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DEC 1 2 2013

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

SUSQUEHANNA STEAM ELECTRIC STATION REVISED PROPOSED RELIEF REQUESTS FOR THE FOURTH TEN-YEAR INSERVICE TESTING INTERVAL FOR SUSQUEHANNA UNITS 1 AND 2 PLA-7120

Docket Nos. 50-387 and 50-388

References: 1. PPL Letter (PLA-7055), "Proposed Relief Requests for the Fourth Ten-Year Inservice Testing Interval for Susquehanna Units 1 and 2," dated October 08, 2013.

2. NRC Letter, "Supplemental Information Needed for Acceptance of Requested Licensing Action RE: Proposed Relief Requests for the Fourth 10-Year Inservice Testing Interval (TAC Nos. MF2905 Through MF2915)," dated November 27, 2013.

The purpose of this letter is to provide the supplemental information, which was requested in Reference 2. The Relief Requests cited have been revised to correct the Code reference errors and to eliminate the request not to test additional valves if the lower setpoint is exceeded. Also, two relief requests have been withdrawn. This letter supersedes Reference 1 in its entirety.

Pursuant to 10CFR50.55a(a)(3)(i) and 10CFR50.55a(a)(3)(ii), PPL Susquehanna, LLC hereby requests NRC authorization of the enclosed relief requests associated with the Fourth Ten-Year Inservice Testing (IST) Interval for the Susquehanna Steam Electric Station (SSES) Units 1 and 2. The Fourth Inspection Interval for the SSES IST program will commence on June 1, 2014 and will comply with the American Society of Mechanical Engineers Operations and Maintenance Code (OM Code) 2004 Edition through the 2006 Addenda.

The following Relief Requests are presented for consideration and review:

1RR-01/2RR-01 – Requests an alternative in accordance with 10CFR50.55a(a)(3)(i) to the requirements of OM Code, Subsection ISTC for the testing of Category "C" Check Valves. In lieu of the requirements of the OM Code, Subsection ISTC requirements, an alternative is requested which provides an acceptable level of quality and safety.

1RR-02/2RR-02 – Requests an alternative in accordance with 10CFR50.55a(a)(3)(i) to the requirements of OM Code, Subsection ISTC for the testing frequency of Pressure Isolation Valves (PIVs). In lieu of the requirements of the OM Code, Subsection ISTC requirements, an alternative is requested which provides an acceptable level of quality and safety.

1RR-03/2RR-03 – Requests an alternative in accordance with 10CFR50.55a(a)(3)(i) to the requirements of OM Code, Appendix I for the testing frequency of pressure relief valves. In lieu of the requirements of the OM Code, Appendix I, a proposed alternative is provided requested which provides an acceptable level of quality and safety.

1RR-04/2RR-04 - Requests an alternative in accordance with 10CFR50.55a(a)(3)(ii) to the requirements of OM Code for pump and valve testing frequencies. Specifically, this alternative provides a "tolerance band" to frequencies specified in the OM Code. In lieu of the frequency requirements of the OM Code, a proposed alternative is requested which provides an acceptable level of quality and safety.

1RR-05 - Requests an alternative in accordance with 10CFR50.55a(a)(3)(i) to the requirements of OM Code, Subsection ISTC for Category "C" Check Valve testing frequency. In lieu of the requirements of the OM Code, Appendix I, a proposed alternative is requested which provides an acceptable level of quality and safety.

PPL Susquehanna, LLC requests that the NRC authorize the attached proposed alternatives by April 25, 2014 to support implementation of the fourth ten-year IST interval. The attached requests are proposed for the duration of the fourth ten-year inspection interval.

Should you have any questions regarding this letter, please contact Mr. John Tripoli, Manager, Nuclear Regulatory Affairs, at (570) 542-3100.

This letter contains no new regulatory commitments.

Sincerely,

J. A. Franke

Attachments:
Attachment 1 - Relief Request 1RR01
Attachment 2 - Relief Request 1RR02
Attachment 3 - Relief Request 1RR03
Attachment 4 - Relief Request 1RR04
Attachment 5 - Relief Request 1RR05
Attachment 6 - Relief Request 2RR01
Attachment 7 - Relief Request 2RR02
Attachment 8 - Relief Request 2RR03
Attachment 9 - Relief Request 2RR04

Copy: NRC Document Control Desk
Mr. J. E. Greives, NRC Sr. Resident Inspector
Mr. J. A. Whited, NRC Project Manager
Mr. L. J. Winker, PA DEP/BRP

Attachment 1 to PLA-7120

Relief Request 1RR01

RELIEF REQUEST 1RR01

Relief in accordance with 10 CFR 50.55a (a)(3)(i)

Alternative Provides Acceptable Level of Quality and Safety

1. ASME Code Component(s) Affected

Valve	System	Cat	Class
XV141F009	Nuclear Boiler	C	1
XV141F070A	Nuclear Boiler	C	1
XV141F070B	Nuclear Boiler	C	1
XV141F070C	Nuclear Boiler	C	1
XV141F070D	Nuclear Boiler	C	1
XV141F071A	Nuclear Boiler	C	1
XV141F071B	Nuclear Boiler	C	1
XV141F071C	Nuclear Boiler	C	1
XV141F071D	Nuclear Boiler	C	1
XV141F072A	Nuclear Boiler	C	1
XV141F072B	Nuclear Boiler	C	1
XV141F072C	Nuclear Boiler	C	1
XV141F072D	Nuclear Boiler	C	1
XV141F073A	Nuclear Boiler	C	1
XV141F073B	Nuclear Boiler	C	1
XV141F073C	Nuclear Boiler	C	1
XV141F073D	Nuclear Boiler	C	1
XV14201	Nuclear Boiler	C	1
XV14202	Nuclear Boiler	C	1
XV142F041	Nuclear Boiler	C	1
XV142F043A	Nuclear Boiler	C	1
XV142F043B	Nuclear Boiler	C	1
XV142F045A	Nuclear Boiler	C	1
XV142F045B	Nuclear Boiler	C	1
XV142F047A	Nuclear Boiler	C	1
XV142F047B	Nuclear Boiler	C	1
XV142F051A	Nuclear Boiler	C	1
XV142F051B	Nuclear Boiler	C	1
XV142F051C	Nuclear Boiler	C	1
XV142F051D	Nuclear Boiler	C	1
XV142F053A	Nuclear Boiler	C	1
XV142F053B	Nuclear Boiler	C	1
XV142F053C	Nuclear Boiler	C	1
XV142F053D	Nuclear Boiler	C	1
XV142F055	Nuclear Boiler	C	1
XV142F057	Nuclear Boiler	C	1
XV142F059A	Nuclear Boiler	C	1
XV142F059B	Nuclear Boiler	C	1
XV142F059C	Nuclear Boiler	C	1
XV142F059D	Nuclear Boiler	C	1
XV142F059E	Nuclear Boiler	C	1
XV142F059F	Nuclear Boiler	C	1

Valve	System	Cat	Class
XV142F059G	Nuclear Boiler	C	1
XV142F059H	Nuclear Boiler	C	1
XV142F059L	Nuclear Boiler	C	1
XV142F059M	Nuclear Boiler	C	1
XV142F059N	Nuclear Boiler	C	1
XV142F059P	Nuclear Boiler	C	1
XV142F059R	Nuclear Boiler	C	1
XV142F059S	Nuclear Boiler	C	1
XV142F059T	Nuclear Boiler	C	1
XV142F059U	Nuclear Boiler	C	1
XV142F061	Nuclear Boiler	C	1
XV143F003A	Reactor Recirculation	C	1
XV143F003B	Reactor Recirculation	C	1
XV143F004A	Reactor Recirculation	C	1
XV143F004B	Reactor Recirculation	C	1
XV143F009A	Reactor Recirculation	C	1
XV143F009B	Reactor Recirculation	C	1
XV143F009C	Reactor Recirculation	C	1
XV143F009D	Reactor Recirculation	C	1
XV143F010A	Reactor Recirculation	C	1
XV143F010B	Reactor Recirculation	C	1
XV143F010C	Reactor Recirculation	C	1
XV143F010D	Reactor Recirculation	C	1
XV143F011A	Reactor Recirculation	C	1
XV143F011B	Reactor Recirculation	C	1
XV143F011C	Reactor Recirculation	C	1
XV143F011D	Reactor Recirculation	C	1

RELIEF REQUEST 1RR01 (continued)

Valve	System	Cat	Class
XV143F012A	Reactor Recirculation	C	1
XV143F012B	Reactor Recirculation	C	1
XV143F012C	Reactor Recirculation	C	1
XV143F012D	Reactor Recirculation	C	1
XV143F040A	Reactor Recirculation	C	1
XV143F040B	Reactor Recirculation	C	1
XV143F040C	Reactor Recirculation	C	1
XV143F040D	Reactor Recirculation	C	1
XV143F057A	Reactor Recirculation	C	1
XV143F057B	Reactor Recirculation	C	1
XV14411A	Reactor Water Cleanup	C	1
XV14411B	Reactor Water Cleanup	C	1
XV14411C	Reactor Water Cleanup	C	1
XV14411D	Reactor Water Cleanup	C	1
XV144F046	Reactor Water Cleanup	C	1
XV149F044A	Reactor Core Isolation Cooling	C	1
XV149F044B	Reactor Core Isolation Cooling	C	1
XV149F044C	Reactor Core Isolation Cooling	C	1
XV149F044D	Reactor Core Isolation Cooling	C	1
XV155F024A	High Pressure Coolant Injection	C	1
XV155F024B	High Pressure Coolant Injection	C	1
XV155F024C	High Pressure Coolant Injection	C	1
XV155F024D	High Pressure Coolant Injection	C	1
XV15109A	Residual Heat Removal	C	1
XV15109B	Residual Heat Removal	C	1

Valve	System	Cat	Class
XV15109C	Residual Heat Removal	C	1
XV15109D	Residual Heat Removal	C	1
XV152F018A	Core Spray	C	1
XV152F018B	Core Spray	C	1

RELIEF REQUEST 1RR01 (continued)

These valves are instrumentation line excess flow check valves (EFCVs) provided in each instrument line process line that penetrates primary containment in accordance with Regulatory Guide 1.11. The EFCVs are designed to close upon rupture of the instrument line downstream of the EFCV and otherwise remain open. The lines are sized and/or orificed such that off-site dose will be substantially below 10 CFR 100 limits in the event of a rupture.

