3.4c to verify that a representative sample (i.e., approximately 20 percent) of instrument-line flow check valves be tested for operability, in accordance with the SFCP, such that all instrument-line flow check valves will be tested at least once every 10 years (nominal). The proposed change is similar to existing performance-based testing programs, such as Inservice Testing (snubbers) and Option B to 10 CFR 50, Appendix J.

The proposed change implements Technical Specification Task Force (TSTF) Traveler TSTF-334, Revision 2, "Relaxed Surveillance Frequency for Excess Flow Check Valve Testing,' dated October 31, 2000 (Reference 1). The proposed change described in TSTF-334, Revision 2; meets the intent the Standard Technical Specifications (STS); however, NMP1 has Custom TS. An administrative variation to TSTF-334, Revision 2 is discussed below.

<u>Variation</u>

The NMP1 TS are Custom Technical Specifications (CTS) and are a different format with different numbering and titles than the NRC Improved Standard Technical Specifications in NUREG-1433 (STS) on which TSTF-334, Revision 2, was based. As an example, the NMP1 CTS does not use the term "Excess Flow Check Valve," instead it uses "Instrument-line flow check valve. This variation is administrative and does not affect the applicability of TSTF-334, Revision 2, to the NMP1 TS.

3.0 TECHNICAL EVALUATION

SR 4.3.4c requires that each instrument-line flow check valve be tested for operability every 24 months in accordance with the SFCP. This surveillance is performed during a plant outage due to required test conditions and the potential for an unplanned transient if the surveillance was performed with the reactor at power. Performance of the SR for most of these valves requires the reactor to be pressurized to near normal operating pressure. The SR is implemented by isolating the affected instrument and opening a drain valve downstream of the instrument-line flow check valve while the process side is exposed to reactor pressure.

The NMP1 Updated Final Safety Analysis Report (UFSAR) Section VI.D., "Containment Isolation System," describes that instrumentation lines penetrating containment from the reactor coolant pressure boundary (RCPB) are provided with valving outside the containment to facilitate testing and maintenance. Additionally, an instrument-line flow check valve is located outside primary containment. Should an instrument line that forms part of the RCPB develop a leak of sufficient flow outside containment, the instrument-line flow check valve will close automatically.

The Boiling Water Reactors (BWR) Owners Group issued a topical report that provides a basis for this request. This report (NEDO 32977-A, Reference 2) provides justification for the relaxation in the Surveillance Requirement (SR) to test a representative sample of instrument-line flow check valves at the current 24-month interval required by the SFCP. The report demonstrates the high degree of instrument-line flow check valve reliability and the low consequences of an instrument-line flow check valve failure.

The manufacturers of the NMP1 instrument-line flow check valves are Dragon Valves, Inc., and Marrotta, respectively. Both vendors are used by many other BWR utilities and are well represented in the NEDO Report. Attachment 2 contains a table of NMP1 specific data, similar to the table in the NEDO Report. This data shows that the NMP1 valves are similar in design, and use, to the valves used by the other member utilities. Furthermore, a failure rate analysis of the NMP1 valves was done over ten operating cycles. These results show that the failure rate study done for the NEDO Report bounds the NMP1 failure rate. Attachment 2 details the results of the study. The generic radiological consequences evaluation performed by GE-Hitachi Nuclear Energy in Attachment 2 to the NEDO Report bounds NMP1. It is thus reasonable to conclude, as the NEDO Report states, that similar results would be expected at NMP1.

Several amendments have been submitted and approved for other BWRs. The format and content of these proposed TS and Bases changes are consistent with the BWR Owners Group Topical Report NEDO 32977-A and the approved generic change TSTF-334, Revision 2 (Reference 1).

3.1 Instrument-Line Flow Check Valve Failure Rate and Release Frequency

The BWR Owners Group Topical Report NEDO 32977-A, dated June 2000 (Reference 2), provides detailed information regarding instrument-line flow check valve surveillance testing data at 12 BWR plants. NMP1 is listed as a participating utility in NEDO 32977-A; however, NMP1 data was not included as one of the 12 BWR plants listed in Table 4-1, "EFCV Failure Rates," of NEDO 32977-A.

The NMP1 instrument-line flow check valve failure rate analysis and release frequency were evaluated in Attachment 2.

The data provided in Table 4-1 of Attachment 2, "Summary of NEDO-32977-A EFCV Failure Rate Analysis Including NMP1, LSCS and Hatch" is similar to the data provided in Table 4-1 of the NEDO 32977-A Report, with the exception that Table 4-1 of Attachment 2 also includes data for NMP1. The original data from the NEDO 32977-A Report was assembled prior to the year 2000. The NMP1 data included in Table 4-1 of Attachment 2 includes more recent data, including failure rate history over the last ten operating cycles.

As stated in Attachment 2, the NMP1 instrument-line flow check valves are similar in design and application of those included in the BWR Owners Group Topical Report. The NMP1 data was found to be consistent in both the time sampled and instrument-line flow check valve reliability when compared to the topical report data. A failure rate analysis of the NMP1 instrument-line flow check valves was done over ten operating cycles, and these results show that the NMP1 best estimate and upper bound failure rates are below the highest failure rates presented in NEDO-32977-A. Furthermore, the NMP1 failure rates consistent with the industry data range and trends.

3.2 Failure Feedback Mechanism and Corrective Action Program

The NRC safety evaluation associated with the BWR Owners Group Topical Report NEDO 32977-A (Reference 3) requires that each plant's corrective action program evaluate equipment failures and establish appropriate corrective actions. The NMP1 Maintenance Rule Program per NEI 18-10 currently tracks failures and establishes corrective actions based on risk assessments as appropriate.

3.3 Radiological Dose Assessment

Radiological consequences for an instrument line break outside containment have been calculated for NMP1 using the Alternative Source Term methodology. The calculation analyzes a release of iodine associated with reactor coolant that is directly released to the environment. The coolant mass released is based on a break of a 1-inch instrument line. The mass energy release assumes operator action resulting in reactor depressurization over a 5.4-hour cooldown period. The radiological consequences for the instrument line break were calculated to be within the guidelines of 10 CFR 50.67 for the control room and a small fraction of the 10 CFR 50.67 guidelines for Exclusion Area Boundary and Low Population Zone offsite receptors.

