UNITED STATES
NUCLEAR REGULATORY COMMISSION
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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO THE INSERVICE TESTING PROGRAM

CONSUMERS ENERGY

PALISADES NUCLEAR PLANT

DOCKET NUMBER 50-255

1.0 INTRODUCTION

The *Code of Federal Regulations*, 10 CFR 50.55a, requires that inservice testing (IST) of certain American Society of Mechanical Engineers (ASME) Code Class 1, 2, and 3 pumps and valves be performed in accordance with Section XI of the ASME *Boiler and Pressure Vessel Code* (the Code) and applicable addenda, except where alternatives have been authorized or relief has been requested by the licensee and granted by the Commission pursuant to Sections (a)(3)(i), (a)(3)(ii), or (f)(6)(i) of 10 CFR 50.55a. In proposing alternatives or requesting relief, the licensee must demonstrate that (1) the proposed alternatives provide an acceptable level of quality and safety, (2) compliance would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety, or (3) conformance is impractical for its facility. Section 50.55a authorizes the Commission to approve alternatives and to grant relief from ASME Code requirements upon making the necessary findings. Guidance related to the development and implementation of IST programs is given in Generic Letter (GL) 89-04, "Guidance on Developing Acceptable Inservice Testing Programs," issued April 3, 1989, and its Supplement 1 issued April 4, 1995. Additional guidance can be found in NUREG-1482, "Guidelines for Inservice Testing at Nuclear Power Plants," NUREG/CR-6396, "Examples, Clarifications, and Guidance on Preparing Requests for Relief from Pump and Valve Inservice Testing Requirements," and "Summary of Public Workshops Held in NRC Regions on Inspection Procedure 73756, 'Inservice Testing of Pumps and Valves,' and Answers to Panel Questions on Inservice Testing Issues."

The 1989 Edition of the ASME Code is the latest edition incorporated by reference in Paragraph (b) of Section 50.55a. Subsection IWV of the 1989 Edition, which provides the requirements for IST of valves, references Part 10 of the American National Standards Institute/ASME *Operations and Maintenance Standards* (OM-10) as the rules for IST of valves. OM-10 replaces specific requirements in previous editions of Section XI, Subsection IWV, of the ASME Code. Subsection IWP of the 1989 Edition, which provides the requirements for IST of pumps, references Part 6 of the American National Standards Institute/ASME *Operations and Maintenance Standards* (OM-6) as the rules for IST of pumps. OM-6 replaces specific requirements in previous editions of Section XI, Subsection IWP of the ASME Code.
Four relief requests were submitted by Consumers Energy (licensee): VRR Nos. 28 and 32 in two letters dated June 3, 1997, and VRR Nos. 30 and 31 in a letter dated July 24, 1997.

VRR No. 32 is a new relief request. VRR Nos. 28, 30, and 31 have been revised by the licensee in response to an NRC safety evaluation dated August 30, 1996, which had denied VRR No. 28 and granted interim relief until August 21, 1997, for VRR Nos. 30 and 31. On August 26, 1997, the NRC issued a request for additional information for the relief requests. In a letter dated September 29, 1997, the licensee responded to this request for additional information. In response to action items identified in the referenced NRC Safety Evaluation, the licensee also submitted a letter dated August 29, 1997, that included a revised pump relief request PRR No. 7 and revised valve relief requests VRR Nos. 7, 12, 23, and 29. The evaluation of the relief requests and ancillary information is provided below. Additionally, several cold shutdown justifications and refueling outage justifications were revised as a result of action items in the referenced Safety Evaluation; however, these revisions are not evaluated herein, but are subject to further review during IST inspections. The Palisades Plant IST Program was developed to the 1989 Edition of ASME Section XI for the third 10-year interval that began August 21, 1995.