2. Applicable Code Edition and Addenda

ASME OM Code 2004 Edition through 2006 Addenda

3. Applicable Code Requirement

ISTC-3522(c), "Category C Check Valves"

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

ISTC- 3700, "Position Verification Testing"

"Valves with remote position indicators shall be observed locally at least once every 2 years to verify that valve operation is accurately indicated."

4. Basis for Relief

Pursuant to 10CFR 50.55a, "Codes and Standards," paragraph (a)(3), relief is requested from the requirements of ASME OM Code ISTC-3522(c) and ISTC-3700. The basis of the relief request is that the proposed alternative would provide an acceptable level of quality and safety.

Testing the subject valves quarterly or during cold shutdown is not practicable, based on plant conditions. These valves have been successfully tested throughout the life of the Susquehanna Steam Electric Station Unit 1 and they have shown no degradation or other signs of aging.

The technology for testing these valves is simple and has been demonstrated effectively during the operating history of Susquehanna Steam Electric Station Unit 1. The basis for this alternative is that testing a sample of EFCVs each refueling outage provides a level of safety and quality equivalent to that of the Code-required testing.

Excess flow check valves are required to be tested in accordance with ISTC-3522, which requires exercising check valves nominally every three months to the positions required to perform their safety functions. ISTC-3522(c) permits deferral of this requirement to every reactor refueling outage. Excess flow check valves are also required to be tested in

RELIEF REQUEST 1RR01 (continued)

accordance with ISTC-3700, which requires remote position verification at least once every 2 years.

The EFCVs are classified as ASME Code Category C and are containment isolation valves. However, these valves are excluded from 10 CFR 50 Appendix J Type C leak rate testing, due to the size of the instrument lines and upstream orificing. Therefore, they have no safety-related seat leakage criterion.

The excess flow check valve is a simple device. The major components are a poppet and spring. The spring holds the poppet open under static conditions. The valve will close upon sufficient differential pressure across the poppet. Functional testing of the valve is accomplished by venting the instrument side of the valve. The resultant increase in flow imposes a differential pressure across the poppet, which compresses the spring and decreases flow through the valve.

Functional testing is required by Technical Specification Surveillance Requirement 3.6.1.3.9. System design does not include test taps upstream of the EFCV. For this reason, the EFCVs cannot be isolated and tested using a pressure source other than reactor pressure.

The testing described above requires removal of the associated instrument or instruments from service. Since these instruments are in use during plant operation, removal of any of these instruments from service may cause a spurious signal, which could result in a plant trip or an unnecessary challenge to safety systems. Additionally, process liquid will be contaminated to some degree, requiring special measures to collect flow from the vented instrument side and will contribute to an increase in personnel radiation exposure.

Industry experience as documented in NEDO-32977-A indicates the ECFVs have a very low failure rate. At Susquehanna, the failure rate has been approximately 1%. Only half of these failures have resulted in replacement of the EFCV. The Susquehanna test history shows no evidence of common mode failure. This Susquehanna test experience is consistent with the findings of NEDO. The NEDO indicates similarly that many reported test failures at other plants were related to test methodologies and not actual EFCV failures. Thus, the ECFVs at Susquehanna, consistent with the industry, have exhibited a high degree of reliability, availability, and provide an acceptable level of quality and safety.

Testing on a cold shutdown frequency is impractical considering the large number of valves to be tested and the condition that reactor pressure greater than 500 psig is needed for testing. In this instance, considering the number of valves to be tested and the conditions required for testing, it is also a hardship to test all these valves during refueling outages. Recent improvements in refueling outage schedules minimized the time that is planned for refueling and testing activities during the outages.

RELIEF REQUEST 1RR01 (continued)

The appropriate time for performing excess flow check valve test is during refueling outages in conjunction with vessel hydrostatic testing. As a result of shortened outages, decay heat levels during hydrostatic tests are higher than in the past. If the hydrostatic test were extended to test all EFCVs, the vessel could require depressurization several times to avoid exceeding the maximum bulk coolant temperature limit. This is an evolution that challenges the reactor operators and thermally cycles the reactor vessel. This evolution should be avoided if possible. Also, based on past experience, excess flow check valve testing during hydrostatic testing becomes the outage critical path and could possibly extend the outage by two days if all EFCVs were to be tested during this time frame.

5. **Proposed Alternate Testing**

As an alternative to testing all EFCVs during the refueling outage, a sampling plan will be implemented. This plan will test certain excess flow check valves immediately preceding the refueling outage while the reactor is at power, while also instituting the appropriate conditions for testing (reactor press > 500 psig). This alternative provides an acceptable level of quality and safety. Performance of this excess flow check valve testing prior to the outage will be scheduled such that, in the event of a failure, the resulting action statement and limiting condition of operation will encompass the planned shutdown for the refueling outage. Using this strategy, unplanned, unnecessary plant shutdowns as a result of excess flow cheek valve testing will be avoided.

Functional testing with verification that flow is checked will be performed per Technical Specification 3.6.1.3.9, either immediately preceding a planned refueling outage or during the refueling outage for certain EFCVs. For those valves tested prior to the refueling outage, appropriate administrative and scheduling controls will be established.

Surveillance Requirement 3.6.1.3.9 allows a "representative sample" of EFCVs to be tested every 24 months, such that each EFCV will be tested at least once every ten years (nominal).

The EFCVs have position indication in the control room. Check valve remote position indication is excluded from Regulatory Guide 1.97 as a required parameter for evaluating containment isolation. The remote position indication will be verified in the closed direction at the same frequency as the exercise test, which will be performed at the frequency prescribed in Technical Specification Surveillance Requirement 3.6.1.3.9. After the close position test, the valve will be reset, and the remote open position indication will be verified. Although inadvertent actuation of an EFCV during operation is highly unlikely due to the spring poppet design, Susquehanna verifies the EFCVs indicate open in the control room at a frequency greater than once every two years.

In summary, considering the extremely low failure rate along with personnel and plant safety concerns to perform testing, the alternative sampling plan proposed provides an acceptable level of quality and safety.

RELIEF REQUEST 1RR01 (continued)

6. Duration of Relief Request

This proposed alternative is requested for the duration of the Fourth Ten-Year Interval Susquehanna Steam Electric Station Unit 1 Inservice Test (IST) program (June 1, 2014 through May 31, 2024). This is similar to the relief request approved for the Third Ten-Year Interval Susquehanna Steam Electric Station Unit 1 IST Program (Accession No. ML050690239).

7. Precedent

NRC Safety Evaluation for Fermi 2, Relief Request VRR-011 Relating to the Third 10-Year Interval Inservice Testing Program, Docket No. 50-341 (TAC NO. ME 2558, ME 2557 and ME 2556).

Attachment 2 to PLA-7120

Relief Request 1RR02

RELIEF REQUEST 1RR02

Relief in accordance with 10 CFR 50.55a (a)(3)(i)

Alternative Provides Acceptable Level of Quality and Safety

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

Valve	System	Category	Class	App J
HV151F008	RHR SHUTDOWN COOLING SUCTION OB ISO VLV	A	1	Yes
HV151F009	RHR SHUTDOWN CLG SUCT IB ISO VLV	A	1	Yes
HV151F015A/B	RHR LOOP A/B INJECTION OB ISO VLVS	A	1	Yes
HV151F022	RHR HEAD SPRAY IB SHUTOFF	A	1	Yes
HV151F023	RHR REACTOR HEAD SPRAY FLOW CONTROL VLV	A	2	Yes
HV151F050A/B	RHR LP A&B TESTABLE CHECK VALVES	A/C	1	No
HV151F122A/B	RHR/LPCI INJECTION TESTABLE CHECK BYPASS VALVES	A	1	No
HV152F005A/B	CORE SPRAY LOOP A IB INJECTION SHUTOFF VLV	A	1	Yes
HV152F006A/B	CORE SPRAY LOOP A/B TESTABLE CKV	A/C	1	Yes
HV152F037A/B	CORE SPRAY LOOP A/B TESTABLE CKV BYPASS AOV	A	1	Yes

These valves are the Category A and A/C Pressure isolation Valves (PIVs) for Residual Heat Removal System (RHR), Low Pressure Coolant Injection System (LPCI), Core Spray and Reactor head Spray for Susquehanna Steam Electric Station (SSES) Unit 1. They provide isolation and prevent over pressurization of the low pressure piping between the Emergency Core Cooling System (ECCS) and Reactor Coolant System (RCS) boundaries.

2. **Applicable Code Edition and Addenda**

ASME OM Code 2004 through 2006 addenda

RELIEF REQUEST 1RR02 (continued)

3. Applicable Code Requirement

This request applies to the pressure isolation valve (PIV) leak test frequency referenced in the following requirements:

ISTC- 3630 Leakage Rate for Other Than Containment Isolation Valves, states that 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 years.

4. Basis for Relief

Pursuant to 10 CFR 50.55a, "Codes and Standards," paragraph (a)(3)(i), relief is requested from the requirement of ASME OM Code ISTC-3630(a). The basis of the relief request is that the proposed alternative would provide an acceptable level of quality and safety.

ISTC-3630 requires that leakage rate testing for PIVs be performed at least once every 2 years. PIVs are not specifically included in the scope for performance-based testing as provided for in 10 CFR Part 50, Appendix J, Option B, while the motor operated PIVs and check valve HV152F006A/B affected by this request are CIVs and tested in accordance with the 10 CFR 50 Appendix J Program. Check valve PIVs, HV151F050A/B and HV151F122A/B are not within the Appendix J scope.

The concept behind the Option B Alternative for containment isolation valves is that licensees should be allowed to adopt cost effective methods for complying with regulatory requirements. Additionally, NEI 94-01, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," describes the risk-informed basis for the extended test intervals under Option B. That justification shows that for valves which have demonstrated good performance by passing their associated leak rates tests for two consecutive 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 statement that "the risk impact associated with increasing [leak rate] test intervals is negligible (less than 0.1% of total risk)." The valves identified in this relief request are all water applications. The PIV testing is performed with water pressurized to normal plant operating pressures in accordance with ISTC-3630. This relief request is intended to provide for a performance-based scheduling of PIV tests at SSES. The reason for requesting this relief is dose reduction / ALARA. Recent historical data was used to identify that PIV testing alone each refuel outage incurs total dose of approximately 500 miliRem. Assuming all of the PIVs remain classified as good

RELIEF REQUEST 1RR02 (continued)

performers the extended test intervals would provide for a savings of approximately 1.0 Rem over the 4-year period.