3.4 Conformance of the Proposed NMP1 CTS to Generic TSTF Guidance

The term "representative sample," as proposed by the BWROG topical report and TSTF-334, Revision 2, is not specifically defined in the TS. The associated Bases for SR 4.3.4c will be revised pursuant to the Bases Control Program and TSTF-334, Revision 2. The Bases will include a statement that "the instrument-line flow check valves in the sample are representative of the various plant configurations, models, sizes and operating environments."

4.0 REGULATORY EVALUATION

4.1 Applicable Regulatory Requirements/Criteria

10 CFR 50.65, "Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," provides that licensees shall monitor the performance or condition of structures, systems, or components, against licensee-established goals, in a manner sufficient to provide reasonable assurance that these structures, systems, and components are capable of fulfilling their intended functions. The NMP1 Corrective Action Program, in combination with the implementation of 10 CFR 50.65 for the instrument-line flow check valves, will address actions associated with failures of instrument-line flow check valves during testing.

The NRC safety evaluation (SE), dated March 14, 2000, associated with the approval of the BWR Owners Group Topical Report NEDO 32977-A, "Excess Flow Check Valve Testing Relaxation," (Reference 3) provides the NRC's position on this topical report and industry implementation. The NRC agreed that the test interval could be extended up to a maximum of 10 years. In conjunction with this finding, the NRC noted that each licensee that adopts the relaxed test interval program for EFCVs must have a failure feedback mechanism and corrective action program to ensure that EFCV performance and reliability continues to be bounded by the topical report results; each licensee is required to perform a plant-specific radiological dose assessment, EFCV failure analysis, and release frequency analysis to confirm that they are bounded by the generic analyses of the topical report.

BWROG topical report NEDO-32977-A, "Excess Flow Check Valve Testing Relaxation," dated June 2000, provides (1) an estimate of steam release frequency (into the reactor building) due to a break in an instrument line concurrent with an EFCV failure to close, and (2) an assessment of the radiological consequences of such a release.

The proposed change revises SR 4.3.4c to verify that a representative sample (approximately 20 percent) of instrument-line flow check valves actuate to the isolation position during a simulated instrument line break signal, in accordance with the Surveillance Frequency Control Program, such that each instrument-line flow check valve will be tested at least once every 10 years (nominal). The proposed change implements Technical Specification Task Force (TSTF) Traveler TSTF-334, Revision 2, "Relaxed Surveillance Frequency for Excess Flow Check Valve Testing," dated October 31, 2000, with one administrative variation: The NMP1 CTS does not use the term "Excess Flow Check Valve," instead it uses "Instrument-line flow check valve.

4.2 Precedents

The following precedents are provided below:

- Letter from B. Vaidya (NRC) to B. Hanson (Exelon), "LaSalle County Station, Units 1 and 2 - Issuance of Amendments to Renewed Facility Operating Licenses RE: Revision of Technical Specifications to Adopt TSTF-334, 'Relaxed Surveillance Frequency for Excess Flow Check Valve Testing Related to Excess Flow Check Valve Testing," dated February 14, 2019. (ADAMS Accession No. ML19025A288)

- Letter from L. N. Olshan (U.S. Nuclear Regulatory Commission) to H. L. Sumner (Southern Nuclear Operating Company, Inc.), "Edwin I. Hatch Nuclear Plant, Units 1 and 2 Re: Issuance of Amendments (TAC Nos. MB2976 and MB2977)," dated April 11, 2002 (ADAMS Accession No. ML020720594)

- Letter from R. G. Schaaf (U.S. Nuclear Regulatory Commission) to R. G. Byram (PPL Susquehanna, LLC), "Susquehanna Steam Electric Station, Units 1 and 2 — Issuance of Amendment [Nos. 193 and 168] Re: Relaxation of Excess Flow Check Valve Surveillance Requirements (TAC Nos. MB0425 and MB0427)," dated April 11, 2001 (ADAMS Accession No. ML010960024)

- Letter from P. S. Tam (U.S. Nuclear Regulatory Commission) to J. H. Mueller (Niagara Mohawk Power Corporation), "Nine Mile Point Nuclear Station Unit No. 2 — Issuance of Amendment [No. 96] Re: Excess Flow Check Valves Surveillance Testing (TAC No. MB0301)," dated July 12, 2001 (ADAMS Accession No. ML0101440365)

- Letter from G. S. Shukla (U.S. Nuclear Regulatory Commission) to D. L. Wilson (Nebraska Public Power District), "Cooper Nuclear Station — Issuance of Amendment to Revise the Technical Specifications Surveillance Test Requirement SR 3.6.1.3.8. for Excess Flow Check Valves (EFCVs) (TAC No. MB1820)," dated October 26, 2001 (ADAMS Accession No. ML01240440)

- Letter from R. Ennis (U.S. Nuclear Regulatory Commission) to H. Keiser (PSEG Nuclear), " Hope Creek Generating Station - Issuance of Amendment RE: Excess Flow Check Valve Testing Requirements (TAC NO. MB1723)," dated August 28, 2001 (ADAMS Accession No. ML012130156)

4.3 No Significant Hazards Consideration

Exelon Generation Company, LLC (Exelon) requests adoption of TSTF-334, Revision 2, "Relaxed Surveillance Frequency for Excess Flow Check Valve Testing,' dated October 31, 2000, which is an approved change to the Improved Standard Technical Specifications (ITS), into the Nine Mile Point Unit 1 (NMP1) Custom Technical Specifications (CTS). NMP1 CTS refers to "excess flow check valves" as "instrument-line flow check valves." The proposed change revises SR 4.3.4c to verify that a representative sample (i.e., approximately 20 percent) of instrument-line flow check valves be tested for operability, in accordance with the SFCP, such that all instrument-line flow check valves will be tested at least once every 10 years (nominal).

Exelon has evaluated whether or not a significant hazards consideration is involved with the proposed amendment by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

1. **Does the proposed amendment involve a significant increase in the probability or consequences of an accident previously evaluated?**

   Response: No.

