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10 CFR 50.55a

December 13, 2018

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

R.E. Ginna Nuclear Power Plant
Renewed Facility Operating License No. DPR-18
NRC Docket No. 50-244

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

In accordance with 10 CFR 50.55a, "Codes and standards," paragraph (z)(2), Exelon Generating Company, LLC (Exelon) requests your review and approval of the attached relief requests associated with the Inservice Testing (IST) Program for the R. E. Ginna Nuclear Power Plant (Ginna). Exelon is requesting approval of these relief requests for the sixth ten-year IST interval which is currently scheduled to start on January 1, 2020.

We request approval of these relief requests by December 13, 2019.

There are no regulatory commitments contained within this submittal.

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

Sincerely,

David T. Gudger
Manager - Licensing & Regulatory Affairs
Exelon Generation Company, LLC

Attachment: Relief Requests Associated with the Sixth Ten-Year Interval for R. E. Ginna Nuclear Power Plant

cc: USNRC Region I, Regional Administrator
USNRC Project Manager, Ginna
USNRC Senior Resident Inspector, Ginna
A. L. Peterson, NYSERDA

ATTACHMENT

Relief Requests Associated with the Sixth Ten-Year Interval for R. E. Ginna Nuclear Power Plant

Relief Request No.	Description
PR-01	Diesel Generator Fuel Oil Transfer Pumps – Flow Rate
PR-02	Auxiliary Feedwater (AFW) Pumps – Flow Rate
GR-01	Reactor Coolant Pressure Boundary Isolation Valve – Leak Testing
VR-01	Service Water Solenoid-Operated Valves (SOVs) – Stroke Time Testing
VR-02	Pressurizer Safety Relief Valves – Position Indication

EXELON GENERATION COMPANY, LLC
IST PROGRAM – RELIEF REQUEST
R. E. Ginna Nuclear Power Plant

Proposed Alternative in Accordance with 10 CFR 50.55a(z)(2)
PR-01 - Diesel Generator Fuel Oil Transfer Pumps – Flow Rate

1. ASME Code Component(s) Affected

Component ID	Description	Code Class	Group
PDG02A	Diesel Fuel Oil Transfer Pump A	3	B
PDG02B	Diesel Fuel Oil Transfer Pump B	3	B

The diesel fuel oil transfer pumps are required to transfer fuel oil from the storage tank to the day tank. This function ensures a continuous fuel supply to support long term operation of the Diesel during accident conditions.

2. Applicable Code Edition and Addenda

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

3. Applicable Code Requirement

ISTB-3550, *Flow Rate,* states, in part, "When measuring flow rate, a rate or quantity meter shall be installed in the pump test circuit. If a meter does not indicate the flow rate directly, the record shall include the method used to reduce the data."

4. Reason for Request

Pursuant to 10 CFR 50.55a, *Codes and Standards*, paragraph (z)(2), an alternative is proposed to the pump testing requirements regarding pump flow rate in OM-2012 Code paragraph ISTB-3550. The basis of the request is that compliance with the specified requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

There are no installed instruments on the diesel fuel oil transfer system that allow a direct measurement of the flow rate when testing the diesel oil fuel transfer pumps. The pump flow rate can be calculated by measuring the change in day tank level or volume and the pump operation time required to make that change. The accuracy of this method is documented in design analysis Engineering Work Request (EWR) 4526-ME-20 (Reference 1). This method determines a flow rate for a pump that can be used to evaluate the pump's hydraulic performance.

5. Proposed Alternative and Basis for Use

Ginna's diesel fuel oil transfer pumps, PDG02A & PDG02B, are positive displacement pumps. The flow rate for these pumps is determined by measuring the indicated level change in the diesel generator fuel oil day tank during a timed pump run and converting this data into fuel oil transfer pump flow rate for both the Group B and comprehensive pump tests.

Level gauges LG-2044 ("A" Emergency Diesel Generator) and LG-2045 ("B" Emergency Diesel Generator) are utilized to measure the change in indicated level while the fuel oil transfer pump is running and restoring fuel oil day tank level. Both LG-2044 and LG-2045 (sight glasses equipped with a reference scale in inches of level) have a range of indicated level of 9 inches (2.5 inches to 11.5 inches).

The respective day tank is drained to an initial indicated level of 5.0 to 5.5 inches before initiating the fuel oil pump start. This level is logged as the initial level. The pump is then started coincident with starting the stopwatch and the system allowed to stabilize. A minimum 2-minute stabilization period is observed for the comprehensive test.