NUREG 0933, "Resolution of Generic Safety Issues," Issue 105 (Interfacing Systems LOCA at LWRs) discussed the need for PIV leak rate testing based primarily on three pre-1980 historical failures of applicable valves industry-wide. These failures all involved human errors in either operations or maintenance. None of these failures involved 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 valves ability to re-position from opened to closed. For check valves, such functional testing is accomplished per ASME OM Code ISTC-3522 and ISTC-3520. Power-operated valves are routinely full stroke tested per ASME OM Code to ensure their functional capabilities. At SSES, these functional tests for motor operated PIVs are performed on a cold shutdown frequency. The functional testing of the PIV check valve is performed in accordance with ISTC-5221 "Valve Obturator Movement". Performance of separate 2 year PIV leak rate testing does not contribute any additional assurance of functional capability; it only determines the seat tightness of the closed valves.

PIV testing is performed with water pressurized to normal plant operating pressures in accordance with ISTC-3630. The intent of this relief request is to allow for a performance-based approach to the scheduling of PIV leakage testing. It has been shown that Interfacing System Loss of Coolant Accident (ISLOCA) represents a small risk impact to BWRs such as SSES.

NUREG/CR-5928, "Final Report of the NRC-sponsored ISLOCA Research Program" (ADAMS Accession No. ML072430731), evaluated the likelihood and potential severity of ISLOCA events in Boiling Water Reactors (BWR) and Pressurized Water Reactors (PWR). The BWR design used as a reference for this analysis was a BWR-4 with Mark I containment. SSES is listed as a similar plant. The BWR systems were individually analyzed and in each case the report concluded that the system was "judged to not be an important consideration with respect to ISLOCA risk." Section 4.3 of the report concluded the BWR portion of the analysis by saying "ISLOCA is not a risk concern for the BWR plant examined here."

The functional tests for PIVs are performed only at a cold shutdown frequency. Such testing is not performed online in order to prevent any possibility of an inadvertent ISLOCA condition. The functional testing of the PIVs is adequate to identify any abnormal condition that might affect closure capability.

5. Proposed Alternative Testing

SSES proposes to perform PIV testing at intervals ranging from every refuel to every third refuel. The specific interval for each valve would be a function of its performance

RELIEF REQUEST 1RR02 (continued)

and would be established in a manner consistent with the Containment Isolation valve (CIV) process under 10CFR50 Appendix J, Option B, program guidance. The test frequency will be established such that if any of the valves subject to a CIV and a PIV test, fail either test, the test interval for both tests will be reduced to once every 24 months until they can be re-classified as good performers per the performance evaluation requirements of Appendix J, Option B. The test intervals for the valves with a PIV-only function will be determined in a similar manner as is done for CIV testing under Option B. The test interval may be extended upon completion of two consecutive periodic PIV tests with results within prescribed acceptance criteria. Any PIV test failure will require a return to the initial interval until good performance can again be established.

6. **Duration of Relief Request**

This proposed alternative is requested for the duration of the Fourth Ten-Year Interval Susquehanna Steam Electric Station Unit 1 IST program (June 1, 2014 through May 31, 2024). This is similar to relief request VRR-07 approved for James A FitzPatrick's Fourth 10-Year IST interval, which commenced on October 1, 2007.

7. **Precedents**

This relief request was approved for Fermi Power Station for the Third 120 month Interval. Letter from R. Pascarelli (US NRC) to J. Davis (Detroit Edison), "Fermi-2 Evaluation of Inservice Testing Program Relief Requests VRR-011, VRR-012, and VRR-013," dated September 28, 2010.

Attachment 3 to PLA-7120

Relief Request 1RR03

RELIEF REQUEST 1RR03

Use of Code Case OMN-17, Revision 0, on the Class 1 Main Steam Relief Valves

Relief in accordance with 10 CFR 50.55a (a)(3)(i)

Alternative Provides Acceptable Level of Quality and Safety

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

Valve	System	Category	Class
PSV141F013A	Nuclear Boiler	C	1
PSV141F013B	Nuclear Boiler	C	1
PSV141F013C	Nuclear Boiler	C	1
PSV141F013D	Nuclear Boiler	C	1
PSV141F013E	Nuclear Boiler	C	1
PSV141F013F	Nuclear Boiler	C	1
PSV141F013G	Nuclear Boiler	C	1
PSV141F013H	Nuclear Boiler	C	1
PSV141F013J	Nuclear Boiler	C	1
PSV141F013K	Nuclear Boiler	C	1
PSV141F013L	Nuclear Boiler	C	1
PSV141F013M	Nuclear Boiler	C	1
PSV141F013N	Nuclear Boiler	C	1
PSV141F013P	Nuclear Boiler	C	1
PSV141F013R	Nuclear Boiler	C	1
PSV141F013S	Nuclear Boiler	C	1

These valves are Main Steam Safety/Relief Valves. They provide overpressure protection for the reactor coolant pressure boundary to prevent unacceptable radioactive release and exposure to plant personnel.

2. **Applicable Code Edition and Addenda**

ASME OM Code 2004 Edition through 2006 Addenda

3. **Applicable Code Requirements**

In ASME OM Code Mandatory Appendix I, "Inservice Testing of Pressure Relief Devices in Light-Water Reactor Nuclear Power Plants," Section I-1320(a), "Class 1 pressure relief valves shall be tested at least once every 5 years, starting with initial electric power generation.

The licensee proposes to extend the test interval for these valves from 5 years to 6 years (with a 6-month grace period) while still maintaining the required 24-month/20%

RELIEF REQUEST 1RR03 (continued)

sampling requirement.

4. Basis for Relief

In accordance with 10 CFR 50.55a(a)(3)(i), the licensee's relief request seeks approval of an alternative to the 5-Year Test Interval requirements of ASME OM Code, Appendix I, Section I-1320(a), for the Susquehanna Main Steam Safety/Relief Valves (MSRVs) for Unit 1. Susquehanna requests that the test interval be increased from 5 years to 72 months in accordance with ASME OM Code Case, OMN-17, "Alternative Rules for Testing ASME Class 1 Pressure Relief / Safety Valves," so that the test interval for any individual valve that is in service shall not exceed 72 months except that a 6-month grace period is allowed to coincide with refueling outages to accommodate extended shutdown periods. The basis of the relief request is that the proposed alternative would provide an acceptable level of quality and safety.

Without Code relief for 24-month fuel cycles, strict Code compliance would restrict Susquehanna's operating philosophy to not operate with weeping MSRVs as Code testing would be required to be completed within 5 years. This testing strategy does not account for any leaking valves that may need to be refurbished. Since Susquehanna's philosophy is to share spare valves between both units, (the valves that are removed from one unit are installed in the other unit's next refueling outage), this testing strategy is less than adequate. This strategy could only be accomplished if a large population of MSRVs are tested each outage or additional spare valves are purchased. More than 8 valves would need to be sent to the offsite testing facility during a refueling outage. The testing and return of these valves would have to be completed expeditiously in order to not impact the refuel outage schedule duration. For this reason, additional expenditures would be incurred to purchase and test a greater number of valves each outage. Without Code relief, the additional outage work would be contrary to the principles of ALARA and could compromise radiation safety. Because of the location of certain MSRVs in the containment, interferences exist that would require the removal of more valves and piping for those valves that must be removed for the sample testing. This results in more radiation exposure to the maintenance personnel than is desirable.

With Code relief, the 16 MSRVs per unit can be tested within 6 years to complete the Code required testing for the total population and accommodate any weeping MSRVs. The increased testing over only 2 refuel cycles would result in no additional safety benefit to the plant. Susquehanna has had excellent performance with MSRVs over the last 10 years. Since 1987, Susquehanna has imposed a more conservative as-left leakage criterion on the testing facility than was specified in the General Electric Specification and incorporated in the PPL Specification for testing Crosby style relief valves. The criterion imposed on the test lab is 0 ml/5 minutes (via the purchase order) compared to a GE Specification "as-left" leakage criterion of 38 ml/5 minutes.

RELIEF REQUEST 1RR03 (continued)

Additionally, a review of the set point testing results (for both units) from initial operation to the present shows that the average of the set point drifts percentages is approximately -0.91%. This indicates that, in general, the MSRVs Set Pressure tends to drift slightly downward, not upward. The calculated standard deviation from the average for the data was determined to be approximately 1.68%.

Also, the testing history shows that since commercial operation, Susquehanna has had only two "as-found" set pressure test acceptance criteria failures (above +3%) of the tested valves, which required additional MSRVs to be tested.

5. Proposed Alternate Testing

For the fourth ten-year interval, Susquehanna proposes to remove at least 20% of the 16 Main Steam Safety/Relief Valves (MSRV) plus weeping valves detected during the previous operating cycle and any valves required to be removed to access scheduled or weeping valves up to a maximum of 8 valves during each refueling outage.

Additional valves above the Code required minimum 20% will be tested if the as-found setpoint exceeds +3%, -5% (as approved per Technical Specification Amendment No. 257) of the nameplate. The additional valves tested will be from the initial population removed that are in excess of the 20% Code required minimum. If one of these valves fail, then all the MSRVs would be removed and tested.

The proposed alternative will provide for disassembly and inspection of the MSRVs to verify parts are free from defects resulting from time-related degradation or maintenance induced wear. This maintenance will also help to reduce the potential for set point drift, and increase the reliability of these Safety Relief Valves to perform their design requirement functions. Consistent with the special maintenance requirement in Code Case OMN-17, critical components will be inspected for wear and defects.

Completion of Code testing will be accomplished over a period of 3 refuel cycles or 6 years. This approach results in maintenance and operational flexibility with the following benefits:

- Provides the ability to both test the Code required valves out of the population not yet tested and replace any weeping MSRVs.
- Maintains relatively leak-free MSRVs, thus minimizing the necessary run time of ECCS systems that provide suppression pool cooling.
- Consistent application of ALARA principles.
- Enhances equipment reliability.
- Results in minimal impact on outage durations.

RELIEF REQUEST 1RR03 (continued)

The MSRVs will be tested such that a minimum of 20% of the valves (previously untested, if they exist) are tested every 24 months, such that all the valves will be tested within 3 refuel cycles. This proposal utilizes the same maintenance and testing approach that was applied in 18-month refuel cycles. This alternative frequency will continue to provide assurance of the valve operational readiness and provides an acceptable level of quality and safety.

Additionally, any failures, either seat leakage or pressure set point, occurring at the test facility, as well as weeping MSRVs that develop during the operating cycle will be documented by the corrective action program, evaluated and dispositioned accordingly.

6. Duration of Relief Request

This proposed alternative is requested for the duration of the Fourth Ten-Year Interval Susquehanna Steam Electric Station Unit 1 IST program (June 1, 2014 through May 31, 2024). This is similar to the relief request approved for the Third Ten-Year Interval Susquehanna Steam Electric Station Unit 1 IST Program (Accession No. ML050690239).

