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November 18, 2008
L-08-347

10 CFR 50.55a

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

SUBJECT:
Perry Nuclear Power Plant
Docket No. 50-440, License No. NPF-58
Eight Separate In-Service Testing Program 10 CFR 50.55a Requests in Support of the Third Ten-Year Interval

In accordance with 10 CFR 50.55a, Nuclear Regulatory Commission (NRC) review and approval is requested for eight separate proposed alternatives to certain requirements associated with the In-Service Testing Program (ISTP) for the Perry Nuclear Power Plant. Enclosures A through H identify the affected components, the applicable code requirements, the reason for the requests, the proposed alternatives and basis for use, and the duration for each of the 10 CFR 50.55a requests.

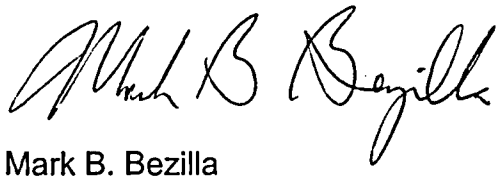
The alternatives are proposed for use during the third ten-year ISTP interval, which begins May 18, 2009. Therefore, FENOC requests approval of the proposed 10 CFR 50.55a requests by May 17, 2009, to coincide with the beginning of the third ten-year ISTP interval.

There are no regulatory commitments contained in this letter. If there are any questions or additional information is required, please contact Mr. Thomas A. Lentz, Manager – Fleet Licensing, at (330) 761-6071.

A047
NRR

I declare under penalty of perjury that the foregoing is true and correct. Executed on November 18, 2008.

Sincerely,



Mark B. Bezilla

Enclosures:

A. 10 CFR 50.55a Request Number PR-1
B. 10 CFR 50.55a Request Number PR-2
C. 10 CFR 50.55a Request Number VR-1
D. 10 CFR 50.55a Request Number VR-2
E. 10 CFR 50.55a Request Number VR-3
F. 10 CFR 50.55a Request Number VR-4
G. 10 CFR 50.55a Request Number VR-5
H. 10 CFR 50.55a Request Number VR-6

cc: NRC Region III Administrator
NRC Resident Inspector
NRR Project Manager

Proposed Alternative
in Accordance with 10 CFR 50.55a(a)(3)(i)

--Alternative Provides Acceptable Level of Quality and Safety--

## 1. ASME Code Component(s) Affected

1E12-C003, RHR Waterleg Pump (Class 2)
1E21-C002, LPCS Waterleg Pump (Class 2)
1E22-C003, HPCS Waterleg Pump (Class 2)
1E51-C003, RCIC Waterleg Pump (Class 2)

Waterleg pumps maintain the discharge piping of safety-related systems full to expedite flow during initiation, and minimize the likelihood of system damage due to water hammer.

## 2. Applicable Code Edition and Addenda

ASME OM Code-2001, with Addenda through OMb-2003

## 3. Applicable Code Requirements

ISTB-3400; Frequency of Inservice Tests. An inservice test shall be run on each pump as specified in Table ISTB-3400-1. Table ISTB-03400-1 specifies that a Group A pump test shall be performed on a quarterly frequency.

ISTB-3300(e)(2); Reference values shall be established within ± 20% of pump design flow for a Group A test, if practicable. If not practicable, the reference point flow rate shall be established at the highest practical flow rate.

## 4. Reason for Request

The waterleg pumps are designed to remain in service during operation at power to ensure the emergency standby systems are maintained pressurized to reduce the likelihood of water hammer. The waterleg pumps run continuously, with flow established through a recirculation line, in order to provide enough head to keep the applicable systems discharge piping full to the highest elevation. During safety-related pump testing, the waterleg pump normal discharge path must be redirected through drain lines to provide

enough flow to establish the selected code reference values. This requires taking the system out of service and racking out safety-related pump breakers for the Residual Heat Removal (RHR) system, the Low Pressure Core Spray (LPCS) system, and the High Pressure Core Spray (HPCS) system, or isolating the Reactor Core Isolation Cooling (RCIC) system pump to prevent system damage due to water hammer or cavitation upon receipt of an auto actuation signal.

Quarterly full flow testing of the listed safety-related waterleg pumps would result in the inoperability of its associated Emergency Core Cooling System without a compensating increase in the level of quality or safety.

## 5. Proposed Alternative and Basis for Use

The waterleg pumps shall be monitored on a quarterly basis by observing pump discharge pressure and bearing vibration. These parameters will be evaluated to adequately assess the pump's performance. The pumps will be full flow tested each refueling outage in conjunction with the comprehensive pump test performed in accordance with the requirements specified in ISTB-5123 Comprehensive Test Procedure.

All of these pumps have adequate margin beyond the capacity required for them to fulfill their function. Using the provisions of this relief request as an alternative to the requirements of ISTB-3400 and ISTB-3300(e)(2) provides a reasonable alternative to the code requirements. The proposed alternative provides an acceptable level of quality and safety for monitoring the pumps and assuring that the pumps are capable of performing their safety function.

## 6. Duration of Proposed Alternative

The proposed alternative identified in this relief request shall be utilized during the Third Ten-Year IST Interval.

## 7. Precedent

Perry Nuclear Power Plant, Docket No. 50-440, Safety Evaluation Report (SER) dated August 9, 1999, "Safety Evaluation of the Inservice Testing Program Second Ten-Year Interval for Pumps and Valves – Perry Nuclear Power Plant, (TAC No. MA3328)." Previously approved as PR-2 in the aforementioned SER. Refer to Attachment 2, Technical Evaluation Report, Section 2.2.

## 8. Reference

Generic Letter 89-04, "Guidance on Developing Acceptable Inservice Testing Programs," Attachment 1, Position 9, "Pump Testing using Minimum-flow Return Line With or Without Flow Measuring Devices."