Following a total minimum run time of 5 minutes (or exceeding an indicated tank level of 11 inches), the pump is stopped coincident with stopping the stopwatch and the day tank level is read in inches to the nearest 0.25 inch. This level is logged as the final level.

The change in day tank level is determined in inches and then converted to total gallons pumped using the constant conversion factor of 24.76 gallons per inch. The constant of 24.76 gallons per inch of indicated level on the day tank sight glass was established by EWR 4526-ME-20 based on the tank's geometrical dimensions. The total gallons pumped is then divided by the total pump run time to arrive at the pump test flow rate in gallons per minute (gpm). This calculation is documented in the pump test procedures.

The test circuit for each pump is a fixed flow path from the storage tank (pump suction) to the day tank (pump discharge). Pump suction pressure is nearly constant because of the very small change in storage tank level. This change in suction pressure during pump operation is considered negligible. The normal rise in day tank level is approximately 5.5 inches, which corresponds to a quantity of approximately 136 gallons pumped during the 5 minutes of pump operation, resulting in a typical flow rate of approximately 27 gpm.

The small rise in day tank level during pump operation does not affect pump discharge pressure or flow rate. This conclusion is supported by the discussion in NUREG-1482, Revision 2, Section 5.5.2, *Use of Tank Level to Calculate Flow Rate for Positive Displacement Pumps*, where the NRC states: "Pump discharge pressure will match system pressure up to the shutoff head of the positive displacement pump. Because of the characteristics of a positive displacement pump, there should be virtually no change in pump discharge flow rate as a result of the rising tank level. Therefore, rising tank level will not have an impact on test results. By having approximately the same level in the tank at the beginning of each test, licensees can achieve repeatable results."

The accuracy of level gauges, LG-2044 and LG-2045, is determined using the 9-inch indicated range of level and the constant of 24.76 gallons per inch. This yields a total volume change of 222.84 gallons. Based on a readability uncertainty of +/-0.125 inch (0.25-inch scaling), which is equivalent to 3.10 gallons, divided by the total indicated volume of 222.84 gallons, the overall accuracy of the sight glass is ± 1.39%.

In addition, the stopwatch used to measure the time the pump is operating and pumping fuel oil is now accurate to within ± 3 seconds per 24 hours (formerly ±0.6% second per minute) for a calibrated accuracy of ±0.004% (formerly ±1.0%). Combining the accuracy of the stopwatch with the level gauge sight glass, using the square root of the sum of the squares method, results in an insignificant decrease with an overall indicated accuracy of ±1.39% (formerly ±1.71%). This overall accuracy has been improved from that which was provided in the alternative previously authorized for use during the fifth 10-year interval IST program.

Therefore, the pump flow rate can be accurately calculated by measuring the change in day tank level or volume and the pump operation time required to make that change. This method determines a flow rate for these pumps that can be used to evaluate the pump's hydraulic performance and provide reasonable assurance of pump operational readiness. Therefore, relief is requested pursuant to 10 CFR 50.55a(z)(2) based on the determination that compliance with the Code pump testing requirements regarding pump flow rate cannot be achieved without resulting in a hardship or unusual difficulty without a compensating increase in the level of quality and safety; and the proposed alternative of using the tank level change vs. time to calculate the flow rate provides reasonable assurance of operational readiness and provides an acceptable level of quality and safety.

6. Duration of Proposed Alternative

This request, upon approval, will be applied to the R. E. Ginna Nuclear Power Plant sixth 10-year IST interval, which begins on January 1, 2020, and is scheduled to end on December 31, 2029.

7. Precedent

This relief request was previously approved for the fifth 10-year interval at Ginna, as documented in NRC safety evaluation, "Alternative Requests for Fifth 10-Year Pump and Valve Inservice Testing Program – R. E. Ginna Nuclear Power Plant (TAC Nos. ME2232, ME2233, ME2234, ME2235, ME2236, ME2237, ME2238, and ME2239)," dated December 30, 2009 (ML093570173).

8. References

1. Engineering Work Request 4526-ME-20, Evaluation of Instrument Setpoints for EDG Fuel Oil System.
2. NUREG-1482, *Guidelines for Inservice Testing at Nuclear Power Plants*, Revision 2.