7. Precedent

NRC Safety Evaluation for Beaver Valley Power Station, Unit Nos. 1 and 2, Relief Request VRR4, (TAC No. ME5752), Docket No. 50-412, February 7, 2012

NRC Safety Evaluation for Dresden Nuclear Power Station, Unit Nos. 2 and 3, Relief Request RV-02C, (TAC Nos. ME9865, ME9866, ME9869, ME9870, ME9871, and ME9872), Docket Nos. 50-237 and 50-249, March 22, 2012

NRC Safety Evaluation for Oyster Creek Nuclear Generating Station, Relief Request VR-01, (TAC No. ME7617), Docket No. 50-219, February 7, 2012

NRC Safety Evaluation for Monticello Nuclear Generating Plant, Relief Request VR-04 Relating to the Fifth 10-Year Interval Inservice Testing Program, Docket No. 50-263. (TAC Nos. ME8067, ME8088, ME8089, ME8090, ME8091, ME8092, ME8093, ME8094, ME8095, AND ME8095), September 26, 2012

Attachment 4 to PLA-7120

Relief Request 1RR04

RELIEF REQUEST 1RR04

Relief in accordance with 10 CFR 50.55a (a)(3)(ii) Hardship or Unusual Difficulty Without Compensating Increase in Level of Quality or Safety

1. ASME Code Component(s) Affected

All Pumps and Valves contained within the Inservice Testing Program scope.

2. Applicable Code Edition and Addenda

ASME OM Code 2004 Edition through 2006 Addenda, and ASME OM Code Case OMN-20 from ASME OM Code 2012 Edition.

3. Applicable Code Requirement

This request applies to the frequency specifications of the ASME OM Code. The frequencies for tests given in the ASME OM Code do not include a tolerance band.

ISTA-3120(a)	"The frequency for the inservice testing shall be in accordance with the requirements of Section IST."
ISTA-3400	Frequency of Inservice Tests
ISTC-3510	Exercising Test Frequency
ISTC-3540	Manual Valves
ISTC-3630(a)	Leakage Rate for Other Than Containment Isolation Valves Test Frequency
ISTC-3700	Position Verification Testing
ISTC-5221(c)(3)	"At least one valve from each group shall be disassembled and examined at each refueling outage; all valves in each group shall be disassembled and examined at least once every 8 years."
Appendix I, I-1320	Test Frequency, Class 1 Pressure Relief Devices
Appendix I, I-1330	Test Frequency, Class 1 Nonreclosing Pressure Relief Devices
Appendix I, I-1340	Test Frequency, Class 1 Pressure Relief Valves that are used for Thermal Relief Application
Appendix I, I-1350	Test Frequency, Classes 2 and 3 Pressure Relief Valves
Appendix I, I-1360	Test Frequency, Classes 2 and 3 Nonreclosing Pressure Relief Devices
Appendix I, I-1370	Test Frequency, Classes 2 and 3 Primary Containment Vacuum Relief Valves

RELIEF REQUEST 1RR04 (continued)

Appendix I, I-1380	Test Frequency, Classes 2 and 3 Vacuum Relief Valves, Except for Primary Containment Vacuum Relief Valves
Appendix I, I-1390	Test Frequency, Classes 2 and 3 Pressure Relief Devices That Are Used for Thermal Relief Application
Appendix II, II-4000(a)(1)	Performance Improvement Activities Interval
Appendix II, II-4000(b)(1)(e)	Optimization of Condition Monitoring Activities Interval

4. **Basis for Relief**

Pursuant to 10 CFR 50.55a, "Codes and standards," paragraph (a)(3)(ii), relief is requested from the frequency specifications of the ASME OM Code. The basis of the relief request is that the Code requirement presents an undue hardship without a compensating increase in the level of quality or safety.

ASME OM Code Section IST establishes the inservice test frequency for all components within the scope of the Code. The frequencies (e.g., quarterly) have always been interpreted as "nominal" frequencies (generally as defined in the Table 3.2 of NUREG 1482, Revision 1) and Owners routinely applied the surveillance extension time period (i.e., grace period) contained in the plant Technical Specifications (TS) Surveillance Requirements (SRs). The TS typically allow for a less than or equal to 25% extension of the surveillance test interval to accommodate plant conditions that may not be suitable for conducting the surveillance (SR 3.0.2). However, regulatory issues have been raised concerning the applicability of the TS "Grace Period" to ASME OM Code required inservice test frequencies irrespective of allowances provided under TS Administrative Controls (i.e., TS 5.5.6, "Inservice Testing Program," invokes SR for various OM Code frequencies).

The lack of a tolerance band on the ASME OM Code inservice test frequency restricts operational flexibility. There may be a conflict where a surveillance test could be required (i.e., its Frequency could expire), but where it is not possible or not desired that it be performed until sometime after a plant condition or associated Limiting Condition for Operation (LCO) is within its applicability. Therefore, to avoid this conflict, the surveillance test should be performed when it can be and should be performed.

The NRC recognized this potential issue in the TS by allowing a frequency tolerance as described in TS SR 3.0.2. The lack of a similar tolerance applied to OM Code testing places an unusual hardship on the plant to adequately schedule work tasks without operational flexibility.

Thus, just as with TS required surveillance testing, some tolerance is needed to allow adjusting OM Code testing intervals to suit the plant conditions and other maintenance and testing activities. This assures operational flexibility when scheduling surveillance

RELIEF REQUEST 1RR04 (continued)

tests that minimize the conflicts between the need to complete the surveillance and plant conditions.

5. **Proposed Alternative Testing**

Susquehanna proposes to use the ASME OM Code Case OMN-20, from the 2012 Edition of the ASME OM Code, as an alternative for grace period associated with Inservice Testing Requirements.

ASME OM Code establishes component test frequencies that are based either on elapsed time periods (e.g., quarterly, 2 years, etc.) or on the occurrence of plant conditions or events (e.g., cold shutdown, refueling outage, upon detection of a sample failure, following maintenance, etc.).

a. Components whose test frequencies are based on elapsed time periods shall be tested at the frequencies specified in ASME Code Section IST with a specified time period between tests as shown in the following table.

Frequency	Specified Time Period Between Tests (all values are 'not to exceed'; no minimum periods are specified)
Quarterly	92 days (or every 3 months)
Semiannually	184 days (or every 6 months)
Annually	366 days (or every year)
x Years	x calendar years where 'x' is a whole number of years ≥ 2

b. The specified time period between tests may be extended as follows:

   i. For periods specified as less than 2 years, the period may be extended by up to 25% for any given test. This is consistent with SSES TS Section 5.5.6, "Inservice Testing Program."
   ii. Period extensions may also be applied to accelerated test frequencies (e.g., pumps in Alert Range).
   iii. For periods specified as greater than or equal to 2 years, the period may be extended by up to 6 months for any given test.

RELIEF REQUEST 1RR04 (continued)

c. Components whose test frequencies are based on the occurrence of plant conditions or events (e.g., cold shutdown, refueling outage, upon detection of a sample failure, following maintenance, etc.) may not have their period between tests extended except as allowed by the ASME OM Code.

6. Duration of Relief Request

This proposed alternative is requested for the duration of the Fourth Ten-Year Interval Susquehanna Steam Electric Station Unit 1 IST program (June 1, 2014 through May 31, 2024).

7. Precedent

Generic relief has not been specifically granted to apply a tolerance band to the ASME OM Code required test frequencies. The NRC has previously accepted the application of TS SR 3.0.2 tolerance to selected OM Code frequencies as denoted in TS 5.5.6.

The prior NRC acceptance of the practice of applying TS tolerance to ASME OM Code required test frequencies provides equivalent precedence for accepting and approving this relief request.

Attachment 5 to PLA-7120

Relief Request 1RR05

RELIEF REQUEST 1RR05

Relief in accordance with 10 CFR 50.55a (a)(3)(i)

Alternative Provides Acceptable Level of Quality and Safety

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

Valve Number	System	Cat.	Safety Class
086018	Control Structure Chilled Water	C	3
086118	Control Structure Chilled Water	C	3
086241	Emergency Service Water	C	3
086341	Emergency Service Water	C	3

Valves 086018 and 086118 are six (6) inch Emergency Condenser Pump 0P171A/B discharge check valves. They have an open safety function to provide a flow path from the Emergency Condenser Pump to the Control Structure Chiller Condenser. These check valves have no closed safety function. Valves 086241 and 086341 are two (2) inch Emergency Service Water (ESW) keepfill check valves. These valves are considered part of the Control Structure Chilled Water (CSCW) system. They are keepfill check valves that allow Service water to maintain the Emergency Condenser Water Circulating (ECWC) subsystem full. The ECWC subsystem is fed from the ESW system. These check valves have a closed safety function to prevent diversion of ESW from the ECWC subsystem when operating under emergency conditions. The check valves have no open safety function.

2. **Applicable Code Edition and Addenda**

ASME OM Code 2004 Edition through 2006 Addenda

3. **Applicable Code Requirements**

ISTC-3522(c) "Category C Check Valves"

"... if exercising is not practical during operation at power and cold shutdowns, it shall be performed during refuel outages."

4. **Basis for Relief**

Pursuant to 10CFR50.55a, "Codes and Standards," paragraph (a)(3), relief is requested from the requirements of ASME OM Code ISTC-3522(c). The basis of the relief request is that the proposed alternative would provide an acceptable level of quality and safety.

The components listed above are check valves with no external means for exercising and no external position indication. The only means to verify closure is by leak testing. This

RELIEF REQUEST 1RR05 (continued)

involves setup of test equipment and system configuration changes that are a hardship without a compensating increase in quality or safety on a quarterly or cold shutdown basis. The leak testing can be performed at intervals other than refueling outages such as during system outage windows.

Prior to performing a system outage on-line, its effect on risk is evaluated in accordance with requirements of 10CFR50.65(a)(4), "Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants." This requirement states in part that "Before performing maintenance activities (including but not limited to surveillance, post-maintenance testing, and corrective and preventive maintenance), the licensee shall assess and manage the increase in risk that may result from the proposed maintenance activities."

Susquehanna Steam Electric Station (SSES) complies with the requirements of 10CFR50.65(a)(4) via application of a program governing maintenance scheduling. The program dictates the requirements for risk evaluations as well as the necessary levels of action required for risk management in each case. The program also controls operation of the on-line risk monitor system, which is based on probabilistic risk assessment (PRA). With the use of risk evaluation for various aspects of plant operations, SSES has initiated efforts to perform additional maintenance, surveillance, and testing, activities during normal operation. Planned activities are evaluated utilizing risk insights to determine the impact on safe operation of the plant and the ability to maintain associated safety margins. Individual system components, a system train, or a complete system may be planned to be out of service to allow maintenance, or other activities, during normal operation.