Proposed Alternative
in Accordance with 10 CFR 50.55a(a)(3)(i)

--Alternative Provides Acceptable Level of Quality and Safety--

## 1. ASME Code Component(s) Affected

1C41-C001A & B, Standby Liquid Control Pumps (Class 2)
1E12-C002A, B & C, Residual Heat Removal Pumps (RHR) (Class 2)
1E12-C003, RHR Waterleg Pump (Class 2)
1E21-C001, Low Pressure Core Spray Pump (LPCS) (Class 2)
1E21-C002, LPCS Waterleg Pump (Class 2)
1E22-C001, High Pressure Core Spray Pump (HPCS) (Class 2)
1E22-C003, HPCS Waterleg Pump (Class 2)
1E51-C001, Reactor Core Isolation Cooling Pump (RCIC) (Class 2)
1E51-C003, RCIC Waterleg Pump (Class 2)
G41-C003A & B, Fuel Pool Cooling and Cleanup Pumps (Class 3)
1P42-C001A & B, Emergency Closed Cooling Pumps (Class 3)
1P45-C001A & B and C002, Emergency Service Water Pumps (Class 3)
P47-C001A & B, Control Complex Chilled Water Pumps (Class 3)
1R45-C001A, B, C & C002A, B, C, Standby Diesel Generator Fuel Oil Pumps (Class 3)

## 2. Applicable Code Edition and Addenda

ASME OM Code-2001, with Addenda through OMb-2003

## 3. Applicable Code Requirements

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

ISTB-3510(b)(2); Digital Instruments shall be selected such that the reference value does not exceed 70% of the calibrated range of the instrument.

## 4. Reason for Request

The ASME Code committees have approved Code Case OMN-6, "Alternative Rules for Digital Instruments," which was included in the OMa-1999. The Nuclear Regulatory Commission has unconditionally approved the use of this code case as reflected in Regulatory Guide 1.192, "Operation and Maintenance Code Case Acceptability, ASME OM Code," June 2003. This code case allows owners to use digital instruments such that the reference value does not exceed 90% of the calibrated range of the instrument. The Code of record for the Third Ten-Year Inservice Testing (IST) interval is OM Code-2001 Edition with Addenda through OMb-2003. As stated in Regulatory Guide (RG) 1.192, the applicable Code for Code Case OMN-6 is OMa-1999. Perry Nuclear Power Plant (PNPP) is requesting the use of Code Case OMN-6 because the applicable edition of the code is other than the code edition/addenda specified as the code of record for the Third Ten-Year Interval.

## 5. Proposed Alternative and Basis for Use

In lieu of the digital instruments requirements specified in ISTB-3510(b)(2), PNPP proposes to utilize the alternative rules for digital instruments specified in Code Case OMN-6. Whereas, digital instruments shall be selected such that the reference value does not exceed 90% the calibrated range of the instrument.

Using the provisions of this relief request as an alternative to the requirements of ISTB-3510(b)(2) provides a reasonable alternative to the code requirements based on the determination that the proposed alternative provides an acceptable level of quality and safety as recognized by RG 1.192.

## 6. Duration of Proposed Alternative

The proposed alternative identified in this relief request shall be utilized during the Third Ten-Year IST Interval.

## 7. Precedent

Perry Nuclear Power Plant, Docket No. 50-440, Safety Evaluation Report (SER) dated March 31, 1999, Safety Evaluation of the Inservice Testing Program Relief Requests for the Second Ten-Year Interval – Perry Nuclear Power Plant, (TAC No. MA3328). Previously approved as PR-6 in the aforementioned SER.

Regulatory Guide 1.192, "Operation and Maintenance Code Case Acceptability, ASME OM Code", June 2003, Table 1, "Acceptable OM Code Cases."

## 8. Reference

Code Case OMN-6, Alternative Rules for Digital Instruments.

Proposed Alternative
in Accordance with 10 CFR 50.55a(a)(3)(i)

--Alternative Provides Acceptable Level of Quality and Safety--

## 1. ASME Code Component(s) Affected

**Category B Valves** (Typical of 177)

1C11-126, Scram Inlet Valve (Class 2)
1C11-127, Scram Exhaust Valve (Class 2)

**Category C Valves** (Typical of 177)

1C11-114, Scram Discharge Header Check Valve (Class 2)
1C11-115, Charging Water Check Valve (Class 2)

These valves operate as an integral part of their respective hydraulic control unit to rapidly insert the control rods in support of a scram function.

## 2. Applicable Code Edition and Addenda

ASME OM Code-2001, with Addenda through OMb-2003

## 3. Applicable Code Requirements

ISTC-3510, "Exercise Test Frequency," requires active Category B and Category C check valves to be exercised nominally every 3 months. If exercising every 3 months is not possible then exercising may be performed during cold shutdowns or refueling outages as permitted by ISTC-3520.

ISTC-5130, "Pneumatically Operated Valves," requires active valves to have their stroke times measured when exercised in accordance with ISTC-3500.

ISTC-5220, "Check Valves," requires that necessary valve obturator to be demonstrated by performing both an open and close test.

## 4. Reason for Request

These valves are not provided with position indication; therefore measuring their full stroke time in accordance with the code is impractical.

The charging water check valve (1C11-115) can only be verified closed by securing the Control Rod Drive (CRD) pumps and depressurizing the charging water header. Securing the CRD pumps would result in a loss of cooling water to the reactor recirculation pumps and all the CRD mechanisms, which would be impractical due to the potential for equipment damage or reactor scram.

Exercising these valves at a frequency other than that specified by Technical Specifications could result in a plant trip, which is burdensome without a compensating increase in the level of quality and safety. Additionally, since the power operated valves are not provided with position indication, special test methods or test equipment would be required to determine valve position, which is also burdensome without a compensating increase in the level of quality and safety.

## 5. Proposed Alternative and Basis for Use

As discussed in NUREG-1482, Rev.1, Section 4.4.6, the rod scram test frequency identified in the plant Technical Specifications may be used as the valve testing frequency to minimize rapid reactivity transients and unnecessary wear of the CRD mechanisms. Verifying that the associated control rod meets the scram insertion time limits defined in the Technical Specifications can be an acceptable alternative method of detecting degradation of these valves in lieu of valve stroke measurement.

Technical Specification Surveillance Requirement (SR) 3.1.4.1 requires the scram time for all control rods to be verified within limits prior to thermal power exceeding 40% of rated thermal power after fuel movement, and prior to thermal power exceeding 40% of rated thermal power after each reactor shutdown ≥120 days. In addition, Technical Specification SR 3.1.4.2 requires testing of a representative sample of the control rods at least once per 120 days of operation in Mode 1. The Technical Specification SRs assure the necessary quality of the system and components are maintained, and that facility operation will be within the Safety Limits and the Limiting Condition of Operation will be met. Therefore, scram insertion timing per Technical Specification SR 3.1.4.1 shall be substituted for individual valve testing.

Using the provisions of this relief request as an alternative to the requirements of ISTC-3510, 5130 and 5220 provides a reasonable alternative to the code requirements. The proposed alternative method of detecting degradation provides reasonable assurance of the valves' operational

readiness. Therefore, the proposed alternative provides an acceptable level of quality and safety, and Perry Nuclear Power Plant (PNPP) requests that relief be granted pursuant to 10 CFR 50.55a(a)(3)(i).