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Proposed Alternative in Accordance with 10 CFR 50.55a(z)(2)
PR-02 - Auxiliary Feedwater (AFW) Pumps – Flow Rate

1. ASME Code Component(s) Affected

Component ID	Description	Code Class	Group
PAF01A	"A" Preferred Motor Driven AFW Pump	3	A
PAF01B	"B" Preferred Motor Driven AFW Pump	3	A
PSF01A	"C" Standby Motor Driven AFW Pump	3	B
PSF01B	"D" Standby Motor Driven AFW Pump	3	B

The AFW pumps are required to be capable of supplying AFW flow to the steam generators during a loss of normal feedwater flow or a steam line break in conjunction with a loss of off-site power. This function maintains steam generator water level to provide a secondary heat sink for residual heat removal of the reactor coolant system.

2. Applicable Code Edition and Addenda

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

3. Applicable Code Requirement

ISTB-3550, *Flow Rate,* states, in part, "When measuring flow rate, a rate or quantity meter shall be installed in the pump test circuit. If a meter does not indicate the flow rate directly, the record shall indicate the method used to reduce the data."

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

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

ISTB-5121*(c)* states "Where it is not practical to vary system resistance, flow rate and pressure shall be determined and compared to their respective reference values."

ISTB-5122, *Group B Test Procedure*, states that "Group B tests shall be conducted with the pump operating as close as practical to a specified reference point and within the variances from the reference point as described in this paragraph. The test parameter value identified in Table ISTB-3000-1 shall be determined and recorded as required by this paragraph."

ISTB-5122*(c)* states, in part, "System resistance may be varied as necessary to achieve a point as close as practical to the reference point. If the reference point is flow rate, the variance from the reference point shall not exceed +2% or –1%."

4. Reason for Request

Pursuant to 10 CFR 50.55a, *Codes and Standards*, paragraph (z)(2), an alternative is proposed to the pump testing requirements regarding pump flow rate in the ASME OM-2012 Code. The basis of the request is that compliance with the specified requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

The AFW pumps each have a minimum flow path that can be utilized for the respective Group A and Group B pump tests. The minimum flow lines provide a fixed resistance flow path from the pump discharge to the condensate storage or demineralized water storage tank, as applicable, then back to the suction of each pump. However, the minimum flow lines are not provided with flow instrumentation.

Compliance with the Code is an undue burden due to existing design limitations in that a flow rate measuring device is not installed in the associated pump minimum flow recirculation line being employed as the pump test circuit. Costly major hardware modifications would be required to provide a permanent flow measuring device in each affected line. It has been estimated that the cost would exceed $75,000 annually to install and maintain temporary flow measuring devices or more than $375,000 to install permanent flow measuring devices into the minimum flow recirculation lines in order to meet the ASME OM Code requirements and support the quarterly testing of the four AFW pumps. Additionally, flow is not variable since an installed flow orifice establishes a 40 gpm flow rate when the pump is operated in the recirculation mode.

The flow path to the steam generators has flow instrumentation; however, this flow path has the potential for service water intrusion and requires a reactivity change. This flow path is used for the biennial comprehensive pump test.

Therefore, the instrumented flow path which has the potential for service water intrusion into the steam generators and requires a reactivity change, and the cost of installing either temporary or permanent flow instrumentation in the minimum flow recirculation lines imposes an undue burden without a compensating increase in the level of quality or safety.

5. Proposed Alternative and Basis for Use

The performance of pump tests using a fixed resistance flow path is an acceptable alternative to the Code requirements per NUREG-1482, Revision 2, Section 5.9, *Pump Testing Using Minimum Flow Return Lines With or Without Flow Measuring Devices*. During the performance of quarterly pump testing, pump differential pressure will be measured and trended. This provides a reference value for differential pressure that can be duplicated during subsequent tests. This methodology provides for the acquisition of repeatable differential pressure, which is an adequate means of monitoring for pump degradation.

Concerns identified in NRC Bulletin 88-04, *Potential Safety-Related Pump Loss*, with regard to minimum recirculation flow line sizing were assessed and verified to not be of concern during pump testing.

Quarterly testing of the designated Group A AFW centrifugal pumps (PAF01A, PAF01B) will be performed on minimum flow recirculation measuring differential pressure across the pump and measuring vibration per ASME OM-2012 Code, paragraph ISTB-5121 and using NUREG-1482, Revision 2, Section 5.9 for guidance.

Quarterly testing of the designated Group B Standby AFW centrifugal pumps (PSF01A, PSF01B) will be performed on minimum flow recirculation measuring differential pressure across the pump per ASME OM-2012 Code, paragraph ISTB-5122 and using NUREG-1482, Revision 2, Section 5.9 for guidance.