Leak testing may involve a system breach, if required to repair a failed valve. However, during the disassembly process to perform maintenance, the subject valve is isolated and the associated section of piping drained. Thus, the system breach does not increase the risk due to internal flooding or internal system loss-of-coolant accident. The risk associated with these activities would be bounded by the risk experienced due to the system outage. Therefore, closure testing of these valves by leak testing during schedule system outages while on-line would have no additional impact on core damage frequency.

SSES performs on-line maintenance on the Control Structure Chilled Water (CSCW) and the ESW systems. Minor maintenance work activities of limited scope require Operations authorization to perform. Also, Operations authorization is required if the activity has the potential to affect or affects a system, structure or component. It may also be scheduled as a System Outage Window. These are preplanned to occur once for each Emergency Core Cooling System (ECCS) per 2-year cycle.

Tasks performed during on-line maintenance include items such as pump inspections, relief valve testing, electrical breaker maintenance and testing, and valve diagnostic

RELIEF REQUEST 1RR05 (continued)

testing. Leak testing of the check valve is expected to take approximately 4 to 6 hours. This IST activity would be conducted simultaneously with other maintenance activities scoped in to the system outage window. Based on maintenance history and past scheduling experience, and work execution, the additional check valve leakage testing will neither extend the system outage window nor increase the overall system unavailability.

Therefore, performing IST activity on-line would change neither the duration of the on-line system outage window nor the core damage probability (CDP) associated with the existing on-line maintenance activities. For these reasons, the risk/(CDP) over the entire operating/shutdown spectrum would remain unchanged with approval of these relief requests.

If the check valve needed to be replaced, the valves used to provide the isolation boundary for the replacement of the check valves have an excellent history of providing adequate isolation. Once adequate isolation is confirmed, it is maintained by passive isolating valves or valves made passive (e.g., de-energized motor operating valves) that are controlled in accordance with SSES's Energy Control Process. A loss of isolation capability under these conditions is not considered credible due to the passive characteristics of the isolation valves.

Risk associated with on-line maintenance activities is controlled through the SSES work management process. This process includes preventive measures for maintaining safety and minimizing risk while performing on-line maintenance activities.

The level of quality associated with IST activities is independent of whether the activity is performed on-line or during an outage. The same personnel, procedures, and acceptance criteria are used in either case. The safe conduct of maintenance and IST activities is built into the work management process. The inspection activities are planned ensuring adequate isolation boundaries are established to protect both maintenance personnel involved in the activity and plant equipment.

SSES manages system outage windows on a recurring cycle. Risk insight is used to ensure that proposed work or inspection activities balance reliability with unavailability. The work selection process provides the means to ensure, through the oversight of knowledgeable personnel, that when system unavailability is to be incurred, the preventive maintenance, corrective maintenance, and other inspections required to maximize the system's reliability are included in the system outage window. In this manner, each window is scoped to maximize the reliability benefit from taking system unavailability while minimizing the unavailability such that it is maintained at a level that minimizes overall risk. PPL is confident that this rigorous work selection, scoping, and risk management system will identify all work that is more appropriately placed in outages, and schedules such work accordingly.

RELIEF REQUEST 1RR05 (continued)

Leak testing check valves and other periodic work activities in the CSCW (and ESW) system(s) will cause CSCW (and ESW) to become INOPERABLE in accordance with Technical Specifications (TS) and Technical Requirements Manual (TRM). In accordance with TS 3.7.3, operation with one Control Room Emergency Outside Air Supply (CREOAS) subsystem INOPERABLE is permitted for up to 7 days. In accordance with TS 3.7.4, operation with one control room floor cooling system INOPERABLE is permitted for up to 30 days. In accordance with TRM 3.7.9, operation with a single division of the Control Structure Chilled Water system INOPERABLE is permitted for up to 30 days. In accordance with TRM 3.8.6 (Unit 1 only), operation with a one required Emergency Switchgear Room Cooling subsystem INOPERABLE is permitted for up to 30 days. Leak testing of CSCW check valves takes between 4 and 6 hours, which would typically be accomplished within a 24-hour system work window.

In accordance with TS 3.7.2, operation with one ESW subsystem INOPERABLE is permitted for up to 7 days. In accordance with TRM 3.8.6 (Unit 1 only), operation with a one required Emergency Switchgear Room Cooling subsystem INOPERABLE is permitted for up to 30 days. Leak testing of ESW check valves takes between 4 and 6 hours, which would typically be accomplished within a 24-hour system work window.

Work that requires entry into a TS LCO REQUIRED ACTION statement is planned and scheduled with the SSES Work Management Process previously described above. The Work Management Process establishes the scope of work such that only 50% of the TS LCO time is required to perform the scheduled work. In addition, Evolution Coordinators/Engineering personnel provide coverage for resolving problems. Spare parts that are necessary for rework are identified and made available in case rework becomes necessary. Based on historical performance, performance of check valve leak testing would not affect the duration of the time spent in the LCO REQUIRED ACTION.

As more system outages are performed on-line, it is evident that selected refueling outage inservice testing activities, (e.g., closure testing by leak testing) could be performed during these system outage windows (SOW) without sacrificing the level of quality or safety. Inservice testing performed on a refueling outage frequency is currently acceptable in accordance with ASME OM Code, 2004 Edition through 2006 Addenda. By specifying testing activities on a frequency commensurate with each refueling outage, ASME OM Code, 2004 Edition through 2006 Addenda, establishes an acceptable time period between testing. Historically, the refueling outage has provided a convenient and defined time period in which testing activities could be safely and efficiently performed. However, an acceptable testing frequency can be maintained separately without being tied directly to a refueling outage. Inservice testing performed on a frequency that maintains the acceptable time period between testing activities during the operating cycle is consistent with the intent of ASME OM Code, 2004 Edition through 2006 Addenda.

Over time, approximately the same number of tests will be performed using the proposed operating cycle frequency as would be performed using the current refueling outage

RELIEF REQUEST 1RR05 (continued)

frequency. Thus, inservice testing activities performed during the proposed operating cycle test frequency provide an equivalent level of quality and safety.

5. **Proposed Alternate Testing**

Pursuant to 10CFR50.55a(a)(3)(i), SSES 1 and 2 proposes an alternative testing frequency for performing inservice testing of the valves identified above. The valves will be closure tested by leak testing on a frequency of at least once per operating cycle in lieu of once each refueling outage as currently allowed by ASME OM Code, 2004 Edition through 2006 Addenda ISTC-3522(c), "Category C Check Valves." The open safety function of check valves 086018 and 086118 will be demonstrated quarterly in conjunction with the Control Structure Chilled Water flow verification (inservice pump test). The open function of check valves 086241 and 086341 is demonstrated continuously through the keepfill function.

6. **Duration of Relief Request**

This proposed alternative is requested for the duration of the Fourth Ten-Year Interval Susquehanna Steam Electric Station Unit 1 IST program (June 1, 2014 through May 31, 2024).

Attachment 6 to PLA-7120

Relief Request 2RR01

RELIEF REQUEST 2RR01

Relief in accordance with 10 CFR 50.55a (a)(3)(i)

Alternative Provides Acceptable Level of Quality and Safety

1. <u>ASME Code Component(s) Affected</u>

Valve	System	Cat	Class
XV241F009	Nuclear Boiler	C	1
XV241F070A	Nuclear Boiler	C	1
XV241F070B	Nuclear Boiler	C	1
XV241F070C	Nuclear Boiler	C	1
XV241F070D	Nuclear Boiler	C	1
XV241F071A	Nuclear Boiler	C	1
XV241F071B	Nuclear Boiler	C	1
XV241F071C	Nuclear Boiler	C	1
XV241F071D	Nuclear Boiler	C	1
XV241F072A	Nuclear Boiler	C	1
XV241F072B	Nuclear Boiler	C	1
XV241F072C	Nuclear Boiler	C	1
XV241F072D	Nuclear Boiler	C	1
XV241F073A	Nuclear Boiler	C	1
XV241F073B	Nuclear Boiler	C	1
XV241F073C	Nuclear Boiler	C	1
XV241F073D	Nuclear Boiler	C	1
XV24201	Nuclear Boiler	C	1
XV24202	Nuclear Boiler	C	1
XV242F041	Nuclear Boiler	C	1
XV242F043A	Nuclear Boiler	C	1
XV242F043B	Nuclear Boiler	C	1
XV242F045A	Nuclear Boiler	C	1
XV242F045B	Nuclear Boiler	C	1
XV242F047A	Nuclear Boiler	C	1
XV242F047B	Nuclear Boiler	C	1
XV242F051A	Nuclear Boiler	C	1
XV242F051B	Nuclear Boiler	C	1
XV242F051C	Nuclear Boiler	C	1

Valve	System	Cat	Class
XV242F051D	Nuclear Boiler	C	1
XV242F053A	Nuclear Boiler	C	1
XV242F053B	Nuclear Boiler	C	1
XV242F053C	Nuclear Boiler	C	1
XV242F053D	Nuclear Boiler	C	1
XV242F055	Nuclear Boiler	C	1
XV242F057	Nuclear Boiler	C	1
XV242F059A	Nuclear Boiler	C	1
XV242F059B	Nuclear Boiler	C	1
XV242F059C	Nuclear Boiler	C	1
XV242F059D	Nuclear Boiler	C	1
XV242F059E	Nuclear Boiler	C	1
XV242F059F	Nuclear Boiler	C	1
XV242F059G	Nuclear Boiler	C	1
XV242F059H	Nuclear Boiler	C	1
XV242F059L	Nuclear Boiler	C	1
XV242F059M	Nuclear Boiler	C	1
XV242F059N	Nuclear Boiler	C	1
XV242F059P	Nuclear Boiler	C	1
XV242F059R	Nuclear Boiler	C	1
XV242F059S	Nuclear Boiler	C	1
XV242F059T	Nuclear Boiler	C	1
XV242F059U	Nuclear Boiler	C	1
XV242F061	Nuclear Boiler	C	1
XV243F003A	Reactor Recirculation	C	1
XV243F003B	Reactor Recirculation	C	1
XV243F004A	Reactor Recirculation	C	1

RELIEF REQUEST 2RR01 (continued)