   The instrument-line flow check valves at Nine Mile Point Station, Unit 1 (NMP1), are designed so that they will not close accidently during normal operations, will close if a

rupture of the instrument line is indicated downstream of the valve, can be reopened when appropriate. This proposed change relaxes the number of instrument-line flow check valves tested for Technical Specifications (TS) Surveillance Requirement (SR) 4.3.4c from "each" to a "representative sample" in accordance with the Surveillance Frequency Control Program (SFCP). There are no physical plant modifications associated with this change. Industry and NMP1 operating experience demonstrate a high reliability of these valves. Neither instrument-line flow check valves nor their failures are capable of initiating previously evaluated accidents; therefore, there can be no increase in the probability of occurrence of an accident regarding this proposed change.

The radiological consequences for the instrument line break were calculated to be within the guidelines of 10 CFR 50.67 for the control room and a small fraction of the 10 CFR 50.67 guidelines for Exclusion Area Boundary and Low Population Zone offsite receptors.

Therefore, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed amendment create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No.

This proposed change allows a reduced number of instrument-line flow check valves to be tested in accordance with the SFCP. The proposed change would revise SR 4.3.4c to verify that "a representative sample" (i.e., approximately 20 percent) of instrument-line flow check valves are tested, in accordance with the SFCP, such that each instrument-line flow check valve will be tested at least once every 10 years (nominal). No other changes in the requirements are being proposed. Industry and NMP1-specific operating experience demonstrate the high degree of reliability of the instrument-line flow check valves and the low consequences of an instrument-line flow check valve failure. The potential failure of an instrument-line flow check valve to isolate by the proposed reduction in test frequency is bounded by the previous evaluation of an instrument line rupture. This change will not alter the operation or process variables, structures, systems, or components as described in the safety analysis. Thus, a new or different kind of accident will not be created from implementation of the proposed change.

Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the proposed amendment involve a significant reduction in a margin of safety?

Response: No.

Radiological consequences for an instrument line break outside containment have been calculated for NMP1 using the Alternative Source Term methodology. The calculation analyzes a release of iodine associated with reactor coolant that is directly released to the environment. The coolant mass released is based on a break of a 1-inch instrument

line. The mass energy release assumes operator action resulting in reactor depressurization over a 5.4-hour cooldown period. The radiological consequences for the instrument line break were calculated to be within the guidelines of 10 CFR 50.67 for the control room and a small fraction of the 10 CFR 50.67 guidelines for Exclusion Area Boundary and Low Population Zone offsite receptors. Thus, the failure of an instrument-line flow check valve, though not expected as a result of the proposed change, does not affect the dose consequences of an instrument line break.

Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

Based on the above, Exelon concludes that the proposed amendment does not involve a significant hazards consideration under the standards set forth in 10 CFR 50.92(c) and, accordingly, a finding of "no significant hazards consideration" is justified.

4.4 Conclusion

In conclusion, based on the considerations discussed above, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operations in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

5.0 ENVIRONMENTAL CONSIDERATION

A review has determined that the proposed amendment would change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or would change an inspection or surveillance requirement. However, the proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

6.0 REFERENCES

1. Technical Specification Task Force TSTF-334, Revision 2, "Relaxed Surveillance Frequency for Excess Flow Check Valve Testing," dated October 31, 2000 (ADAMS Accession No. ML003751245)

2. BWROG Licensing Topical Report, NEDO-32977-A, "Excess Flow Check Valve Testing Relaxation," June 2000 (ADAMS Accession No. ML003729011)

3. Letter from S. A. Richards (U.S. Nuclear Regulatory Commission) to W. G. Warren (BWR Owners Group Chairman), "Safety Evaluation of General Electric Nuclear Energy Topical Report B21-00658-01, 'Excess Flow Check Valve Testing Relaxation,`--TAC Nos. MA7884 and MB4809)," dated March 14, 2000 (ADAMS Accession No. ML003691722)

ATTACHMENT 2

License Amendment Request

Nine Mile Point Nuclear Station, Unit 1
Docket No. 50-220

Revise Technical Specifications to Adopt TSTF-334, Revision 2, "Relaxed Surveillance Frequency for Excess Flow Check Valve Testing"

GEH 006N1767 R0 - Nine Mile Point Unit 1 (NMP1)
Excess Flow Check Valve (EFCV) Failure Rate Analysis
Total 15 pages

HITACHI

GE Hitachi Nuclear Energy

006N1767

Revision 0

July 2020

Non-Proprietary Information

Nine Mile Point Unit 1 (NMP1) Excess Flow Check Valve (EFCV) Failure Rate Analysis

DISCLAIMER OF RESPONSIBILITY

REVISION SUMMARY

Rev #	Section Modified	Revision Summary
A	N/A	Initial Release
0	Section 3.0	Updated the failure event descriptions and used NMP1's 2-year cycle for the calculation of public risk with the current testing basis

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LIST OF TABLES

1.0 INTRODUCTION

Nine Mile Point Unit 1 (NMP1) is pursuing a license amendment request and a relief request to relax testing frequency for Excess Flow Check Valves (EFCVs), utilizing NEDO-32977-A Reference 1), which has been approved by the Nuclear Regulatory Commission (NRC). Reference 1 was commissioned by the Boiling Water Reactor Owners' Group (BWROG) with the final report being issued in June of 2000. Although Nine Mile Point Unit 1 and Unit 2 participated in the development of Licensing Topical Report (LTR) NEDO-32977-A, NMP1 EFCV data was not included in the analysis. However, adequate NMP1 EFCV data have since been compiled and, as detailed in this report, the design data and failure rates are comparable to the other utilities documented in Reference 1.

1.1 Purpose

The purpose of the NMP1 EFCV failure rate analysis is to provide justification for relaxing EFCV testing frequency. The relaxed testing frequency allows for a representative sample of EFCVs to be tested each cycle (24 months) such that each EFCV will be tested at least once every 10 years. This means that approximately 20 percent of the EFCVs will be tested every cycle.

1.2 Basis for Proposed Changes

The justification for the proposed changes is found in NEDO-32977-A (Reference 1), which was developed by General Electric for the EFCV committee of the BWROG. Reference 1 contains an upper limit failure rate that was compiled with data from 12 different stations (see NEDO-32977-A Section 4.2 and Table 4-1). NMP1 data were not included in the original composite upper limit failure rate. Though NMP1 data were not included in Reference 1, it was determined that the conclusions of Reference 1 are applicable to the NMP1 EFCV system, as discussed in the remainder of this report.