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

6. Duration of Proposed Alternative

This request, upon approval, will be applied to the R. E. Ginna Nuclear Power Plant sixth 10-year IST interval, which begins on January 1, 2020, and is scheduled to end on December 31, 2029.

7. Precedent

This relief request was previously approved for the fifth 10-year interval at Ginna, as documented in NRC safety evaluation, "Alternative Requests for Fifth 10-Year Pump and Valve Inservice Testing Program – R.E. Ginna Nuclear Power Plant (TAC Nos. ME2232, ME2233, ME2234, ME2235, ME2236, ME2237, ME2238, and ME2239)," dated December 30, 2009 (ML093570173).

8. References

1. NUREG-1482, *Guidelines for Inservice Testing at Nuclear Power Plants*, Revision 2.
2. NRC Bulletin 88-04, *Potential Safety-Related Pump Loss.*

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Proposed Alternative in Accordance with 10 CFR 50.55a(z)(2),
GR-01 - Reactor Coolant Pressure Boundary Isolation Valve – Leak Testing

1. ASME Code Component(s) Affected

Valve	System	Code Class	Category	Configuration/Type
878A	SI	2	A	MOV
878C	SI	2	A	MOV
877A	SI	1	A/C	Event V CV
877B	SI	1	A/C	Event V CV
878F	SI	1	A/C	Event V CV
878H	SI	1	A/C	Event V CV

The Reactor Coolant System (RCS) Pressure Isolation Valves (PIVs) function to provide reactor coolant system pressure boundary isolation.

2. Applicable Code Edition and Addenda

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

3. Applicable Code Requirement

ISTC-3630, *Leakage Rate for Other Than Containment Isolation Valves,* states, in part, 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 yr."

4. Reason for Request

Pursuant to 10 CFR 50.55a, *Codes and Standards*, paragraph (z)(2), an alternative to the requirement of ASME OM-2012 Code, paragraph ISTC-3630(a) is proposed. The basis of the request is that compliance with the specified requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

Safety injection (SI) hot leg check valves 877A, 877B, 878F, and 878H and motor operated valves (MOVs) 878A and 878C are considered to be passive. During operation, the check valves are normally closed and their associated MOV is also closed and de-energized.

Leakage testing for these valves, including testing requirements, is governed by plant Technical Specification (TS) 3.4.14, *RCS Pressure Isolation Valve (PIV) Leakage.* TS Surveillance Requirement (SR) 3.4.14.2 requires that Ginna verify leakage from each SI system hot leg injection line RCS PIV at a prescribed differential pressure. The seat leakage is measured, analyzed, and compared to permissible leakage rates at a frequency prescribed by the Surveillance Frequency Control Program (SFCP), which is 40 months for SR 3.4.14.2.

Due to the lack of test connections, each series pair of check valves (877A/878F and 877B/878H) form one of the two pressure boundaries required to be tested with the second boundary being its associated MOV. Failure of a leakage test of a tested pair would require that both check valves be declared inoperable and in need of rework. Any valve failing the acceptance criteria of TS 3.4.14 shall be declared inoperable and entered into a TS Action in TS Section 3.4.14. Testing of series pairs of check valves in this configuration is allowed by the OM Code, paragraph ISTC-5223, *Series Valves in Pairs*, and utilizes the guidance found in NUREG-1482, Revision 2, Section 4.1.1, *Closure Verification for Series Check Valves without Intermediate Test Connections*, which states that testing of the pair of valves is acceptable if the configuration does not require two valves and the safety analysis for such a configuration would credit either of the two valves.

Since the series pairs of check valves 877A/878F and 877B/878H do not have the needed test connections to individually test each valve and since testing of these valves with their adjacent MOVs is specified adequately by TS, it is an undue burden to comply with the OM Code requirements to perform separate leak rate tests. The plant TS establish the maximum permissible leakage rates, test pressure requirements, test frequency requirements, and the required action if the leak rate limit is exceeded. To make modifications to include the proper test connections and perform leak rate testing in accordance with the OM Code would be costly and increase personnel radiation exposure and would not result in a compensating increase in the level of quality and safety.

5. Proposed Alternative and Basis for Use

In lieu of the Code-required separate leak rate tests, these series pair check valves will be leak rate tested in accordance with the RCS PIV leak rate testing per TS 3.4.14. The proposed alternative testing will provide reasonable assurance of the valves' operational readiness. Therefore, this alternative to the Code required leakage rate testing of the RCS PIVs is proposed pursuant to 10 CFR 50.55a(z)(2).