Valve	System	Cat	Class
XV243F004B	Reactor Recirculation	C	1
XV243F009A	Reactor Recirculation	C	1
XV243F009B	Reactor Recirculation	C	1
XV243F009C	Reactor Recirculation	C	1
XV243F009D	Reactor Recirculation	C	1
XV243F010A	Reactor Recirculation	C	1
XV243F010B	Reactor Recirculation	C	1
XV243F010C	Reactor Recirculation	C	1
XV243F010D	Reactor Recirculation	C	1
XV243F011A	Reactor Recirculation	C	1
XV243F011B	Reactor Recirculation	C	1
XV243F011C	Reactor Recirculation	C	1
XV243F011D	Reactor Recirculation	C	1
XV243F012A	Reactor Recirculation	C	1
XV243F012B	Reactor Recirculation	C	1
XV243F012C	Reactor Recirculation	C	1
XV243F012D	Reactor Recirculation	C	1
XV243F040A	Reactor Recirculation	C	1
XV243F040B	Reactor Recirculation	C	1
XV243F040C	Reactor Recirculation	C	1
XV243F040D	Reactor Recirculation	C	1

Valve	System	Cat	Class
XV243F057A	Reactor Recirculation	C	1
XV243F057B	Reactor Recirculation	C	1
XV24411A	Reactor Water Cleanup	C	1
XV24411B	Reactor Water Cleanup	C	1
XV24411C	Reactor Water Cleanup	C	1
XV24411D	Reactor Water Cleanup	C	1
XV244F046	Reactor Water Cleanup	C	1
XV249F044A	Reactor Core Isolation Cooling	C	1
XV249F044B	Reactor Core Isolation Cooling	C	1
XV249F044C	Reactor Core Isolation Cooling	C	1
XV249F044D	Reactor Core Isolation Cooling	C	1
XV255F024A	High Pressure Coolant Injection	C	1
XV255F024B	High Pressure Coolant Injection	C	1
XV255F024C	High Pressure Coolant Injection	C	1
XV255F024D	High Pressure Coolant Injection	C	1
XV25109A	Residual Heat Removal	C	1
XV25109B	Residual Heat Removal	C	1
XV25109C	Residual Heat Removal	C	1
XV25109D	Residual Heat Removal	C	1
XV252F018A	Core Spray	C	1
XV252F018B	Core Spray	C	1

RELIEF REQUEST 2RR01 (continued)

These valves are instrumentation line excess flow check valves (EFCVs) provided in each instrument line process line that penetrates primary containment in accordance with Regulatory Guide 1.11. The EFCVs are designed to close upon rupture of the instrument line downstream of the EFCV and otherwise remain open. The lines are sized and/or orificed such that off-site dose will be substantially below 10 CFR 100 limits in the event of a rupture.

2. **Applicable Code Edition and Addenda**

ASME OM Code 2004 Edition through 2006 Addenda

3. **Applicable Code Requirement**

ISTC-3522(c), "Category C Check Valves"

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

ISTC- 3700, "Position Verification Testing"

"Valves with remote position indicators shall be observed locally at least once every 2 years to verify that valve operation is accurately indicated."

4. **Basis for Relief**

Pursuant to 10CFR 50.55a, "Codes and Standards," paragraph (a)(3), relief is requested from the requirements of ASME OM Code ISTC-3522(c) and ISTC-3700. The basis of the relief request is that the proposed alternative would provide an acceptable level of quality and safety.

Testing the subject valves quarterly or during cold shutdown is not practicable, based on plant conditions. These valves have been successfully tested throughout the life of the Susquehanna Steam Electric Station Unit 2 and they have shown no degradation or other signs of aging.

The technology for testing these valves is simple and has been demonstrated effectively during the operating history of Susquehanna Steam Electric Station Unit 2. The basis for this alternative is that testing a sample of EFCVs each refueling outage provides a level of safety and quality equivalent to that of the Code-required testing.

Excess flow check valves are required to be tested in accordance with ISTC-3522, which requires exercising check valves nominally every three months to the positions required to perform their safety functions. ISTC-3522(c) permits deferral of this requirement to every reactor refueling outage. Excess flow check valves are also required to be tested in

RELIEF REQUEST 2RR01 (continued)

accordance with ISTC-3700, which requires remote position verification at least once every 2 years.

The EFCVs are classified as ASME Code Category C and are also containment isolation valves. However, these valves are excluded from 10 CFR 50 Appendix J Type C leak rate testing, due to the size of the instrument lines and upstream orificing. Therefore, they have no safety-related seat leakage criterion.

The excess flow check valve is a simple device. The major components are a poppet and spring. The spring holds the poppet open under static conditions. The valve will close upon sufficient differential pressure across the poppet. Functional testing of the valve is accomplished by venting the instrument side of the valve. The resultant increase in flow imposes a differential pressure across the poppet, which compresses the spring and decreases flow through the valve.

Functional testing is required by Technical Specification Surveillance Requirement 3.6.1.3.9. System design does not include test taps upstream of the EFCV. For this reason, the EFCVs cannot be isolated and tested using a pressure source other than reactor pressure.

The testing described above requires removal of the associated instrument or instruments from service. Since these instruments are in use during plant operation, removal of any of these instruments from service may cause a spurious signal, which could result in a plant trip or an unnecessary challenge to safety systems. Additionally, process liquid will be contaminated to some degree, requiring special measures to collect flow from the vented instrument side and also will contribute to an increase in personnel radiation exposure.

Industry experience as documented in NEDO-32977-A, indicates the ECFVs have a very low failure rate. At Susquehanna, the failure rate has been approximately 1%. Only half of these failures have resulted in replacement of the EFCV. The Susquehanna test history shows no evidence of common mode failure. This Susquehanna test experience is consistent with the findings of NEDO. The NEDO indicates similarly that many reported test failures at other plants were related to test methodologies and not actual EFCV failures. Thus, the ECFVs at Susquehanna, consistent with the industry, have exhibited a high degree of reliability, availability, and provide an acceptable level of quality and safety.

Testing on a cold shutdown frequency is impractical considering the large number of valves to be tested and the condition that reactor pressure greater than 500 psig is needed for testing. In this instance, considering the number of valves to be tested and the conditions required for testing, it is also a hardship to test all these valves during refueling outages. Recent improvements in refueling outage schedules minimized the time that is planned for refueling and testing activities during the outages.

RELIEF REQUEST 2RR01 (continued)

The appropriate time for performing excess flow check valve test is during refueling outages in conjunction with vessel hydrostatic testing. As a result of shortened outages, decay heat levels during hydrostatic tests are higher than in the past. If the hydrostatic test were extended to test all EFCVs, the vessel could require depressurization several times to avoid exceeding the maximum bulk coolant temperature limit. This is an evolution that challenges the reactor operators and thermally cycles the reactor vessel. This evolution should be avoided if possible. Also, based on past experience, excess flow check valve testing during hydrostatic testing becomes the outage critical path and could possibly extend the outage by two days if all EFCVs were to be tested during this time frame.

5. Proposed Alternate Testing

As an alternative to testing all EFCVs during the refueling outage, a sampling plan will be implemented. This plan will test certain excess flow check valves immediately preceding the refueling outage while the reactor is at power, while also instituting the appropriate conditions for testing (reactor press > 500 psig). This alternative provides an acceptable level of quality and safety. Performance of this excess flow check valve testing prior to the outage will be scheduled such that, in the event of a failure, the resulting action statement and limiting condition of operation will encompass the planned shutdown for the refueling outage. Using this strategy, unplanned, unnecessary plant shutdowns as a result of excess flow cheek valve testing will be avoided.

Functional testing with verification that flow is checked will be performed per Technical Specification 3.6.1.3.9, either immediately preceding a planned refueling outage or during the refueling outage for certain EFCVs. For those valves tested prior to the refueling outage, appropriate administrative and scheduling controls will be established.

Surveillance Requirement 3.6.1.3.9 allows a "representative sample" of EFCVs to be tested every 24 months, such that each EFCV will be tested at least once every ten years (nominal).

The EFCVs have position indication in the control room. Check valve remote position indication is excluded from Regulatory Guide 1.97 as a required parameter for evaluating containment isolation. The remote position indication will be verified in the closed direction at the same frequency as the exercise test, which will be performed at the frequency prescribed in Technical Specification Surveillance Requirement 3.6.1.3.9. After the close position test, the valve will be reset, and the remote open position indication will be verified. Although inadvertent actuation of an EFCV during operation is highly unlikely due to the spring poppet design, Susquehanna verifies the EFCVs indicate open in the control room at a frequency greater than once every two years.

In summary, considering the extremely low failure rate along with personnel and plant safety concerns to perform testing, the alternative sampling plan proposed provides an acceptable level of quality and safety.

RELIEF REQUEST 2RR01 (continued)

6. Duration of Relief Request

This proposed alternative is requested for the duration of the Fourth Ten-Year Interval Susquehanna Steam Electric Station Unit 2 IST program (June 1, 2014 through May 31, 2024). This is similar to the relief request approved for the Third Ten-Year Interval Susquehanna Steam Electric Station Unit 2 IST Program (Accession No. ML050690239).

7. Precedent

NRC Safety Evaluation for Fermi 2, Relief Request VRR-011 Relating to the Third 10-Year Interval Inservice Testing Program, Docket No. 50-341 (TAC NO. ME 2558, ME 2557 AND ME 2556).

Attachment 7 to PLA-7120

Relief Request 2RR02

RELIEF REQUEST 2RR02

Relief in accordance with 10 CFR 50.55a (a)(3)(i)

Alternative Provides Acceptable Level of Quality and Safety

1. ASME Code Component(s) Affected

Valve	System	Category	Class	App J
HV251F008	RHR SHUTDOWN COOLING SUCTION OB ISO VLV	A	1	Yes
HV251F009	RHR SHUTDOWN CLG SUCT IB ISO VLV	A	1	Yes
HV251F015A/B	RHR LOOP A/B INJECTION OB ISO VLVs	A	1	Yes
HV251F022	RHR HEAD SPRAY IB SHUTOFF	A	1	Yes
HV251F023	RHR REACTOR HEAD SPRAY FLOW CONTROL VLV	A	2	Yes
HV251F050A/B	RHR LP A&B TESTABLE CHECK VALVES	A/C	1	No
HV251F122A/B	RHR/LPCI INJECTION TESTABLE CHECK BYPASS VALVES	A	1	No
HV252F005A/B	CORE SPRAY LOOP A IB INJECTION SHUTOFF VLV	A	1	Yes
HV252F006A/B	CORE SPRAY LOOP A/B TESTABLE CKV	A/C	1	Yes
HV252F037A/B	CORE SPRAY LOOP A/B TESTABLE CKV BYPASS AOV	A	1	Yes

These valves are the Category A and A/C Pressure isolation Valves (PIVs) for Residual Heat Removal System (RHR), Low Pressure Coolant Injection (LPCI), Core Spray and Reactor Head Spray for Susquehanna Steam Electric Station (SSES) Unit 2. They provide isolation and prevent over pressurization of the low pressure piping between the Emergency Core Cooling System (ECCS) and Reactor Coolant System (RCS) boundaries.