## 6. Duration of Proposed Alternative

The proposed alternative identified in this relief request shall be utilized during the Third Ten-Year IST Interval.

## 7. Precedent

Perry Nuclear Power Plant, Docket No. 50-440, Safety Evaluation Report (SER) dated August 9, 1999, "Safety Evaluation of the Inservice Testing Program Second Ten-Year Interval for Pumps and Valves – Perry Nuclear Power Plant, (TAC No. MA3328)." Previously approved as VR-1 in the aforementioned SER.

## 8. References

1. Technical Specification SR 3.1.4.1, Control Rod Scram Times.

2. Technical Specification SR 3.1.4.2, Control Rod Scram Times.

3. NUREG-1482, Guidelines for Inservice Testing at Nuclear Power Plants, Revision 1, January 2005, Section 4.4.6, Testing Individual Scram Valves for Control Rods in Boiling-Water Reactors.

Proposed Alternative
in Accordance with 10 CFR 50.55a(a)(3)(i)

--Alternative Provides Acceptable Level of Quality and Safety--

## 1. ASME Code Component(s) Affected

1B21-F041A, Dikkers Valve (ADS)
1B21-F041B, Dikkers Valve (ADS)
1B21-F041C, Dikkers Valve
1B21-F041D, Dikkers Valve
1B21-F041E, Dikkers Valve (ADS)
1B21-F041F, Dikkers Valve (ADS)
1B21-F041G, Dikkers Valve
1B21-F041K, Dikkers Valve
1B21-F047B, Dikkers Valve
1B21-F047C, Dikkers Valve
1B21-F047D, Dikkers Valve (ADS)
1B21-F047F, Dikkers Valve (LLS)
1B21-F047G, Dikkers Valve
1B21-F047H, Dikkers Valve (ADS)
1B21-F051A, Dikkers Valve (LLS)
1B21-F051B, Dikkers Valve (LLS)
1B21-F051C, Dikkers Valve (ADS/LLS)
1B21-F051D, Dikkers Valve (LLS)
1B21-F051G, Dikkers Valve (ADS/LLS)

The Nuclear Boiler System provides Reactor Pressure Vessel (RPV) over pressurization protection by opening the Safety/Relief Valves (SRVs). The SRVs open at the high reactor pressure trip set point. In addition to overpressure protection, the SRVs provide RPV pressure relief by opening to release steam and decrease vessel pressure. Pressure in the vessel is thereby maintained below the American Society of Mechanical Engineers (ASME) Code required limit.

In addition to the above, the Automatic Depressurization System (ADS) and the individual SRVs shall be capable of being manually operated from the main Control Room. This provides the capability to manually depressurize the RPV in the event of the main condenser is not available as a heat sink.

The Nuclear Boiler System ADS shall provide automatic depressurization of the RPV under certain small break Loss Of Coolant Accident (LOCA) conditions so that the low pressure Emergency Core Cooling Systems (ECCS) can adequately cool the core. Note that all of the SRVs, those used for ADS as well as those assigned purely for pressure relief, are used for overpressure protection. All of the SRVs work together to ensure that the ASME Code limit is not exceeded.

2. **Applicable Code Edition and Addenda**

ASME OM Code-2001, with Addenda through OMb-2003

3. **Applicable Code Requirements**

ASME OM Code, Appendix I (1995), Section 3410(d), requires that each valve that has been maintained or refurbished in place, removed for maintenance and testing, or both, and reinstalled, shall be remotely actuated at reduced or normal system pressure to verify open and close capability of the valve before resumption of electric power generation. Set-pressure verification is not required.

4. **Reason for Request**

The nuclear industry experience as a whole has shown that repeated manual actuation of the SRVs and ADS valves can lead to valve seat leakage during plant operation. This experience is substantiated within NUREG-0626, "Generic Evaluation of Feedwater Transients and Small Break Loss-of-Coolant Accidents in GE-Designed Operating Plants and Near Term Operating License Applications," and NUREG-0123, "Standard Technical Specifications for GE Boiling Water Reactors (BWR/5)," which recommend reducing the number of challenges to the ADS valves.

5. **Proposed Alternative and Basis for Use**

This relief request will allow testing of the SRVs to be performed in two separate stages. Stage 1 will be manual actuation of the valves at the qualified test facility. This will verify the open and close function of the valve with the actuator coupled to the valve stem, and includes both solenoids and the air block valve. Each solenoid is energized, one at a time, resulting in two separate lifts of the SRV disk from the seat. Stage 2 will be manual actuation of the SRV actuators following installation into the plant with the actuator uncoupled from the valve stem. The plant installed testing will verify full operation of the electrical circuitry, manual actuation solenoid valve, block valve, and the actuator. Therefore, all components associated with the SRVs will continue to be tested.

This uncoupled test may also be performed following any maintenance activity that could affect the relief mode of the associated SRV.

With this relief request the existing test method will also remain acceptable, i.e., full stroke exercise from the control room at adequate reactor steam pressure and flow.

The proposed test alternative provides verification of proper control connections by requiring the pneumatic and electrical controls to cycle the actuator on each SRV following installation, without stroking the SRV disk. The plant installed testing will verify full operation of the electrical circuitry, manual actuation solenoid valve, block valve, and the actuator. In addition, the test populations of SRVs removed each refuel outage for setpoint testing would also be tested in the relief mode during bench testing. This setpoint testing provides assurance that the SRV would perform as expected when control air pressure is applied to the actuator assembly.

The proposed test alternative continues to demonstrate full functionality of the SRVs while minimizing the potential for creating valve seat leakage caused by cycling the valve unnecessarily. Therefore, the proposed test alternative provides an acceptable level of quality and safety. Manual actuation of the valves at the qualified test facility will verify the open and close function of the valve with the actuator coupled to the valve stem. This actuation includes both solenoids and the air block valve, with each solenoid being energized, one at a time, and results in two separate lifts of the SRV disk from the seat.

Upon re-installation, uncoupled manual actuation will verify the appropriate function of the electric circuit, both solenoid valves, air block valve, and the valve actuator. This actuation includes both solenoids by lifting of the actuator with the first solenoid and maintaining the actuator open using the second solenoid, thereby, only lifting the actuator once.

Using the provisions of this relief request as an alternative to ASME OM Code, Appendix I (1995), Section 3410(d), provides a reasonable alternative to the Code requirements, based on the determination that the proposed alternative provides an acceptable level of quality and safety. In addition, the method of uncoupled exercising is recognized as acceptable per OM Code-2004, I-3410(d) whereas main disk movement is not required subsequent to installation after maintenance.