6. Duration of Proposed Alternative

This request, upon approval, will be applied to the R. E. Ginna Nuclear Power Plant sixth 10-year IST interval, which begins on January 1, 2020, and is scheduled to end on December 31, 2029.

7. Precedent

This relief request was previously approved for the fifth 10-year interval at Ginna, as documented in NRC safety evaluation, "Alternative Request GR-01 for the Fifth 10-Year Pump and Valve Inservice Testing Program – R. E. Ginna Nuclear Power Plant (TAC No. ME2238)," dated April 14, 2010 (ML100890237).

8. **References**

1. NUREG-1482, *Guidelines for Inservice Testing at Nuclear Power Plants*, Revision 2.
2. Technical Specifications 3.4.14, *RCS Pressure Isolation Valve (PIV) Leakage*, and associated TS Surveillance Requirement SR-3.4.14.2.
3. Ginna Surveillance Frequency Control Program

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Proposed Alternative in Accordance with 10 CFR 50.55a(z)(2),
VR-01 - Service Water Solenoid-Operated Valves (SOVs) – Stroke Time Testing

1. ASME Code Component(s) Affected

Valve	Description	Code Class	Category
4324	TDAFW Pump SW Strainer Bypass SOV	3	B
4325	MDAFW Pump A SW Strainer Bypass SOV	3	B
4326	MDAFW Pump B SW Strainer Bypass SOV	3	B

These service water (SW) valves open upon an auxiliary feedwater (AFW) pump bearing cooling water supply high strainer differential pressure (DP) to provide cooling water to the driver's bearings.

2. Applicable Code Edition and Addenda

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

3. Applicable Code Requirement

ISTC-5150, *Solenoid-Operated Valves*, paragraph ISTC-5151, *Valve Stroke Testing*, states:
"*(a)* Active valves shall have their stroke times measured when exercised in accordance with para. ISTC-3500.
(b) The limiting value(s) of full-stroke time of each valve shall be specified by the Owner.
(c) Stroke time shall be measured to at least the nearest second.
(d) Any abnormality or erratic action shall be recorded (see para. ISTC-9120), and an evaluation shall be made regarding need for corrective action."

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

4. Reason for Request

Pursuant to 10 CFR 50.55a, *Codes and Standards*, paragraph (z)(2), an alternative to the requirements of ASME OM-2012 Code, paragraphs ISTC-5151 and ISTC-5152 is proposed. The basis of the request is that compliance with the specified requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

These SOVs are normally closed rapid acting valves that automatically actuate to the open position on high differential pressure across the supply strainer. Measurement of stroke times during manual actuation using conventional methods cannot be performed to produce consistent, meaningful or trendable test results. The valves are not provided with control switches to allow for conventional stroke timing methodology. Additionally, there is no remote valve position indication or other positive means to determine valve disc position. Without concise methods of initiating valve movement or determining when the stroke is completed, it is difficult to obtain repeatable stroke time data to monitor for degradation. It would be necessary to disassemble the respective differential pressure switch in order to

control actuation of these valves and as a result of this disassembly, stroke timing during power operation would require rendering these valves inoperable and entering a limiting condition for operation (LCO) from which prompt restoration would not be possible.

These valves are tested on a quarterly frequency during AFW pump testing. This testing includes strainer cleaning, strainer isolation, high differential pressure simulation, verification of valve operation, and flow observation. Failure of these valves to stroke in conjunction with a clogged strainer would result in a lack of pressure at the bearing cooler inlet and a high DP alarm, at which time an Operator would be dispatched to manually trip the respective valve.

This quarterly verification, while not measuring stroke time or monitoring for valve degradation, does provide an indication that each SOV is moving to its safety position by verifying disc movement and is consistent with the guidelines provided in NUREG-1482, Revision 2, Section 4.2.3, *Stroke Time for Solenoid-Operated Valves.*

Therefore, relief is requested pursuant to 10 CFR 50.55a(z)(2) based on the determination that compliance with the Code SOV testing requirements regarding stroke timing cannot be achieved without resulting in a hardship or unusual difficulty without a compensating increase in the level of quality and safety; and the proposed alternative testing including strainer cleaning, strainer isolation, high differential pressure simulation, verification of valve operation, and flow observation provides reasonable assurance of operational readiness and provides an acceptable level of quality and safety.