2. Applicable Code Edition and Addenda

ASME OM Code 2004 through 2006 Addenda

RELIEF REQUEST 2RR02 (continued)

3. Applicable Code Requirement

This request applies to the pressure isolation valve (PIV) leak test frequency referenced in the following requirements:

ISTC- 3630 Leakage Rate for Other Than Containment Isolation Valves, states that 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 years."

4. Basis for Relief

Pursuant to 10 CFR 50.55a, "Codes and Standards," paragraph (a)(3)(i), relief is requested from the requirement of ASME OM Code ISTC-3630(a). The basis of the relief request is that the proposed alternative would provide an acceptable level of quality and safety.

ISTC-3630 requires that leakage rate testing for PIVs be performed at least once every 2 years. PIVs are not specifically included in the scope for performance-based testing as provided for in 10 CFR Part 50, Appendix J, Option B While the motor operated PIVs and check valve HV252F006A/B affected by this request are CIVs and tested in accordance with the 10 CFR 50 Appendix J Program. Check valve PIVs, HV251F050A/B and HV251F122A/B are not within the Appendix J scope.

The concept behind the Option B Alternative for containment isolation valves is that licensees should be allowed to adopt cost effective methods for complying with regulatory requirements. Additionally, NEI 94-01, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," describes the risk-informed basis for the extended test intervals under Option B. That justification shows that for valves which have demonstrated good performance by passing their associated leak rates tests for two consecutive 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 statement that "the risk impact associated with increasing [leak rate] test intervals is negligible (less than 0.1% of total risk)." The valves identified in this relief request are all water applications. The PIV testing is performed with water pressurized to normal plant operating pressures in accordance with ISTC-3630. This relief request is intended to provide for a performance-based scheduling of PIV tests at SSES. The reason for requesting this relief is dose reduction / ALARA. Recent historical data was used to identify that PIV testing alone each refuel outage incurs total dose of approximately 500 miliRem. Assuming all of the PIVs remain classified as good

RELIEF REQUEST 2RR02 (continued)

performers the extended test intervals would provide for a savings of approximately 1.0 Rem over the 4-year period.

NUREG 0933, “Resolution of Generic Safety Issues,” Issue 105 (Interfacing Systems LOCA at LWRs) discussed the need for PIV leak rate testing based primarily on three pre-1980 historical failures of applicable valves industry-wide. These failures all involved human errors in either operations or maintenance. None of these failures involved 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 valves ability to re-position from open to closed. For check valves, such functional testing is accomplished per ASME OM Code ISTC-3520 and ISTC-3522. Power-operated valves are routinely full stroke tested per ASME OM Code to ensure their functional capabilities. At SSES, these functional tests for motor operated PIVs are performed on a cold shutdown frequency. The functional testing of the PIV check valves is performed in accordance with ISTC-5221 “Valve Obturator Movement.” Performance of separate 2 year PIV leak rate testing does not contribute any additional assurance of functional capability; it only determines the seat tightness of the closed valves.

PIV testing is performed with water pressurized to normal plant operating pressures in accordance with ISTC-3630. The intent of this relief request is to allow for a performance-based approach to the scheduling of PIV leakage testing. It has been shown that Interfacing Systems LOCA (ISLOCA) represents a small risk impact to BWRs such as SSES.

NUREG/CR-5928, "Final Report of the NRC-sponsored ISLOCA Research Program" (ADAMS Accession No. ML072430731) evaluated the likelihood and potential severity of ISLOCA events in Boiling Water Reactors (BWR) and Pressurized Water Reactors (PWR). The BWR design used as a reference for this analysis was a BWR-4 with Mark I containment. SSES is listed as a similar plant. The BWR systems were individually analyzed and in each case the report concluded that the system was "judged to not be an important consideration with respect to ISLOCA risk." Section 4.3 of the report concluded the BWR portion of the analysis by saying "ISLOCA is not a risk concern for the BWR plant examined here."

The functional tests for PIVs are performed only at a cold shutdown frequency. Such testing is not performed online in order to prevent any possibility of an inadvertent Interfacing System Loss of Coolant Accident (ISLOCA) condition. The functional testing of the PIVs is adequate to identify any abnormal condition that might affect closure capability.

RELIEF REQUEST 2RR02 (continued)

5. Proposed Alternative Testing

SSES proposes to perform PIV testing at intervals ranging from every refuel to every third refuel. The specific interval for each valve would be a function of its performance and would be established in a manner consistent with the Containment Isolation valve (CIV) process under 10CFR50 Appendix J, Option B, program guidance. The test frequency will be established such that if any of the valves, subject to a CIV and a PIV test, fail either test, the test interval for both tests will be reduced to once every 24 months until they can be re-classified as good performers per the performance evaluation requirements of Appendix J, Option B. The test intervals for the valves with a PIV-only function will be determined in a similar manner as is done for CIV testing under Option B. The test interval may be extended upon completion of two consecutive periodic PIV tests with results within prescribed acceptance criteria. Any PIV test failure will require a return to the initial interval until good performance can again be established.

6. Duration of Relief Request

This proposed alternative is requested for the duration of the Fourth Ten-Year Interval Susquehanna Steam Electric Station Unit 2 IST program (June 1, 2014 through May 31, 2024). This is similar to relief request VRR-07 approved for James A FitzPatrick's fourth 10-year IST interval, which commenced on October 1, 2007.

7. Precedents

This relief request was approved for Fermi Power Station for the Third 120 month Interval. Letter from R. Pascarelli (US NRC) to J. Davis (Detroit Edison), "Fermi-2 Evaluation of In-Service Testing Program Relief Requests VRR-011, VRR-012, and VRR-013," dated September 28, 2010.

Attachment 8 to PLA-7120

Relief Request 2RR03

RELIEF REQUEST 2RR03

Use of Code Case OMN-17, Revision 0, on the Class 1 Main Steam Relief Valves

Relief in accordance with 10 CFR 50.55a (a)(3)(i)

Alternative Provides Acceptable Level of Quality and Safety

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

Valve	System	Category	Class
PSV241F013A	Nuclear Boiler	C	1
PSV241F013B	Nuclear Boiler	C	1
PSV241F013C	Nuclear Boiler	C	1
PSV241F013D	Nuclear Boiler	C	1
PSV241F013E	Nuclear Boiler	C	1
PSV241F013F	Nuclear Boiler	C	1
PSV241F013G	Nuclear Boiler	C	1
PSV241F013H	Nuclear Boiler	C	1
PSV241F013J	Nuclear Boiler	C	1
PSV241F013K	Nuclear Boiler	C	1
PSV241F013L	Nuclear Boiler	C	1
PSV241F013M	Nuclear Boiler	C	1
PSV241F013N	Nuclear Boiler	C	1
PSV241F013P	Nuclear Boiler	C	1
PSV241F013R	Nuclear Boiler	C	1
PSV241F013S	Nuclear Boiler	C	1

These valves are Main Steam Safety/Relief Valves. They provide overpressure protection for the reactor coolant pressure boundary to prevent unacceptable radioactive release and exposure to plant personnel.

2. **Applicable Code Edition and Addenda**

ASME OM Code 2004 Edition through 2006 Addenda

RELIEF REQUEST 2RR03 (continued)

3. **Applicable Code Requirements**

In ASME OM Code Mandatory Appendix I, "Inservice Testing of Pressure Relief Devices in Light-Water Reactor Nuclear Power Plants," Section I-1320(a), "Class 1 pressure relief valves shall be tested at least once every 5 years, starting with initial electric power generation."

The licensee proposes to extend the test interval for these valves from 5 years to 6 years (with a 6-month grace period) while still maintaining the required 24-month/20% sampling requirement.

4. **Basis for Relief**

In accordance with 10 CFR 50.55a(a)(3)(i), the licensee's relief request seeks approval of an alternative to the 5-Year Test Interval requirements of ASME OM Code, Appendix I, Section I-1320(a), for the Susquehanna Main Steam Safety/Relief Valves (MSRVs) for Unit 2. Susquehanna requests that the test interval be increased from 5 years to 72 months in accordance with ASME OM Code Case, OMN-17, "Alternative Rules for Testing ASME Class 1 Pressure Relief / Safety Valves," so that the test interval for any individual valve that is in service shall not exceed 72 months except that a 6-month grace period is allowed to coincide with refueling outages to accommodate extended shutdown periods. The basis of the relief request is that the proposed alternative would provide an acceptable level of quality and safety.

Without Code relief for 24-month fuel cycles, strict Code compliance would restrict Susquehanna's operating philosophy to not operate with weeping MSRVs as Code testing would be required to be completed within 5 years. This testing strategy does not account for any leaking valves that may need to be refurbished. Since Susquehanna's philosophy is to share spare valves between both units, (the valves that are removed from one unit are installed in the other unit's next refueling outage), this testing strategy is less than adequate. This strategy could only be accomplished if a large population of MSRVs are tested each outage or additional spare valves are purchased. More than 8 valves would need to be sent to the offsite testing facility during a refueling outage. The testing and return of these valves would have to be completed expeditiously in order to not impact the refuel outage schedule duration. For this reason, additional expenditures would be incurred to purchase and test a greater number of valves each outage. Without Code relief, the additional outage work would be contrary to the principles of ALARA and could compromise radiation safety. Because of the location of certain MSRVs in the containment, interferences exist that would require the removal of more valves and piping for those valves that must be removed for the sample testing. This results in more radiation exposure to the maintenance personnel than is desirable.

With Code relief, the 16 MSRVs per unit can be tested within 6 years to complete the Code required testing for the total population and accommodate any weeping MSRVs. The increased testing over only 2 refuel cycles would result in no additional safety

RELIEF REQUEST 2RR03 (continued)

benefit to the plant. Susquehanna has had excellent performance with MSRVs over the last 10 years. Since 1987, Susquehanna has imposed a more conservative as-left leakage criterion on the testing facility than was specified in the General Electric Specification and incorporated in the PPL Specification for testing Crosby style relief valves. The criterion imposed on the test lab is 0 ml/5 minutes (via the purchase order) compared to a GE Specification "as-left" leakage criterion of 38 ml/5 minutes.