1.3 Scope

In support of a license amendment request and relief request to relax testing frequency for NMP1's EFCVs, GEH has developed this report for EFCV failure rate analysis using NMP1's plant-specific data. This report documents the results as both best estimate and upper limit failure rates. The method follows NEDO-32977-A (Reference 1), which has been approved by the NRC. The report format is similar to Hatch TSTF-334 LAR Attachment 2 to Enclosure 1, "Hatch Excess Flow Check Valve Failure Rate Results" (Reference 2). This report also incorporates the Hatch Request for Additional Information (RAI) (Reference 3) responses into the analysis, as applicable.

Although Hatch was not one of the 12 BWR plants referenced in NEDO-32977-A (similar to NMP1), the Hatch data were found to be consistent in both the time sampled and EFCV reliability when compared to the Reference 1 data. In addition, the Hatch plant-specific EFCV failure and release rates are comparable to industry data and consistent with the NRC Safety Evaluation (SE) (Reference 4) conclusions for NEDO-32977-A.

2.0 NMP1 PLANT-SPECIFIC EFCV DATA

The NMP1 plant-specific EFCV data, similar to the data presented in the tables of NEDO-32977-A (Reference 1), are provided in Table 2-1.

The values of the parameters presented in Table 2-1 were compiled based on information from Marotta and Dragon Valves EFCV drawings, as well as input from NMP1 Radiation Protection (RP), and Outage Planning. These data show that the NMP1 EFCVs are similar in design, and use, to the valves used by the other member utilities documented in Reference 1.

Table 2-1: NMP1 Plant-Specific EFCV Data

Parameter	Value
Design pressure (psig)	1250
Hydrostatic pressure (psig)	3000
Make and model	Original Make: Marotta, Model: FVL1 Replacement Make: Dragon, Part Number: 14437N-11SE
Nominal line size (inlet/outlet, inches)	¾” (inlet/outlet)
Main Valve Body Orifice (inches)	0.43
Minimum flow for closure (gpm)	25
Test method	Testing is performed offline (refuel outages) during the vessel pressure test.
Testing performed	Flow reduction Audible sound
Testing on critical path	Yes, testing during vessel pressure test, impact estimated at 3 to 4 hours.
Person-Rem exposure during testing (mrem)	180
Person-hours for testing during a cycle	172
Types of failures	Corrosion product fouling
PM program	No
Failed EFCVs replaced or repaired?	Replaced

3.0 FAILURE RATE ANALYSIS

This failure rate study of the NMP1 EFCVs was conducted by reviewing the data over the data collection period between 2000 and 2020. A total of 52 EFCVs were tested each cycle with only 7 valve failures attributed to the valve actually failing to check flow. The details regarding each of the EFCV failures at NMP1 are provided in Table 3-1, and the operating failure rates based on this failure event data are calculated in the remainder of this section.

Table 3-1: Summary of NMP1 EFCV Failure Events

EFCV	Valve Type	Failure Year	Event Report Number	Failure Type (see note 1)
CKV-36-165	Marotta	2003	AR 2108715	CKV-36-165 failed to check at test reactor pressure of 850 psig and at 905 psig. Valve checked with reactor pressure at 1005 psig. Subsequent closure repeated at lower pressures. Valve not replaced. Counted as an EFCV failure event in the failure rate calculation.
CKV-32-215	Marotta	2005	AR 2019669	CKV-32-215 failed to check. A new check valve was installed, and post-maintenance testing of the check valve was performed satisfactorily. Replacement Valve Type: Marotta. Counted as an EFCV failure event in the failure rate calculation.
CKV-32-138	Marotta	2009	IR 2480732	CKV-32-138 failed to check flow. A new check valve was installed, and post-maintenance testing of the check valve was performed satisfactorily. Replacement Valve Type: Dragon. Counted as an EFCV failure event in the failure rate calculation.
CKV-36-165	Marotta	2011	IR 1974088	CKV-36-165 checked flow but would not reset. This failure event is not counted for the failure rate calculation for EFCV failures to check flow (see note 1). Replacement Valve Type: Dragon.

Table 3-1: Summary of NMP1 EFCV Failure Events

EFCV	Valve Type	Failure Year	Event Report Number	Failure Type (see note 1)
CKV-32-106	Marotta	2013	IR 1996576	CKV-32-106 cannot purge the line of air. The vent line was successfully flushed by FIN using residual reactor pressure via the EFCV bypass line. Valve not replaced. Counted as an EFCV failure event in the failure rate calculation.
CKV-32-138	Dragon	2013	IR 1996504 IR 2479713	CKV-32-138 failed to check flow. A new check valve was installed, and post-maintenance testing of the check valve was performed satisfactorily. Likely not an actual failure to check based on subsequent determination in 2015 that the Dragon valves required revised acceptance criteria (see note 2). Replacement Valve Type: Dragon. Counted as an EFCV failure event in the failure rate calculation.
CKV-32-138	Dragon	2015	IR 2479713 IR 2480732	CKV-32-138 failed to check flow. A new check valve was installed, and post-maintenance testing of the check valve was performed satisfactorily. Likely not an actual failure to check based on subsequent determination in 2015 that the Dragon valves required revised acceptance criteria (see note 2). Replacement Valve Type: Dragon. Counted as an EFCV failure event in the failure rate calculation.
CKV-36-165	Dragon	2015	IR 2479655 IR 2480732	CKV-36-165 initially declared a failure to check based on procedure requiring both a hammer sound and substantial reduction in flow noise. Declared a satisfactory test based on revised acceptance criteria (see notes 1 and 3). Valve not replaced. This failure event is not counted for the failure rate calculation for EFCV failures to check flow.

Table 3-1: Summary of NMP1 EFCV Failure Events

EFCV	Valve Type	Failure Year	Event Report Number	Failure Type (see note 1)
CKV-36-165	Dragon	2019	IR 4236919	CKV-36-165 would not check the flow during excess flow test. A replacement of this valve has been performed to correct the conditions (See note 4 for apparent cause). Replacement Valve Type: Dragon. Counted as an EFCV failure event in the failure rate calculation.