## 6. Duration of Proposed Alternative

The proposed alternative identified in this relief request shall be utilized during the Third Ten-Year IST Interval.

## 7. Precedents

Perry Nuclear Power Plant, Docket No. 50-440, Safety Evaluation Report (SER) dated February 10, 2005, "Valve Relief Request VR-13, RE: Testing of Safety/Relief Valves (TAC NO. MC2518)." Previously approved as VR-13 in the aforementioned SER.

## 8. References

1. NUREG-0626, "Generic Evaluation of Feedwater Transients and Small Break Loss-of-Coolant Accidents in GE-Designed Operating Plants and Near Term Operating License Applications."

2. NUREG-0123, "Standard Technical Specifications for GE Boiling Water Reactors (BWR/5)."

3. OM Code-2004, Appendix I, Paragraph I-3410(d).

Proposed Alternative
in Accordance with 10 CFR 50.55a(a)(3)(i)

--Alternative Provides Acceptable Level of Quality and Safety--

## 1. ASME Code Component(s) Affected

Certain motor-operated valve assemblies currently included in the Perry Nuclear Power Plant (PNPP) Motor-Operated Valve (MOV) Program

## 2. Applicable Code Edition and Addenda

ASME OM Code-2001, with Addenda through OMb-2003

## 3. Applicable Code Requirements

ISTA-3130, "Application of Codes Cases," ISTA-3130(b) states that code cases shall be applicable to the edition and addenda specified in the test plan.

ISTC-5120, "Motor-Operated Valves," ISTC-5121(a) states that active valves shall have their stroke times measured when exercised in accordance with ISTC-3500.

ISTC-3700, "Position Verification Testing," states in part, that valves with remote position indicators shall be observed locally at least once every two (2) years to verify that valve position is accurately indicated.

## 4. Reason for Request

Code Case OMN-1 has been determined by the NRC to provide an acceptable level of quality and safety when implemented in conjunction with the conditions imposed in RG 1.192.

Since the NRC staff recommends licensees implement ASME Code Case OMN-1, PNPP proposes to implement Code Case OMN-1, Revision 1 in lieu of the stroke-time provisions specified in ISTC-5120 for MOVs as well as the position verification testing in ISTC-3700.

## 5. Proposed Alternative and Basis for Use

NUREG-1482, Revision 1, "Alternatives to Stroke-Time Testing," Section 4.2.5 states in part, that as an alternative to MOV stroke-time testing, ASME developed Code Case OMN-1, "Alternative Rules for Preservice and Inservice Testing of Certain Electric Motor-Operated Valve Assemblies in LWR Power Plants," which provides periodic exercising and diagnostic testing for use in assessing the operational readiness of MOVs. Section 4.2.5 further states that the NRC staff recommends licensees implement ASME Code Case OMN-1, as accepted by the NRC (with certain conditions) in the regulations, or Regulatory Guide (RG) 1.192, Revision 0, "Operation and Maintenance Code Case Acceptability, ASME OM Code," as alternatives to the stroke-time testing provisions in the ASME Code for MOV.

RG 1.192 allows licensees with an applicable code of record to implement ASME Code Case OMN-1 (in accordance with the provisions in the regulatory guide) as an alternative to the code provisions for MOV stroke-time testing, without submitting request for relief from their code of record. The code of record for PNPP Third 10-Year IST Interval is OM Code-2001 Edition with Addenda through OMb-2003 and the applicable Code for OMN-1, as stated in RG 1.192, is OMa-1999.

Using the provisions of this relief request as an alternative to the MOV stroke-time testing requirements of ISTC-5120 and position indication verification of ISTC-3700 provides an acceptable level of quality for the determination of valve operational readiness. Code Case OMN-1, Revision 1 should be considered acceptable for use with OM Code-2001 with OMb-2003 Addenda as the code of record.

## 6. Duration of Proposed Alternative

The proposed alternative identified in this relief request shall be utilized during the Third Ten-Year IST Interval.

## 7. Precedents

1. NUREG-1482, Revision 1, Section 4.2.5, "Alternatives to Stroke-Time Testing."
2. Regulatory Guide 1.192, Revision 0, "Operation and Maintenance Code Case Acceptability, ASME OM Code," Table 2, "Conditionally Acceptable OM Code Cases."

## 8. Reference

Code Case OMN-1, "Alternative Rules for Preservice and Inservice Testing of Certain Electric Motor-Operated Valve Assemblies in LWR Power Plants."

Proposed Alternative
in Accordance with 10 CFR 50.55a(a)(3)(i)

--Alternative Provides Acceptable Level of Quality and Safety--

## 1. ASME Code Component(s) Affected

1B21-F024A,B,C,D - Inboard Main Steam Isolation Valve (MSIV) Accumulator Supply Checks (Class 3)

1B21-F029A,B,C,D - Outboard MSIV Accumulator Supply Checks (Class 3)

1P57-F555A & F556A - Safety Related Air "A" Accumulator Supply Checks (Class 3)

1P57-F555B & F556B - Safety Related Air "B" Accumulator Supply Checks (Class 3)

1P57-F572B & F574B - Outboard MSIV Accumulator Normal Supply Checks (Class 3)

1P53-F587B & F588B – Airlock Containment Upper Inner Door Accumulator Supply Checks (Class 2)

1P53-F587A & F588A – Airlock Containment Upper Outer Door Accumulator Supply Checks (Class 2)

1P53-F572B & F573B – Airlock Containment Lower Inner Door Accumulator Supply Checks (Class 2)

1P53-F572A & F573A – Airlock Containment Lower Outer Door Accumulator Supply Checks (Class 2)

1P53-F601B & F602B - Airlock Drywell Inner Door Accumulator Supply Checks (Class 2)

1P53-F601A & F602A - Airlock Drywell Outer Door Accumulator Supply Checks (Class 2)

1B21-R011A-F, R011A-G, R011B-F, R011B-G, R011C-F, R011C-G, R011D-F, R011D-G – Reactor Vessel Reference Level Backfill Supply Checks (Class 2)

1D23-F010A & B, F020A & B, F030A & B, F040A & B, F050 – Containment Instrument Line Isolation Valves (Class 2)

1M17-F055 & F065 - Containment Vacuum A & B Isolation Valves (Class 2)

1G43-F050A & B, F060 - Suppression Pool A, B & C Wet Leg Isolation Valves (Class 2)