Additionally, a review of the set point testing results (for both units) from initial operation to the present shows that the average of the set point drifts percentages is approximately -0.91%. This indicates that, in general, the MSRVs Set Pressure tends to drift slightly downward, not upward. The calculated standard deviation from the average for the data was determined to be approximately 1.68%.

Also, the testing history shows that since commercial operation, Susquehanna has had only two "as-found" set pressure test acceptance criteria failures (above +3%) of the tested valves, which required additional MSRVs to be tested.

5. <u>Proposed Alternate Testing</u>

For the fourth ten-year interval, Susquehanna proposes to remove at least 20% of the 16 Main Steam Safety/Relief Valves (MSRV) plus weeping valves detected during the previous operating cycle and any valves required to be removed to access scheduled or weeping valves up to a maximum of 8 valves during each refueling outage.

Additional valves above the Code required minimum 20% will be tested if the as-found setpoint exceeds +3%, -5% (as approved per Technical Specification Amendment No. 257) of the nameplate. The additional valves tested will be from the initial population removed that are in excess of the 20% Code required minimum. If one of these valves fail, then all the MSRVs would be removed and tested.

The proposed alternative will provide for disassembly and inspection of the MSRVs to verify parts are free from defects resulting from time-related degradation or maintenance induced wear. This maintenance will also help to reduce the potential for set point drift, and increase the reliability of these Safety Relief Valves to perform their design requirement functions. Consistent with the special maintenance requirement in Code Case OMN-17, critical components will be inspected for wear and defects.

Completion of Code testing will be accomplished over a period of 3 refuel cycles or 6 years. This approach results in maintenance and operational flexibility with the following benefits:

- Provides the ability to both test the Code required valves out of the population not yet tested and replace any weeping MSRVs.

RELIEF REQUEST 2RR03 (continued)

- Maintains relatively leak-free MSRVs, thus minimizing the necessary run time of ECCS systems that provide suppression pool cooling.
- Consistent application of ALARA principles.
- Enhances equipment reliability.
- Results in minimal impact on outage durations.

The MSRVs will be tested such that a minimum of 20% of the valves (previously untested, if they exist) are tested every 24 months, such that all the valves will be tested within 3 refuel cycles. This proposal utilizes the same maintenance and testing approach that was applied in 18-month refuel cycles. This alternative frequency will continue to provide assurance of the valve operational readiness and provides an acceptable level of quality and safety.

Additionally, any failures, either seat leakage or pressure set point, occurring at the test facility, as well as weeping MSRVs that develop during the operating cycle will be documented by the corrective action program, evaluated and dispositioned accordingly.

6. Duration of Relief Request

This proposed alternative is requested for the duration of the Fourth Ten-Year Interval Susquehanna Steam Electric Station Unit 2 IST program (June 1, 2014 through May 31, 2024). This is similar to the relief request approved for the Third Ten-Year Interval Susquehanna Steam Electric Station Unit 2 IST Program (Accession No. ML050690239).

7. Precedents

NRC Safety Evaluation for Beaver Valley Power Station, Unit Nos. 1 and 2, Relief Request VRR4, (TAC No. ME5752), Docket No. 50-412, February 7, 2012

NRC Safety Evaluation for Dresden Nuclear Power Station, Unit Nos. 2 and 3, Relief Request RV-02C, (TAC Nos. ME9865, ME9866, ME9869, ME9870, ME9871, AND ME9872), Docket Nos. 50-237 and 50-249, March 22, 2012

NRC Safety Evaluation for Oyster Creek Nuclear Generating Station, Relief Request VR-01, (TAC No. ME7617), Docket No. 50-219, February 7, 2012

NRC Safety Evaluation for Monticello Nuclear Generating Plant, Relief Request VR-04 Relating to the Fifth 10-Year Interval Inservice Testing Program, Docket No. 50-263. (TAC Nos. ME8067, ME8088, ME8089, ME8090, ME8091, ME8092, ME8093, ME8094, ME8095, AND ME8095), September 26, 2012

Attachment 9 to PLA-7120

Relief Request 2RR04

RELIEF REQUEST 2RR04

Relief in accordance with 10 CFR 50.55a (a)(3)(ii) Hardship or Unusual Difficulty Without Compensating Increase in Level of Quality or Safety

1. ASME Code Components Affected

All Pumps and Valves contained within the Inservice Testing Program scope.

2. Applicable Code Edition and Addenda

ASME OM Code 2004 Edition through 2006 Addenda, and ASME OM Code Case OMN-20 from ASME OM Code 2012 Edition

3. Applicable Code Requirement

This request applies to the frequency specifications of the ASME OM Code. The frequencies for tests given in the ASME OM Code do not include a tolerance band.

ISTA-3120(a)	"The frequency for the inservice testing shall be in accordance with the requirements of Section IST."
ISTA-3400	Frequency of Inservice Tests
ISTC-3510	Exercising Test Frequency
ISTC-3540	Manual Valves
ISTC-3630(a)	Leakage Rate for Other Than Containment Isolation Valves Test Frequency
ISTC-3700	Position Verification Testing
ISTC-5221(c)(3)	"At least one valve from each group shall be disassembled and examined at each refueling outage; all valves in each group shall be disassembled and examined at least once every 8 years."
Appendix I, I-1320	Test Frequency, Class 1 Pressure Relief Devices
Appendix I, I-1330	Test Frequency, Class 1 Nonreclosing Pressure Relief Devices
Appendix I, I-1340	Test Frequency, Class 1 Pressure Relief Valves that are used for Thermal Relief Application
Appendix I, I-1350	Test Frequency, Classes 2 and 3 Pressure Relief Valves
Appendix I, I-1360	Test Frequency, Classes 2 and 3 Nonreclosing Pressure Relief Devices
Appendix I, I-1370	Test Frequency, Classes 2 and 3 Primary Containment Vacuum Relief Valves
Appendix I, I-1380	Test Frequency, Classes 2 and 3 Vacuum Relief Valves, Except for Primary Containment Vacuum Relief Valves

RELIEF REQUEST 2RR04 (continued)

Appendix I, I-1390	Test Frequency, Classes 2 and 3 Pressure Relief Devices That Are Used for Thermal Relief Application
Appendix II, II-4000(a)(1)	Performance Improvement Activities Interval
Appendix II, II-4000(b)(1)(e)	Optimization of Condition Monitoring Activities Interval

4. **Basis for Relief**

Pursuant to 10 CFR 50.55a, "Codes and standards," paragraph (a)(3)(ii), relief is requested from the frequency specifications of the ASME OM Code. The basis of the relief request is that the Code requirement presents an undue hardship without a compensating increase in the level of quality or safety.

ASME OM Code Section IST establishes the inservice test frequency for all components within the scope of the Code. The frequencies (e.g., quarterly) have always been interpreted as "nominal" frequencies (generally as defined in the Table 3.2 of NUREG 1482, Revision 1) and Owners routinely applied the surveillance extension time period (i.e., grace period) contained in the plant Technical Specifications (TS) Surveillance Requirements (SRs). The TS typically allow for a less than or equal to 25% extension of the surveillance test interval to accommodate plant conditions that may not be suitable for conducting the surveillance (SR 3.0.2). However, regulatory issues have been raised concerning the applicability of the TS "Grace Period" to ASME OM Code required inservice test frequencies irrespective of allowances provided under TS Administrative Controls (i.e., TS 5.5.6, "Inservice Testing Program," invokes SR for various OM Code frequencies).

The lack of a tolerance band on the ASME OM Code inservice test frequency restricts operational flexibility. There may be a conflict where a surveillance test could be required (i.e., its Frequency could expire), but where it is not possible or not desired that it be performed until sometime after a plant condition or associated Limiting Condition for Operation (LCO) is within its applicability. Therefore, to avoid this conflict, the surveillance test should be performed when it can be and should be performed.

The NRC recognized this potential issue in the TS by allowing a frequency tolerance as described in TS SR 3.0.2. The lack of a similar tolerance applied to OM Code testing places an unusual hardship on the plant to adequately schedule work tasks without operational flexibility.

Thus, just as with TS required surveillance testing, some tolerance is needed to allow adjusting OM Code testing intervals to suit the plant conditions and other maintenance and testing activities. This assures operational flexibility when scheduling surveillance tests that minimize the conflicts between the need to complete the surveillance and plant conditions.

RELIEF REQUEST 2RR04 (continued)

5. **Proposed Alternative Testing**

Susquehanna proposes to use the ASME OM Code Case OMN-20, from the 2012 Edition of the ASME OM Code, as an alternative for grace period associated with Inservice Testing Requirements.

ASME OM Code establishes component test frequencies that are based either on elapsed time periods (e.g., quarterly, 2 years, etc.) or on the occurrence of plant conditions or events (e.g., cold shutdown, refueling outage, upon detection of a sample failure, following maintenance, etc.).

a. Components whose test frequencies are based on elapsed time periods shall be tested at the frequencies specified in ASME Code Section IST with a specified time period between tests as shown in the following table.

Frequency	**Specified Time Period Between Tests (all values are 'not to exceed'; no minimum periods are specified)**
Quarterly	92 days (or every 3 months)
Semiannually	184 days (or every 6 months)
Annually	366 days (or every year)
x Years	x calendar years where 'x' is a whole number of years ≥ 2

b. The specified time period between tests may be extended as follows:

   i. For periods specified as less than 2 years, the period may be extended by up to 25% for any given test. This is consistent with SES TS Section 5.5.6, "Inservice Testing Program."
   ii. Period extensions may also be applied to accelerated test frequencies (e.g., pumps in Alert Range).
   iii. For periods specified as greater than or equal to 2 years, the period may be extended by up to 6 months for any given test.

c. Components whose test frequencies are based on the occurrence of plant conditions or events (e.g., cold shutdown, refueling outage, upon detection of a

RELIEF REQUEST 2RR04 (continued)

sample failure, following maintenance, etc.) may not have their period between tests extended except as allowed by the ASME OM Code.

6. Duration of Proposed Relief Request

This proposed alternative is requested for the duration of the Fourth Ten-Year Interval Susquehanna Steam Electric Station Unit 2 IST program (June 1, 2014 through May 31, 2024).

7. Precedent

Generic relief has not been specifically granted to apply a tolerance band to the ASME OM Code required test frequencies. The NRC has previously accepted the application of TS SR 3.0.2 tolerance to selected OM Code frequencies as denoted in TS 5.5.6.

The prior NRC acceptance of the practice of applying TS tolerance to ASME OM Code required test frequencies provides equivalent precedence for accepting and approving this relief request.