Notes:

1. Total EFCV failure events used in the failure rate calculation for EFCV failures to check flow:
   - 4 of the 5 Marotta EFCV considered failed (excluded the 2011 CKV-36-165 failure to reset)
   - 3 of the 4 Dragon considered failed (excluded the 2015 CKV-36-165 based on passing the revised criteria)

   The assumed 3 out the 4 Dragon failures is a conservative assumption. The failures of the Dragon valves for the 2013 and 2015 of the CKV-32-138 and the 2015 CKV-36-165 were not actual failures. The assessment is that the observed failure to check flow were the result of the misapplication of a criterion developed for the Marotta valves that was not revised to reflect the flow characteristics of the Dragon valves and these observed failures are not considered repeat failures, see notes 2 and 3 for more details.
2. In 2015, IR 2479713 concluded, "The verification of excess flow check valve 'function to check' was not recognized as different between the different model EFCVs (i.e. the newer Model #14437N-11SE Dragon valve have slightly different flow reduction noise than the older style). The confusion is explained in the evaluation for a subsequent misdiagnosed EFV failure in IR#2479655. Therefore, two supposed failures occurred before the test methodology differences were realized (IR#02479713 / CKV-32-138); neither was a real failure."
3. In 2015, IR 2479655 was written to document a failure of CKV-36-165. The acceptance criteria within the NMP1 EFCV testing procedure stated, "Confirm Flow Check Valve (number) CLOSED as indicated by a solid hammer OR hammering sound AND substantial reduction in audible flow noise." The technicians stated that a reduction of flow noise was heard and believed that that valve had closed. It was concluded that due to different flow characteristics with the Dragon valve that a hammering sound may not occur and that a reduction in flow noise is acceptable to confirm that the valve had closed. The NMP1 EFCV testing procedure was revised to change the acceptance criteria to "Confirm Flow Check Valve (number) CLOSED as indicated by at least one of the following: A solid hammer, Hammering sound, Substantial reduction in audible flow noise."

4. The 2019 IR 4236919 documents the failure to check flow based on the application of the revised criteria described in note 3. Engineering has concluded the most probable cause for the failure to check in 2019 of this Dragon valve is reactor corrosion products causing valve guide binding. This determination is qualitative based on review of previously documented apparent cause IR 1996504 written for the first Dragon valve failure observed at NMP1 in 2013 for the CKV-32-138 valve. Industry operating experience also identifies corrosion product causing EFCV failure to check as a frequently observed cause for EFCV failures.

The calculation of the best estimate failure rate was performed by dividing the number of failures by the total valve operating time. The calculation of the upper limit failure rate was performed using the formula listed in Section 4.2 of NEDO-32977-A (Reference 1), as follows:

$$\lambda_U = \frac{1}{2T}\chi^2_{\alpha:2r+2}$$

where:

λ_U = the upper limit failure rate per hour

T = the operating time in hours

r = the number of failures

$\chi^2_{\alpha:2r+2}$ = the value taken from the chi-square distribution tables which corresponds to 2r+2 degrees of freedom and α = 0.05 (1 - α = 0.95, the specified confidence level).

For NMP1, the operating time used was 20 years, since failure rate history was obtained from 2000 through 2020. NMP1 had a total of 7 EFCV failures to check flow. The Chi-square value for 16 degrees of freedom and a confidence level of 0.95 is 26.3. From these values, and using the formula listed above, the upper limit failure value for NMP1 is 1.44E-6 per hour. A summary of the NMP1 EFCV failure rate analysis is provided in Table 3-2.

Table 3-2: Summary of NMP1 EFCV Failure Rate Analysis

Unit	Total Valve Operating Time (hours)	Failures	Best Estimate Failure Rate (per hour)	Chi-Square Value (χ^2) 95% Confidence	Upper Limit of Expected Failures (per hour)
1	9.11E+06	7	7.68E-7	26.3	1.44E-6

The risk impact on the public health and safety from EFCVs can be evaluated as the product of a release frequency (due to a break in an instrument line concurrent with an EFCV failure to close) and the consequence of the release (Section 3.1 of NEDO-32977-A). The release frequency can be calculated based on the instrument line break frequency (Section 4.3 of NEDO-32977-A) and EFCV failure to close probability.

Table 3-3 shows the release frequency from a single instrument line, assuming different test intervals for the EFCV. The release frequency was calculated using the following equations:

$$RF = I \times \bar{A}$$

$$\bar{A} = \lambda \frac{\theta}{2}$$

where:

RF = release frequency per year

I = instrument line break frequency per year (Section 4.3 of NEDO-32977-A)

$\bar{A}$ = EFCV unavailability (failure to close probability)

λ = EFCV failure rate per hour (Table 3-2)

θ = EFCV surveillance test interval in hours

Table 3-3: Release Frequency from a Single Instrument Line (based on 5.34E-6/year instrument line break frequency, and 1.44E-6/hour EFCV failure rate)

EFCV Test Interval (years)	EFCV Test Interval (hours)	EFCV Unavailability	Release Frequency (per year)
1.5	13140	9.48E-03	5.06E-08
2	17520	1.26E-02	6.75E-08
6	52560	3.79E-02	2.03E-07
10	87600	6.32E-02	3.38E-07

Based on the release frequency shown in Table 3-3 for one instrument line, and assuming 52 instrument lines with testable EFCVs in a plant (consistent with NEDO-32977-A), the release frequency from any broken instrument line is:

- For 2-year surveillance test interval 52 * 6.75E-8 = 3.51E-6 events/year
- For 10-year surveillance test interval 52 * 3.38E-7 = 1.76E-5 events/year

The risk to the public can be shown by combining the release frequencies calculated above with a consequence of release (Section 3.1 of NEDO-32997-A). Consistent with the radiological consequence calculations performed in NEDO-32997-A, the corresponding public risk with the current testing basis can be shown as follows:

3.51E-6 events/year x 0.05 Rem/event = 1.8E-4 mRem/year (Whole Body)

With an extended testing interval, this value changes to the following:

1.76E-5 events/year x 0.05 Rem/event = 8.8E-4 mRem/year (Whole Body)

These values are approximately five orders of magnitude below 10CFR20.1301(a) annual exposure limits to the general public of 100 mRem/year (Whole Body). Therefore, these release frequencies are sufficiently low that it can be concluded that a change in surveillance test frequency has minimal impact on the valve reliability and radiological consequences.