1C41-F033A & B – Standby Liquid Control Pump Discharge Checks (Class 2)

1E12-F019 – Residual Heat Removal (RHR) Head Spray Inboard Isolation Check (Class 1)

1E12-F023 - RHR Head Spray Outboard Isolation Motor Operated Valve (MOV) (Class 1)

1E12-F050A & B – RHR Shutdown Cooling Isolation Checks (Class 2)

1E12-F053A & B – RHR Shutdown Cooling Isolation MOVs (Class 2)

1N27-F737A & B, F742A & B – Feedwater Leakage Control System (FWLCS) Supply Inboard Isolation Checks (Class 2)

1N27-F737, F740 - FWLCS Supply Outboard Isolation MOVs (Class 2)

For Category A or AC valves, seat leakage is limited to a specific maximum amount in the closed position for fulfilling the required function of shutting down a reactor to the safe shutdown condition, in maintaining the safe shutdown condition, or in mitigating the consequences of an accident. These valves are grouped by the safety function requiring a seat leakage limit. These functions include; 1) accumulator pressure boundary leakage, 2) instrumentation failure (failure of backfill, sensing transmitters, cabinets, etc.), 3) high-to-low system interface, other than designated Reactor Coolant System (RCS) Pressure Isolation Valves (PIVs), 4) system leakage integrity, and 5) parallel pump bypass flow.

## 2. Applicable Code Edition and Addenda

ASME OM Code-2001, with Addenda through OMb-2003

## 3. Applicable Code Requirement

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

ISTC-3630(a), Frequency, requires leakage tests to be conducted at least once every 2 years.

## 4. Reason for Request

A performance-based testing program has been developed that would eliminate the prescriptive test frequency requirements and allow test intervals to be based on system and component performance. Through its own regulatory improvement program, the NRC staff has institutionalized an ongoing effort to eliminate requirements marginal to safety and to reduce the regulatory burden on utilities. A performance-based testing program, utilizing an extended testing interval based on the successful completion of two (2) or

more consecutive leakage rate tests, would take advantage of the findings of NUREG-1493, Appendix A, "Performance Based Containment Leak Test Program." The conclusions drawn by the NUREG suggests that if a component does not fail within two operating cycles, further failures appear to be governed by the random failure rate of the component. The NUREG also states that any test scheme considered should require a failed component to pass at least two consecutive tests before allowing an extended test interval.

The performance-based testing program for ASME valves requiring leakage tests, was developed in much the same manner as the Option B Program for Appendix J testing, which was permitted by amendment of the Code of Federal Regulations on October 26, 1995.

In the studies performed in support of the code change, it was concluded that performance-based testing is feasible without significant risk (NUREG 1493). Also, EPRI Research Project Report TR-104285, "Risk Impact Assessment of Revised Containment Leak Rate Testing Intervals," reaffirmed this position by stating that changes in leakage testing frequencies are feasible without significant risk impact.

The development of this performance-based testing program started with the generation of a leakage test history for each valve that is to be included in the program. Then a review of the test histories for each valve was conducted to establish if a minimum of two (2) consecutive periodic tests had passed and whether any erratic behavior could be detected. All the valves were then placed into a type category (i.e., check, globe, gate, etc.) to establish which types may be more prone to failure. By performing this, a direct comparison could be made of like valves in like systems to determine if some of those valves with good test histories should be monitored more frequently. Valves that pass a minimum of two consecutive tests without any erratic behavior and are not considered suspect valves will be put on an extended interval of four (4) years or 2 refueling cycles, whichever is longer. Any valve not meeting the minimum threshold requirement will be left on a 2-year test interval until at least 2 consecutive tests are acceptable. In addition, if a failure occurs on any extended interval valve, the initial test frequency of 2 years must be re-established until two consecutive tests pass.

## 5. Proposed Alternative and Basis for Use

Leakage rate testing of Category A and AC valves will be performed in accordance with the ASME Performance-Based Testing Program. Valves that have met the threshold of passing two consecutive tests will be permitted to be tested every 4 years or 2 refueling cycles, whichever is longer. Valves which fail their acceptance criterion will be tested each refuel until they pass a minimum of 2 consecutive tests.

Using the provisions of this relief request as an alternative to the leakage test frequency specified by ISTC-3630(a) provides an acceptable level of quality and safety.

The following is a listing of valve groups to be included in the ASME Performance-Based Program. Valves affected by this relief request are categorized by the safety function requiring a seat leakage limit. Valves shall meet the applicable guidelines of the Nuclear Energy Institute (NEI) 94-01, "Industry Guidelines for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," for a performance based testing program. Using the guidelines, valves that have passed a minimum of two consecutive leakage rate tests may be placed on an extended testing interval. All valves placed on an extended testing interval for seat leakage will still have all other associated code required testing (i.e., exercising and position verification), performed at the required frequency by the Inservice Testing Program. Valves that have not passed the minimum of two consecutive tests will continue to be tested during each refueling outage until their test histories become satisfactory to permit an extended testing interval.

Each valve or combination of valves is being assigned an operational frequency rating that is indicative of the expected frequency at which the valve would perform an active function (i.e., opening and closing). The operational frequency assigned would be inverse to the expected rate of valve degradation (i.e., valves seldom exercised would not be expected to lose their valve seat integrity as rapidly as those valves exercised more frequently). The operational frequency ratings will be assigned as follows: seldom, infrequent, occasional and frequent.

SELDOM - Maintenance or convenience type valves in which operation is seldom desired or required.

INFREQUENT - Valves in which operation would be expected at a cold shutdown or greater frequency for testing or other evolutions.

OCCASIONAL - Valves in which operation would be expected at a quarterly frequency for testing or other evolutions.

FREQUENT - Valves in which operation is expected during normal plant operation, for reasons other than testing. Valves assigned as FREQUENT, would be considered for exclusion from the performance based testing program.

ACCUMULATOR PRESSURE BOUNDARY LEAKAGE

Safety related components might rely upon an accumulator as a pressure source for actuation or backup actuation. The following valves are used in the isolation integrity of the accumulator's pressure boundary.

Inboard Main Steam Isolation Valve (MSIV) Accumulator - INFREQUENT:

Inboard Accumulator Supply Check valves (1B21-F024A/B/C/D) makeup an integral portion of the Inboard MSIV accumulator pressure boundary. These accumulators are supplied by the non-safety instrument air system (1P52) and the supply check valves must close and limit seat leakage to a specific maximum amount upon loss of the air supply. Maintaining the seat leakage below the specified limit ensures that the accumulator will maintain sufficient pressure for proper cycling of the Inboard MSIV. Seat leakage is currently being measured by the feed rate required to maintain test pressure in the test volume.