5. Proposed Alternative and Basis for Use

These valves will be stroke tested during associated AFW pump testing by closing the valve downstream of the strainer. Acceptable valve operation will be based on:

- Verifying locally that the valve has de-energized and tripped open.
- Verifying the presence of a steady stream of water from the affected floor drain funnel.
- Verifying that the associated main control board annunciator alarms.

The proposed alternative testing will accurately reflect obturator position and will provide reasonable assurance of the valves' operational readiness. Thus, this alternative to the requirements of the Code-required stroke time testing of the SW SOVs is proposed pursuant to 10 CFR 50.55a(z)(2).

6. Duration of Proposed Alternative

This request, upon approval, will be applied to the R. E. Ginna Nuclear Power Plant sixth 10-year interval, which begins on January 1, 2020, and is scheduled to end on December 31, 2029.

7. Precedent

This relief request was previously approved for the fifth 10-year interval at Ginna, as documented in NRC safety evaluation, "Alternative Requests for Fifth 10-Year Pump and Valve Inservice Testing Program – R. E. Ginna Nuclear Power Plant (TAC Nos. ME2232, ME2233, ME2234, ME2235, ME2236, ME2237, ME2238, and ME2239)," dated December 30, 2009 (ML093570173).

8. Reference

1. NUREG-1482, *Guidelines for Inservice Testing at Nuclear Power Plants*, Revision 2.

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Proposed Alternative in Accordance with 10 CFR 50.55a(z)(2),
VR-02 - Pressurizer Safety Relief Valves – Position Indication

1. ASME Code Component(s) Affected

Component ID	Description	Code Class	Category
434	Pressurizer Relief Valve	1	C
435	Pressurizer Relief Valve	1	C

The Pressurizer Safety Relief Valves provide over-pressurization protection for the Reactor Coolant System (RCS)/Pressurizer.

2. Applicable Code Edition and Addenda

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

3. Applicable Code Requirement

Mandatory Appendix I, paragraph I-7310, *Class 1 Safety Valves,* states, in part, "Tests before maintenance or set-pressure adjustment, or both, shall be performed for subparas. I-7310(a) through (c) in sequence. The remaining shall be performed after maintenance or set-pressure adjustment."

Subparagraph I-7310(f) states, "determination of operation and electrical characteristics of position indicators."

4. Reason for Request

Pursuant to 10 CFR 50.55a, *Codes and Standards*, paragraph (z)(2), an alternative to the requirement of ASME OM Code Mandatory Appendix I, subparagraph I-7310(f) is proposed. The basis of the request is that compliance with the specified requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

These valves are mechanical spring-actuated valves with an externally mounted Linear Voltage Differential Transformer (LVDT) stem position indicator. The position indicator must be removed in order to permit removal of the safety valves each refueling outage for shipment to an off-site vendor for set pressure testing. It would be necessary to intentionally challenge RCS pressure limits to actuate these safety valves in order to perform position indication testing prior to removal for set pressure testing. Also, if these safety valves were actuated for a position indication test following re-installation, they would again need to be retested to ensure the set pressure has not been adversely affected. This involves increased testing and unnecessary radiation exposure to test personnel and results in a hardship without a compensating increase in the level of quality and safety.

5. Proposed Alternative and Basis for Use

In accordance with plant administrative procedures, channel checks for Pressurizer safety relief valve position indication are performed once per shift and validated by comparison with tailpipe temperature indication. The valves are also simulated to actuate using station calibration procedures. The procedure utilizes movement of the valve's coil (up/down) and verifies position via an alarm in the Control Room. Calibration of these position indicators is governed by plant calibration procedures and is performed on a refueling outage frequency. These procedures verify that the proper clearance is obtained to ensure obturator position is accurately represented and provide reasonable assurance of valve operational readiness. Thus, this alternative to the Code-required testing of the pressurizer safety relief valves is proposed pursuant to 10 CFR 50.55a(z)(2).

6. Duration of Proposed Alternative

This request, upon approval, will be applied to the R. E. Ginna Nuclear Power Plant sixth 10-year IST interval, which begins on January 1, 2020, and is scheduled to end on December 31, 2029.

7. Precedent

This relief request was previously approved for the fifth 10-year interval at Ginna, as documented in NRC safety evaluation, "Alternative Requests for Fifth 10-Year Pump and Valve Inservice Testing Program – R.E. Ginna Nuclear Power Plant (TAC Nos. ME2232, ME2233, ME2234, ME2235, ME2236, ME2237, ME2238, and ME2239)," dated December 30, 2009 (ML093570173).