4.0 CONCLUSIONS

Though NMP1 data were not included in NEDO-32977-A (Reference 1), it was determined that the conclusions of Reference 1 are applicable to the NMP1 EFCV system, as discussed below and shown in Table 4-1. Table 4-1 summarizes the results from the NEDO-32977-A EFCV failure rate analysis, as well as the results of the EFCV failure rate analysis for Hatch, LaSalle and NMP1.

It should be noted that NMP1 was not one of the 12 BWR plants referenced in NEDO-32977-A (similar to Hatch and LaSalle County Station (LSCS)); however, as shown in Table 4-1, the NMP1 data were found to be consistent in both the time sampled and EFCV reliability when compared to the topical report data.

Table 4-1: Summary of NEDO-32977-A EFCV Failure Rate Analysis Including NMP1, LSCS and Hatch

Plant	Make of EFCV	Operating Time (years)	Operating Time (hours)	Number of Failures	Best Estimate Failure Rate (per hour)	Upper Limit Failure Rate (per hour)
Browns Ferry	Marotta	100.5	8.80E+5	3	3.41E-6	8.81E-6
Brunswick	Valcor	267	2.34E+6	0	0	1.28E-6
Clinton	Dragon	220	1.93E+6	0	0	1.55E-6
Duane Arnold (DAEC)	Marotta	1974	1.73E+7	0	0	1.73E-7
Dresden	Chemquip	922	8.07E+6	0	0	3.71E-7
Fermi 2	Dragon	930	8.15E+6	0	0	3.68E-7
Fitzpatrick	Marotta	2019	1.77E+7	0	0	1.69E-7
Hatch	Dragon and Marotta	783	6.86E+6	4	5.83E-7	1.33E-6
LSCS	Dragon	2376	2.08E+7	10	4.81E-7	8.15E-7
Monticello	Chemquip	2314	2.03E+7	1	4.93E-8	2.34E-7
NMP1	Dragon and Marotta	1040	9.11E+6	7	7.68E-7	1.44E-6
Oyster Creek	Chemquip	465	4.07E+6	0	0	7.36E-7
Susquehanna	Marotta and Valcor	144	1.26E+6	4	3.17E-6	7.26E-6
Vermont Yankee	Chemquip	1725	1.51E+7	1	6.62E-8	3.14E-7
Columbia (WNP2)	Dragon	1344	1.18E+7	2	1.69E-7	5.34E-7

The manufacturers of the NMP1 EFCVs are Marotta and Dragon Valves. As shown in Table 4-1, both vendors are used by many other BWR utilities and are well represented in Reference 1.

Section 2.0 of this report contains a table of NMP1 specific EFCV data, similar to the tables in Reference 1. These data show that the NMP1 EFCVs are similar in design, and use, to the valves used by the other member utilities. Furthermore, a failure rate analysis of the NMP1 EFCVs was done over 10 operating cycles. As depicted in Table 4-1, these results show that the NMP1 best estimate and upper bound failure rates are below the highest failure rates presented in Reference 1, and that the NMP1 failure rates do not deviate from the industry data range or trends. Section 3.0 of this report details the results of the study. This information supports the conclusion that the generic radiological consequences evaluation performed in Attachment B to NEDO-32977-A is applicable to NMP1. It is thus reasonable to conclude, as Reference 1 states, that similar results are expected at NMP1.

In summary, the NMP1 EFCVs are similar in design and application of those of the utilities that participate in the BWROG EFCV committee. Additionally, the NMP1 failure rate study indicates that the performance of the NMP1 EFCVs is comparable to the performance of those EFCVs from the utilities listed in Reference 1. Accordingly, it is reasonable to state that the conclusions of Reference 1 are applicable to NMP1 and that NMP1 is justified in seeking the surveillance frequency relaxation.

5.0 REFERENCES

1. GE Nuclear Energy, "Excess Flow Check Valve Testing Relaxation," NEDO-32977-A, June 2000.
2. Letter from H. L. Summer (Southern Nuclear Operating Company, Inc.) to U.S. Nuclear Regulatory Commission, "Edwin I. Hatch Nuclear Plant Request to Revise Technical Specifications: Excess Flow Check Valve Relaxation," HL-6105, September 19, 2001.
3. Letter from H. L. Summer (Southern Nuclear Operating Company, Inc.) to U.S. Nuclear Regulatory Commission, "Edwin I. Hatch Nuclear Plant Response to Requests for Additional Information on Technical Specification Change Request: Excess Flow Check Valve Surveillance Requirements (EFCV)," HL-6208, March 11, 2002.
4. Letter from L. N. Olshan (U.S. Nuclear Regulatory Commission) to H. L. Summer (Southern Nuclear Operating Company, Inc.), "Edwin I. Hatch Nuclear Plant, Units 1 andd 2 Re: Issuance of Amendments (TAC Nos. MB2976 and MB2977)," April 11, 2002.

ATTACHMENT 3

License Amendment Request

Nine Mile Point Nuclear Station, Unit 1
Docket No. 50-220

Revise Technical Specifications to Adopt TSTF-334, Revision 2, "Relaxed Surveillance Frequency for Excess Flow Check Valve Testing"

Proposed Markup of Technical Specification Page

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LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
c. If Specifications 3.3.4 a and b are not met, the reactor coolant system temperature shall be reduced to a value less than 215°F within ten hours.	c. In accordance with the Surveillance Frequency Control Program, ~~each~~ a representative sample of instrument-line flow check ~~valve~~ valves will be tested for operability.