Outboard Main Steam Isolation Valve (MSIV) Accumulator - INFREQUENT:

Outboard accumulator supply check valves (1B21-F029A/B/C/D) makeup an integral portion of the outboard MSIV accumulator pressure boundary. These accumulators are supplied by the non-safety instrument air system (1P52) and the supply check valves must close and limit seat leakage to a specific maximum amount upon loss of the air supply. Post accident, these accumulators are also supplied by the safety related air system (1P57) after it is manually initiated, at which point these valves are no longer required to maintain seat leakage to a specific maximum amount. Maintaining the seat leakage below the specified limit ensures that the accumulator will maintain sufficient pressure for proper cycling of the inboard MSIV until the safety related instrument air supply can be aligned. Seat leakage is currently being measured by the feed rate required to maintain test pressure in the test volume.

Safety Related Air "A" Accumulator - SELDOM:

Safety Related Air "A" Accumulator Supply Check Valves (1P57-F555A & F556A) makeup an integral portion of the Safety Related Air system pressure boundary. These check valves are considered maintenance only. The accumulator is supplied by the non-safety related portion of the safety related air system and the supply check valves must close and limit seat leakage to a specific maximum amount upon loss of the air supply. Maintaining the seat leakage below the specified limit ensures that the accumulator will maintain sufficient pressure for proper cycling of Automatic Depressurization System Safety Relief Valves. Seat leakage measurement is currently satisfied by measuring leakage through a downstream telltale connection while maintaining test pressure on one side.

Safety Related Air "B" Accumulator - SELDOM

Safety related air "B" accumulator supply check valves (1P57-F555B & F556B) makeup an integral portion of the safety related air system pressure boundary. These check valves are considered maintenance only. The accumulator is supplied by the non-safety related portion of the safety related air system and the supply check valves must close and limit seat leakage to a specific maximum amount upon loss of the air supply. Maintaining the seat leakage below the specified limit ensures that the accumulator will maintain sufficient pressure for proper cycling of Automatic Depressurization System Safety Relief Valves. Seat leakage measurement is currently satisfied by measuring leakage through a downstream telltale connection while maintaining test pressure on one side.

Additionally, outboard MSIV accumulator normal supply check valves (1P57-F572B & F574B) makeup an integral portion of the safety related air system pressure boundary after manual initiation. The outboard MSIV accumulators are supplied by the instrument air system and the supply check valves must close and limit seat leakage to a specific maximum amount upon loss of the air supply. Maintaining the seat leakage below the specified limit ensures that the accumulators will maintain sufficient pressure for proper cycling of Automatic Depressurization System Safety Relief Valves as well as closure force for the outboard MSIVs. Seat leakage measurement is currently satisfied by measuring leakage through a downstream telltale connection while maintaining test pressure on one side.

Upper & Lower Containment Inner & Outer Door Airlock Accumulators - FREQUENT:

Airlock door accumulator air supply check valves [1P53-F587B & F588B (Upper Inner Door), 1P53-F587A & 1P53-F588A (upper outer door), 1P53-F572B & F573B (lower inner door), 1P53-F572A & F573A (lower outer door)] makeup an integral part of the specific door seal accumulator pressure boundary. These accumulators are supplied by the non-safety instrument air system and the supply check valves must close and limit seat leakage to a specific maximum amount upon loss of the air supply. Maintaining the seat leakage below the specified limit ensures that the accumulator will maintain sufficient pressure for proper inflation of the small and large door seal. Seat leakage measurement is currently being determined by measuring pressure decay in the test volume and assigning the total apparent leakage rate to the valve combination.

Drywell Inner & Outer Door Airlock Accumulators - INFREQUENT:

Airlock door accumulator air supply check valves [1P53-F601B & F602B (drywell inner door), and 1P53-F601A & F602A (drywell outer door)] makeup an integral part of the specific door seal accumulator pressure boundary. These accumulators are supplied by the non-safety instrument air system and the supply check valves must close and limit seat leakage to a specific maximum amount upon loss of the air supply. Maintaining the seat leakage below the specified limit ensures that the accumulator will maintain sufficient pressure for proper inflation of the small and large door seal. Seat leakage measurement is currently being determined by measuring pressure decay in the test volume and assigning the total apparent leakage rate to the valve combination.

INSTRUMENTATION LEAKAGE ISOLATION

RPV Water Level Instruments Continuous Backfill - INFREQUENT:

Reactor vessel reference level backfill supply check Valves (1B21-R011A-F, R011A-G, R011B-F, R011B-G, R011C-F, R011C-G, R011D-F and R011D-G) provide a makeup flow path from the Control Rod Drive (CRD) System to each of the reactor pressure vessel level sensing line reference legs. This backfill is used to prevent errors in reactor vessel level indication during normal and transient operating conditions. The backfill allows makeup flow of approximately 0.02 gpm, which prevents the buildup of non-condensable gasses. The portion of the CRD system which supplies the makeup flow is non-safety related and the supply check valves must close and limit seat leakage to a specific maximum amount upon loss of the CRD system. Maintaining the seat leakage below the specified limit ensures proper reactor vessel level indication and minimizes a potential reactor coolant leakage path. Seat leakage measurement is currently satisfied by measuring leakage through a downstream telltale connection while maintaining test pressure on one side.

Containment Atmosphere and Water Level Instrumentation - OCCASIONAL:

Containment atmosphere and water level instrument isolation valves (1D23-F010A/B, F020A/B, F030A/B, F040A/B & F050, 1M17-F055 & F065 and 1G43-F050A/B & F060) isolate instrument lines which are considered closed loops outside containment. These solenoid isolation valves are normally open and remotely closed in the unlikely event of a failure of the closed loop portion (i.e., instrumentation) of the system outside containment in order to limit seat leakage to a specific maximum amount. Maintaining the seat leakage below the specified limit ensures the proper isolation of the primary containment pressure boundary if an instrument line failure were to

occur. Seat leakage is currently being measured by the feed rate required to maintain test pressure in the test volume.

HIGH-TO-LOW SYSTEM INTERFACE PRESSURE ISOLATION

High-to-low system interface PIVs are defined as two normally closed valves in series that isolate a high pressure liquid system from an attached low pressure liquid system. Valve testing prevents the unlikely condition of excessive seat leakage from causing a system overpressure condition. These valves remain closed during normal plant operation to limit seat leakage to a specific maximum amount. Maintaining the seat leakage below a specified limit ensures that proper intersystem isolation exists between systems. Reactor coolant system pressure isolation valves are not included in this group. Seat leakage is currently being measured by the feed rate required to maintain test pressure in the test volume.