2.0 REVISED RELIEF REQUEST VRR NO. 7

The Safety Evaluation dated August 30, 1996, recommended that, by August 30, 1997, the licensee (1) determine if the safety injection refueling water tank discharge valves CK-ES3239 and -ES3240 should be changed from Category C to Category A/C and revise, as necessary, the valve table, and (2) incorporate a performance review into the testing procedure for these valves so that adjustment in the inspection (or testing) frequency can be made as needed. In response to the safety evaluation, valves CK-ES3239 and -ES3240 have been determined by the licensee to be Category A/C; and the valve test table and the relief request, VRR No. 7, have been revised to specify the correct category. Also, the licensee has included performance reviews of test results to predict valve degradation so that valves can be repaired prior to failure. The proposed alternative in VRR No. 7 was previously approved by the NRC safety evaluation dated August 30, 1996, pursuant to 10 CFR 50.55a(a)(3)(ii). Therefore, the revised relief request remains approved as discussed in the referenced NRC safety evaluation. No further NRC review is included herein. The implementation of the commitments made in this relief request is subject to further review during NRC inspections.

3.0 REVISED RELIEF REQUEST VRR NO. 12

The proposed alternative in VRR No. 12 was previously approved by the NRC safety evaluation dated August 30, 1996, pursuant to 10 CFR 50.55a(a)(3)(ii). The safety evaluation recommended that, by August 30, 1997, the licensee should consider incorporating a preventive maintenance program for power-operated valves CV-0944 and CV-0977B to monitor degradation. In response to this recommendation, a discussion of the preventive maintenance program for the valves in question was added to relief request VRR No. 12. Therefore, the revised relief request remains approved as discussed in the referenced NRC safety evaluation. Implementation of the commitments made in this relief request is subject to further review during NRC inspections.

4.0 REVISED RELIEF REQUEST VRR NO. 23

The proposed alternative in VRR No. 23 was previously approved by the NRC safety evaluation dated August 30, 1996, pursuant to 10 CFR 50.55a(a)(3)(i). The safety evaluation recommended that the licensee revise VRR No. 23 to include Category C valves. In response to the safety evaluation, VRR-23 has been revised to extend applicability to Category C valves. Therefore, the revised relief request remains approved as discussed in the referenced NRC safety evaluation.

5.0 REVISED RELIEF REQUEST VRR NO. 28

Revised VRR No. 28 requests relief from OM-1, paragraph 1.3.4(b), regarding the test frequency for Class 2 and 3 pressure relief devices. This relief request pertains to valves RV-0402, RV-0403, RV-0954, RV-0955, and RV-2109 in the Component Cooling System and the Engineering Safeguards System.

5.1 Licensee's Basis for Request

The licensee provided the following basis for the relief request:

> Complying with the Code is impractical. RV-0402, RV-0403, RV-0954, RV-0955, and RV-2109 are designed to protect the Shutdown Cooling Heat Exchangers and the Spent Fuel Pool Heat Exchangers from thermal expansion of water when the exchangers are isolated for maintenance or inspection. These thermal relief valves have no safety function when the heat exchangers are in service. The associated heat exchangers must be isolated to remove these thermal relief valves. However, the Shutdown Cooling Heat Exchangers cannot be isolated during power operations, nor isolated when there is fuel in the Reactor Vessel. An alternate means of cooling the Spent Fuel Pool would be required to isolate the Spent Fuel Pool Heat Exchangers.
>
> Compliance would result in a hardship without a compensating increase in safety. RV-0402, RV-0403, RV-0954, and RV-0955 are welded in, and are located in a high radiation area. Removal testing and reinstallation would cause an increase of radiation exposure to plant staff. RV-0402, RV-0403, RV-0954, RV-0955, and RV-2109 do not provide overpressure protection when the associated heat exchanger is in service, nor do they serve a safety function during Plant operation to ensure the reactor can be safely shutdown or to mitigate the consequences of an accident. These relief valves are only needed when the associated heat exchanger is isolated. Therefore, there is no benefit in verifying the set point for these relief valves while the heat exchangers remain in service. In lieu of the 10-year periodic testing required by OM-1987, these relief valves will be tested on the basis of need during periods of isolation of the heat exchangers.
>
> Due to the critical timing and sequencing of proceduralized test activities, the incorporation into surveillance test procedures of plant and Nuclear Plant Reliability Data System (NPRDS) valve performance data reviews to make adjustments to testing frequency or methods is not considered practicable. Rather, Palisades, on a much

broader spectrum than plant data and NPRDS, reviews industry and operating experiences applicable to valve testing per its Engineering Manual and Administrative Procedures. These procedures ensure significant events that would affect valve testing frequencies and test methods are evaluated by appropriate plant personnel. The responsible engineer would then make any necessary revisions to test procedures.