ATTACHMENT 4

License Amendment Request

Nine Mile Point Nuclear Station, Unit 1
Docket No. 50-220

Revise Technical Specifications to Adopt TSTF-334, Revision 2, "Relaxed Surveillance Frequency for Excess Flow Check Valve Testing"

Proposed Markup of Technical Specification Bases Page
(for information only)

Bases Page

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The list of primary containment isolation valves is contained in the procedure governing controlled lists and has been removed from the Technical Specifications per Generic Letter 91-08. Revisions will be processed in accordance with Quality Assurance Program requirements.

a representative sample (approximately 20 percent) of

Double isolation valves are provided on lines penetrating the primary containment and open to the free space of the containment. Closure of one of the valves in each line would be sufficient to maintain the integrity of the pressure suppression system. Except where check valves are used as one or both of a set of double isolation valves, the isolation valves shall be capable of automatic initiation. Automatic initiation is required to minimize the potential leakage paths from the containment in the event of a loss-of-coolant accident. Details of the isolation valves are discussed in Section VI-D[1]. For allowable leakage rate specification, see Section 3.3.3/4.3.3.

For the design basis loss-of-coolant accident fuel rod perforation would not occur until the fuel temperature reached 1700°F which occurs in approximately 200 seconds[2]. The required closing times for all primary containment isolation valves are established to prevent fission product release through lines connecting to the primary containment.

valves

For reactor coolant system temperatures less than 215°F, the containment could not become pressurized due to a loss-of-coolant accident. The 215°F limit is based on preventing pressurization of the reactor building and rupture of the blowout panels.

In addition to routine surveillance as outlined in Section VI-D.1.0[1] ~~each~~ instrument-line flow check ~~valve~~ will be tested for operability. All instruments on a given line will be isolated at each instrument. The line will be purged by isolating the flow check valve, opening the bypass valves, and opening the drain valve to the equipment drain tank. When purging is sufficient to clear the line of non-condensibles and crud the flow-check valve will be cut into service and the bypass valve closed. The main valve will again be opened and the flow-check valve allowed to close. The flow-check valve will be reset by closing the drain valve and opening the bypass valve depressurizing part of the system. Instruments will be cut into service after closing the bypass valve. Repressurizing of the individual instruments assures that flow-check valves have reset to the open position. Alternatively, operability testing of excess flow check valves may be performed prior to installation using a test set-up that simulates an instrument line break condition.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

The nominal 10-year interval is based on performance testing as discussed in NEDO-32977-A, "Excess Flow Check Valve Testing Relaxation." Furthermore, any instrument-line flow check valve failures will be evaluated to determine if additional testing in that test interval is warranted to ensure overall reliability is maintained. Operating experience has demonstrated that these components are highly reliable and that failures to isolate are very infrequent. Therefore, testing of a representative sample was concluded to be acceptable from a reliability standpoint.

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(1) UFSAR

(2) UFSAR Section XV-C.2.0

ATTACHMENT 5

License Amendment Request

Nine Mile Point Nuclear Station, Unit 1
Docket No. 50-220

Revise Technical Specifications to Adopt TSTF-334, Revision 2, "Relaxed Surveillance Frequency for Excess Flow Check Valve Testing"

10CFR 50.55a Relief Request GV-RR-09
Related to Excess Flow Check Valve Testing Frequency

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)
GV-RR-09 (Unit 1) – Related to Excess Flow Check Valve Testing Frequency

1. ASME Code Component(s) Affected

Component ID	Class	Cat.	System
CKV-36-57	1	C	Emergency Cooling
CKV-36-62	1	C	Emergency Cooling
CKV-36-67	1	C	Emergency Cooling
CKV-36-72	1	C	Emergency Cooling
CKV-01-76	1	C	Main Steam
CKV-01-77	1	C	Main Steam
CKV-01-78	1	C	Main Steam
CKV-01-79	1	C	Main Steam
CKV-32-100	1	C	Reactor Recirculation
CKV-32-106	1	C	Reactor Recirculation
CKV-32-112	1	C	Reactor Recirculation
CKV-32-118	1	C	Reactor Recirculation
CKV-32-125	1	C	Reactor Recirculation
CKV-32-131	1	C	Reactor Recirculation
CKV-32-138	1	C	Reactor Recirculation
CKV-32-144	1	C	Reactor Recirculation
CKV-32-151	1	C	Reactor Recirculation
CKV-32-157	1	C	Reactor Recirculation
CKV-32-164	1	C	Reactor Recirculation
CKV-32-170	1	C	Reactor Recirculation
CKV-32-177	1	C	Reactor Recirculation
CKV-32-183	1	C	Reactor Recirculation
CKV-32-204	1	C	Reactor Recirculation
CKV-32-210	1	C	Reactor Recirculation
CKV-32-215	1	C	Reactor Recirculation
CKV-32-221	1	C	Reactor Recirculation
CKV-32-226	1	C	Reactor Recirculation
CKV-32-232	1	C	Reactor Recirculation

Component ID	Class	Cat.	System
CKV-32-237	1	C	Reactor Recirculation
CKV-32-243	1	C	Reactor Recirculation
CKV-32-248	1	C	Reactor Recirculation
CKV-32-254	1	C	Reactor Recirculation
CKV-32-64	1	C	Reactor Recirculation
CKV-32-70	1	C	Reactor Recirculation
CKV-32-76	1	C	Reactor Recirculation
CKV-32-82	1	C	Reactor Recirculation
CKV-32-88	1	C	Reactor Recirculation
CKV-32-94	1	C	Reactor Recirculation
CKV-44.1-07	1	C	Reactor Recirculation
CKV-44.1-12	1	C	Reactor Recirculation
CKV-36-120	1	C	Reactor Vessel Instrumentation
CKV-36-125	1	C	Reactor Vessel Instrumentation
CKV-36-130	1	C	Reactor Vessel Instrumentation
CKV-36-135	1	C	Reactor Vessel Instrumentation
CKV-36-140	1	C	Reactor Vessel Instrumentation
CKV-36-145	1	C	Reactor Vessel Instrumentation
CKV-36-160	1	C	Reactor Vessel Instrumentation
CKV-36-165	1	C	Reactor Vessel Instrumentation
CKV-36-170	1	C	Reactor Vessel Instrumentation
CKV-36-175	1	C	Reactor Vessel Instrumentation
CKV-36-48	1	C	Reactor Vessel Instrumentation
CKV-36-53	1	C	Reactor Vessel Instrumentation

2. <u>Applicable Code Edition and Addenda</u>

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-3522(a), "Category C Check Valves," states, in part, "During operation at power, each check valve shall be exercised or examined in a manner that verifies obturator travel by using the methods in ISTC-5221."