Residual Heat Removal (RHR) Head Spray Line - INFREQUENT:

RHR head spray inboard isolation check valve (1E12-F019) and outboard isolation motor operated valve (1E12-F023) are the head spray line intersystem pressure isolation valves used to prevent over-pressurization of a low pressure safety related system.

Residual Heat Removal (RHR) Shutdown Cooling Return Lines - INFREQUENT:

RHR shutdown cooling isolation check valves (1E12-F050A/B) and isolation motor operated valves (1E12-F053A/B) are the shutdown cooling return lines intersystem pressure isolation valves used to prevent over-pressurization of a low pressure safety related system.

Feedwater Leakage Control System (FWLCS) Supply Lines - INFREQUENT:

FWLCS supply inboard isolation check valves (1N27-F739A/B & 1N27-F742A/B) and Outboard isolation motor operated valves (1N27-F737 & F740) are the supply lines intersystem pressure isolation valves used to prevent over-pressurization of a low pressure safety related system.

PARALLEL PUMP BYPASS FLOW

Systems that require an active isolation of the parallel pump loop to perform the desired safety function may require the isolation feature to allow minimum system leakage.

Standby Liquid Control (SLC) System Parallel Pump Bypass Flow - OCCASIONAL:

SLC pump discharge check valves (1C41-F033A/B) allow flow of borated coolant to the reactor vessel upon activation and prevent pump bypass flow upon closure. Maintaining seat leakage below the specified limit ensures minimum pump bypass flow. Seat leakage is currently being measured by the feed rate required to maintain test pressure in the test volume.

## 6. Duration of Proposed Alternative

The proposed alternative identified in this relief request shall be utilized during the Third Ten-Year IST Interval.

## 7. Precedent

Perry Nuclear Power Plant, Docket No. 50-440, Safety Evaluation Report (SER) dated August 9, 1999, Safety Evaluation of the Inservice Testing Program Second Ten-Year Interval for Pumps and Valves – Perry Nuclear Power Plant, (TAC No. MA3328). Previously approved as VR-10 in the aforementioned SER. Refer to Attachment 3. Note that the SER did not authorize relief for Category A and AC Reactor Coolant System (RCS) Pressure Isolation Valves (PIVs).

## 8. References

1. NUREG-1493, "Performance-Based Containment Leak Test Program." January 1995.

2. EPRI Research Project Report TR-104285, "Risk Impact Assessment of Revised Containment Leak Rate Testing Intervals."

3. Nuclear Energy Institute (NEI-94-01),"Industry Guidelines for Implementing Performance-Based Option of 10CFR Part 50, Appendix J."

Proposed Alternative
in Accordance with 10 CFR 50.55a(a)(3)(i)

--Alternative Provides Acceptable Level of Quality and Safety--

## 1. ASME Code Component(s) Affected

1M51-F210A & B – Upper Containment Sample Line Isolation Valve (Class 2)
1M51-F220A & B – Drywell Sample Line Isolation Valve (Class 2)
1M51-F230A & B – Drywell Head Sample Line Isolation Valve (Class 2)
1M51-F240A & B – Lower Containment Sample Line Isolation Valve (Class 2)
1M51-F250A & B – Hydrogen Analyzer Sample Exhaust Line Isolation Valve (Class 2)
1P87-F083 – Suppression Pool Return Line Isolation Valve (Class 2)
1P87-F264 – Suppression Pool Return Line Isolation Valve (Class 2)

The 1M51 valves are associated with the Combustible Gas Control/Combustible Gas Control Hydrogen Analysis System. The valves open to permit sampling and measuring of hydrogen concentrations in the drywell and containment during post-accident conditions. The valves also perform a containment isolation function in the closed position.

The 1P87 valves are associated with the Post Accident Sampling System, which is designed to monitor the containment environment and reactor coolant systems after a loss-of-coolant accident. 1P87-F083 and F264 must open to provide a sample return flow path to the suppression pool. The valves also perform a containment isolation function in the closed position.

## 2. Applicable Code Edition and Addenda

ASME OM Code-2001, with Addenda through OMb-2003

## 3. Applicable Code Requirements

ISTC-3500, "Valve Testing Requirements," specifies that active and passive valves in the categories defined shall be tested in accordance with the paragraphs specified in Table ISTC-3500-1 and the applicable requirements of ISTC-5100 and ISTC-5200.

ISTC-5150, "Solenoid Valves," ISTC-5151(a) requires active valves to have their stroke times measured when exercised in accordance with ISTC-3500.

## 4. Reason for Request

These rapid-acting solenoid valves are provided with an unusual control circuitry in that there are three groups of valves with each group actuating from a common control switch. Valves 1M51-F210A/B, F220A/B, F230A/B, F240A/B and F250A/B are actuated by manipulation of a keylock control switch. Valves 1P87-F083 and F264 are also actuated by manipulation of a control switch.

These valves are provided with individual position indicating lights; however, the indicating lights are located in close proximity to one another. In the worst case, individually timing these valves would require five operators each using calibrated stop watches in a relatively confined area. Timing the valves in this manner is not an efficient method of individually stroke timing the valves when the valves stroke in less than two seconds. Therefore, it is requested to conduct timing the slowest valve in the group adequately demonstrates functionality of all the valves. If the slowest valve exceeds the maximum stroke time of two seconds then the valves will be re-timed to ensure the failure is properly attributed to the correct valve.

The proposed test alternative demonstrates full functionality of all the valves within the group in that they are all verified to stroke in less than two seconds and is considered to provide an acceptable level of quality and safety.

## 5. Proposed Alternative and Basis for Use

The slowest valve in a particular group will be timed upon switch manipulation; thereby demonstrating that all valves in the group stroke in less than two seconds.

Using the provisions of this relief request as an alternative to ISTC-3500 and 5151(a) provides a reasonable alternative to the Code requirements, based on the determination that the proposed alternative provides an acceptable level of quality and safety.

## 6. Duration of Proposed Alternative

The proposed alternative identified in this relief request shall be utilized during the Third Ten-Year IST Interval.