A performance review of plant data and the NPRDS to recommend adjustments in testing frequencies and/or methods is not practicable within the scope of Palisades Technical Specification Surveillance Procedures. Technical Specification Surveillance Procedures provide plant personnel with step by step instructions to test plant equipment, and include test result acceptance criteria. Often, the timing and sequencing of proceduralized test activities are critical, with testing posing a potential impact on the operational status of other plant systems.

Completed Technical Specification Surveillance Procedures are reviewed by the Inservice Testing (IST) valve coordinator in accordance with the requirements of Palisades Nuclear Plant Engineering Manual Procedure EM-09-02, "Inservice Testing of Plant Valves." A Condition Report is initiated for any valve testing failure, and is evaluated in accordance with Palisades Administrative Procedure 3.03, "Corrective Action Process." The results of Corrective Action Evaluations may include adjustments to valve testing frequencies and/or test methods.

Palisades also has procedures in place to evaluate and incorporate data from NPRDS, Industry Experience, and plant data. Palisades Engineering Manual Procedure EM-20, "System Performance Monitoring," is intended to provide indications of system performance, system operations, component maintenance, and structure conditions that could contribute to inadequate system and structure performance. This procedure requires System Engineers to perform industry experience reviews for systems, structures, and components assigned to them. During the review, System Engineers determine if a Condition Report is needed to resolve potential safety issues. Any such finding is documented in a System Health Report and includes a discussion of any further evaluation and action taken.

An evaluation of a Condition Report requires a review of industry data to learn from previous similar instances or to identify causes which may not be apparent when considering information from one event. Industry data available includes the Nuclear Network, NPRDS, and the Palisades Industry Experience Data Base. In the future, Palisades will be utilizing INPO's [Institute of Nuclear Power Operations'] successor to NPRDS, the "Equipment Performance Information Exchange (EPIX)" data base. EPIX is the data base used for industry failure reporting.

In order to remove RV-2109 for testing, an alternate means of cooling the spent fuel pool must be established. Aligning the Shutdown Cooling System (SDC) to the Spent Fuel Pool (SFP) Cooling System is an approved alternate means of cooling the SFP. The alignment of SDC to the SFP is controlled by Palisades Nuclear Plant System Operating Procedure SOP-27, "Fuel Pool System," and can only be accomplished under two plant conditions:

1. With the Reactor Cavity Flooded.

    a. Primary Coolant at zero pressure.
    b. Reactor Cavity and SFP flooded to equal level.
    c. South tilt pit gate removed.
    d. Transfer tube open.
    e. SDC in service per Palisades System Operating Procedure SOP-3, "Safety Injection and Shutdown Cooling System."

2. With the full core removed from the Reactor Vessel and stored in the SFP.

    a. Core off-load complete.
    b. SFP isolated from the Reactor Cavity.
    c. SFP monitoring requirements of Technical Specification 3.8.5 are met.
    d. SDC System secured per System Operating Procedure SOP-3, "Safety Injection and Shutdown Cooling System."
    e. Pre-job briefings completed for all operators involved in this evolution.