ISTC-3522(c), Category C Check Valves, "If exercising is not practicable during operation at power and cold shutdowns, it shall be performed during refueling outages."

4. Reason for Request

In accordance with 10 CFR 50.55a, "Codes and Standards," paragraph (z)(1), relief is requested from the requirements of ASME OM Code ISTC-3522 for the subject valves. The basis of the relief request is that the proposed alternative would provide an acceptable level of quality and safety.

The ASME OM Code requires check valves to be exercised quarterly during plant operation, or if valve exercising is not practicable during plant operation and cold shutdown, it shall be performed during refueling outages.

Nine Mile Point Unit 1 (NMP1) is presently testing all excess flow check valves (EFCVs, aka, instrument-line flow check valves) in accordance with the Surveillance Frequency Control Program (SFCP) per the NMP1 Technical Specifications (TS) Surveillance Requirement (SR) 4.3.4c frequency. The NMP1 SFCP currently requires testing each EFCV on a 24-month frequency for SR 4.3.4c. A license amendment request (LAR), which is being submitted to the NRC in parallel to this request, proposes to implement Technical Specification Task Force Improved Standard Technical Specifications Change Traveler TSTF-334, Revision 2, by relaxing the number of EFCVs tested by TS SR 4.3.4c from "each" to "a representative sample" in accordance with the SFCP. The representative sample is based on approximately 20 percent of the EFCVs such that each valve will be tested at least once every 10 years (nominal).

In the interim, NMP1 will continue testing the EFCVs on a 24-month frequency, as documented in the SFCP, and in accordance with the IST Program requirements pursuant to 10 CFR 50.55a. NMP1 proposes, following approval of the LAR discussed above, to implement a sampling program for testing the EFCVs such that all EFCVs will be tested on a representative sampling basis (i.e., approximately 20 percent every refueling outage), and that all EFCVs will be tested at least once within a ten-year frequency.

These valves are located on instrument lines that provide information to station operations personnel, as well as automatic trip systems for normal and emergency operation of the station. Exercising the excess flow check valves during normal operation imposes an undue risk to plant operations personnel since the fluid medium is reactor coolant at normal reactor pressure and temperature and requires system alignment to provide a test medium source. The instruments on the lines protected by these check valves are typically required to operate during cold shutdowns as well as during normal operation. Exercising the excess flow check valve requires removing the corresponding instrument from service.

This could cause spurious instrument signal fluctuations to occur, resulting in the inadvertent automatic initiation or trip of systems.

Valve testing requires extensive equipment setup and system reconfiguration. Exercising during cold shutdowns is costly and burdensome with no increase in safety and is not considered practical. (Ref. NUREG-1482, Rev. 2, Section 4.1.6).

5. Proposed Alternative and Basis for Use

EFCV reverse flow exercising will be conducted by testing a representative sample (i.e., approximately 20 percent) of EFCVs every refueling outage, such that each EFCV will be tested at least once every 10 years.

Industry experience as documented in Boiling Water Reactor (BWR) Owners Group Licensing Topical Report NEDO-32977-A, "Excess Flow Check Valve Testing Relaxation," indicates that EFCVs have a very low failure rate. A review of the maintenance history for NMP1 EFCVs has shown that they have been highly reliable over the life of the plant. The NMP1 test experience is consistent with the findings in the NEDO document. The NEDO document indicates that many reported test failures at other plants were related to test methodologies and not actual EFCV failures. A detailed analysis of the maintenance history and a comparison to the acceptance criteria in NEDO-32977-A are provided in Attachments 1 and 2 to the License Amendment Request that this relief request is associated with. The analysis concludes the maintenance history for the NMP1 EFCVs supports use of the representative sampling basis for determining the testing schedule for the EFCVs. Thus, the EFCVs at NMP1, consistent with the industry, have exhibited a high degree of reliability, availability, and the alternate sampling approach justified by application of the NEDO-32977 analysis provides an acceptable level of quality and safety.

The basis of the relief request is that the proposed alternative would provide an acceptable level of quality and safety. Therefore, this proposed alternative is 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 began on January 1, 2019, and is scheduled to end on December 31, 2028.

7. Precedent

1. Letter from M. Markley (U.S. Nuclear Regulatory Commission) to C. R. Pierce (Southern Nuclear Operating Company, Inc.), "Edwin I. Hatch Nuclear Plant, Units 1 and 2 – Inservice Testing Program Relief Request and Alternatives for Pumps and Valves – Fifth Ten-Year Interval (CAC Nos. M176238, MF6239, MF6240, MF6241, MF6242, MF6243, MF6244, MF6245, MF6246, and MF6247)," dated December 30, 2015 (ADAMS Accession No. ML15310A406)

2. Letter from R. J. Laufer (U.S. Nuclear Regulatory Commission) to B. L. Shriver (PPL Susquehanna, LLC), "Susquehanna Steam Electric Station, Units 1 and 2 – Third 10-Year Interval Inservice Testing (IST) Program Plans [Request No. RR03] (TAC Nos. MC3382, MC3383, MC3384, MC3385, MC3386, MC3387, MC3388, MC3389, MC4421, MC4422)," dated March 10, 2005 (ADAMS Accession No. ML050690239)

3. Letter from P. Tam (U.S. Nuclear Regulatory Commission) to J. H. Mueller (Niagara Mohawk Power Corporation), "Nine Mile Point Nuclear Station, Unit No. 2 – Authorization of Alternative [Request GVRR-08] Regarding Excess Flow Check Valve Testing Frequency (TAC No. MB1491)," dated September 17, 2001 (ADAMS Accession No. ML012340462)

4. Letter from D. Wrona (U.S. Nuclear Regulatory Commission) to B. Hanson (Exelon Generation Company, LLC), "LaSalle County Station, Units 1 and 2 - Relief from the Requirements of the ASME Code for Operation and Maintenance of Nuclear Power Plants (EPID L-2019-LLR-0004)," dated July 3, 2018 (ADAMS Accession No. ML18163A054)