## 7. Precedents

None

## 8. References

None

Proposed Alternative
in Accordance with 10 CFR 50.55a(a)(3)(i)

--Alternative Provides Acceptable Level of Quality and Safety--

## 1. ASME Code Component(s) Affected

1B21-F041A, Dikkers Valve (ADS) (Class1)
1B21-F041B, Dikkers Valve (ADS) (Class1)
1B21-F041C, Dikkers Valve (Class1)
1B21-F041D, Dikkers Valve (Class1)
1B21-F041E, Dikkers Valve (ADS) (Class1)
1B21-F041F, Dikkers Valve (ADS) (Class1)
1B21-F041G, Dikkers Valve (Class1)
1B21-F041K, Dikkers Valve (Class1)
1B21-F047B, Dikkers Valve (Class1)
1B21-F047C, Dikkers Valve (Class1)
1B21-F047D, Dikkers Valve (ADS) (Class1)
1B21-F047F, Dikkers Valve (LLS) (Class1)
1B21-F047G, Dikkers Valve (Class1)
1B21-F047H, Dikkers Valve (ADS) (Class1)
1B21-F051A, Dikkers Valve (LLS) (Class1)
1B21-F051B, Dikkers Valve (LLS) (Class1)
1B21-F051C, Dikkers Valve (ADS/LLS) (Class1)
1B21-F051D, Dikkers Valve (LLS) (Class1)
1B21-F051G, Dikkers Valve (ADS/LLS) (Class1)

The Nuclear Boiler System provides Reactor Pressure Vessel (RPV) overpressurization protection by opening the Safety/Relief Valves (SRVs). The SRVs open at their reactor pressure set point. In addition to overpressure protection, the SRVs provide RPV pressure relief by opening to release steam and decrease vessel pressure. Pressure in the vessel is thereby maintained below the American Society of Mechanical Engineers (ASME) Code required limit.

In addition to the above, the Automatic Depressurization System (ADS) and the individual SRVs shall be capable of being manually operated from the main control room. This provides the capability to manually depressurize the RPV in the event the main condenser is not available as a heat sink.

The Nuclear Boiler System ADS provides automatic depressurization of the RPV under certain small break Loss Of Coolant Accident (LOCA) conditions so that the low pressure Emergency Core Cooling Systems (ECCS) can adequately cool the core.

Note that all of the SRVs, those used for ADS as well as those assigned purely for pressure relief, are used for overpressure protection.

## 2. Applicable Code Edition and Addenda

ASME OM Code-2001, with Addenda through OMb-2003

## 3. Applicable Code Requirements

Appendix I, Paragraph I-1320(a), "5-Year Test Interval," specifies that Class 1 pressure relief valves shall be tested at least once every five (5) years, starting with initial electric power generation. No maximum limit is specified for the number of valves to be tested within each interval; however, a minimum of 20% of the valves from each valve group shall be tested within any 24-month interval. This 20% shall consist of valves that have not been tested during the current five 5-year interval, if they exist. The test interval for any individual valve shall not exceed 5 years.

## 4. Reason for Request

The Perry Nuclear Power Plant (PNPP) transitioned from an 18-month fuel cycle to a 24-month fuel cycle on August 29, 2000 via Amendment 115. Prior to transitioning to the 24-month fuel cycle, ASME Code requirements could be satisfied by removing and testing approximately one-third of the 19 SRVs each refueling outage in order to comply with the 5-year test interval requirements for Class 1 pressure relief valves imposed by the code of record during that time. Since transitioning to the 24-month fuel cycle, PNPP must remove at least one-half of the subject relief valves each refueling outage for off-site testing.

The removal of half of the 19 valves versus a third of the valves each outage requires the removal of additional insulation, instrumentation, and other interferences. This additional work also results in an undesirable increase in radiation exposure to maintenance personnel. Therefore, PNPP proposes that each SRV be tested at least once every three refueling cycles (approximately six years) with a minimum of 20% of the valves tested within any 24-month interval.

## 5. Proposed Alternative and Basis for Use

As an alternative to the code required 5-year test interval per Appendix I, paragraph I-1320(a), PNPP proposes that the subject Class 1 pressure relief valves be tested at least once every three refueling cycles (approximately six years) with a minimum of 20% of the valves tested within any 24-month interval. This 20% would consist of valves that have not been tested during the current three cycle interval, if they exist. The test interval for any individual valve would not exceed three refueling cycles.

To provide technical basis for the proposed request, the setpoint testing results were evaluated for the time period from initial operation to the present time (approximately 20 years, 150 data points). The evaluation showed that the average variance from the setpoint testing data is 0.91% and the calculated standard deviation from the average is 0.72 of the nominal setpoint values.

Amendment 101, which was approved on March 3, 1999, changed the allowable SRV setpoint test range from ±1% to ±3%. As part of the evaluation of the setpoint testing data, it was identified that fifty-two (52) tests exceeded the Technical Specifications as-found value of ±1%. There were four (4) failures where the as-found setpoint exceeded the ±3%, with all 4 of the ±3% failures occurring prior to the adoption of Amendment 101. Two (2) of the 4 failures had no as-found setpoint data obtained, due to severe seat leakage. The PNPP data indicates a slight tendency toward higher as-found setpoints, but this tendency is well within the PNPP Technical Specification required limits, which require current SRV setpoint deviations to be within ±3%.

The proposed alternative of increasing the test interval for the subject Class 1 pressure relief valves from five years to three fuel cycles (approximately six years) would continue to provide an acceptable level of quality and safety while restoring the operational and maintenance flexibility that was lost when the 24-month fuel cycle created the unintended consequences of more frequent testing. This proposed alternative will continue to provide assurance of the valves' operational readiness and provides an acceptable level of quality and safety pursuant to 10 CFR 50.55a(a)(3)(i).

## 6. Duration of Proposed Alternative

The proposed alternative identified in this relief request shall be utilized during the Third Ten-Year IST Interval.

## 7. Precedent

Nine Mile Point Nuclear Station, Docket No. 50-410, Safety Evaluation Report (SER) dated April 17, 2001, "Safety Evaluation of the Alternative to ASME Code Regarding Inservice Testing of Main Steam Safety/Relief Valves, (TAC No. MB0290)."

## 8. References

1. SER dated August 29, 2000. Perry Nuclear Power Plant, Unit 1, Docket No. 50-440, "Issuance of Amendment RE: Revisions of Various Surveillance Requirements to Support a 24-Month Operating Cycle (TAC No. MA5930)."

2. SER dated March 3, 1999. Perry Nuclear Power Plant, Unit 1, Docket No. 50-440, Amendment 101 (SRVs 1% to 3%) to Facility Operating License No. NPF-58 Perry Nuclear Power Plant, Unit 1 (TAC NO. MA2290).