Both plant conditions require two eight inch swing elbows that cross-tie the Fuel Pool and Shutdown Cooling Systems to be placed in position. One of the swing elbows is located in the Spent Fuel Pool Heat Exchanger Room, and the other is located in the West Engineering Safeguards Room. To swing each elbow, Shutdown Cooling and SFP Cooling are isolated from the piping by manual isolation valves. Failure of a SDC manual isolation valve during elbow positioning would cause Primary Coolant to leak into the West Engineering Safeguards Room or the Spent Fuel Pool Heat Exchanger Room, and a potential loss of SDC. Failure of a SFP manual isolation valve would cause SFP coolant to leak into the West Engineering Safeguards Room or the Spent Fuel Pool Heat Exchanger Room. Leakage of either Primary Coolant or SFP Coolant could cause personnel exposure to transuranic contamination due to the presence of a fuel rod which failed in 1993. Swinging the West Engineering Safeguards cross-tie elbows places personnel in a 14 mrem/hr to 25 mrem/hr field. Swinging the cross-tie elbow in the SDC Heat Exchanger Room places personnel in a 10 mrem/hr to 30 mrem/hr field and near a 600 mrem/hr hot spot.

The hardships involved in using alternate means of cooling the Spent Fuel Pool include the following.

1. Risk of a loss of Shutdown Cooling through open eight inch piping.
2. Potential Primary Coolant System and/or Spent Fuel Pool Coolant leakage.
3. Potential for transuranic contamination and personnel exposure.
4. Increased radiation exposure to plant personnel.

It is not practicable to test and repair RV-0402, RV-0403, RV-0954, RV-0955, or RV-2109 prior to entering a potential overpressure condition. This is because it is the process of isolating the heat exchangers that causes the overpressure condition, and

the relief valves can only be removed for testing after the heat exchangers are isolated and drained.

RV-0402 and RV-0403 are the Shutdown Cooling Heat Exchangers tube side thermal relief valves. They are designed to protect the Shutdown Cooling Heat Exchangers from overpressure due to thermal expansion of water when the exchangers are isolated from the Primary Coolant System (PCS). RV-0954 and RV-0955 are the Shutdown Cooling Heat Exchangers shell side thermal relief valves. They are designed to protect the Shutdown Cooling Heat Exchangers from overpressure when the exchangers are isolated from the Component Cooling Water (CCW) System. RV-0402, RV-0403, RV-0954, and RV-0955 are located within the boundaries of the heat exchangers' isolation valves. It is the process of isolating a heat exchanger that places it in a potential overpressure condition due to thermal expansion of water. After the Shutdown Cooling Heat Exchangers are isolated and vented, then the potential overpressure condition is eliminated. However, it is only after the Shutdown Cooling Heat Exchangers are isolated, vented, and drained that the relief valves can be removed and tested.

RV-2109 is the Spent Fuel Pool Heat Exchanger shell side thermal relief valve. It is design to protect the Spent Fuel Pool Heat Exchanger from overpressure when the exchanger is isolated from the Component Cooling Water (CCW) System. RV-2109 is located within the boundaries of the heat exchangers' isolation valves. It is the process of isolating the heat exchanger that places it in a potential overpressure condition due to thermal expansion of water. After the Spent Fuel Pool Heat Exchanger is isolated and vented, then the potential overpressure condition is eliminated. However, it is only after the Spent Fuel Pool Heat Exchanger is isolated, vented, and drained that RV-2109 can be removed and tested.

5.2 Proposed Alternate Testing

The licensee proposed the following alternative:

Each time Shutdown Cooling Heat Exchanger E-60A is removed from service and isolated for maintenance or inspection, RV-0402 and RV-0954 will be tested in accordance with OM-1987, Part 1, Section 8 unless [they have been] tested within the last 48 months.

Each time Shutdown Cooling Heat Exchanger E-60B is removed from service and isolated for maintenance or inspection, RV-0403 and RV-0955 will be tested in accordance with OM-1987, Part 1, Section 8 unless they have been tested within the last 48 months.

Each time Spent Fuel Pool Heat Exchangers E-53A and E-53B are removed from service and isolated for maintenance or inspection, RV-2109 will be tested in accordance with OM-1987, Part 1, Section 8 unless [they have] been tested within the last 48 months.

RV-0402, RV-0403, RV-0954, RV-0955, and RV-2109 will be removed and setpoint tested at least once every ten years.

5.3 Evaluation

OM-10, which references OM-1, requires testing of pressure relief devices that provide overpressure protection to components that function to shut down the reactor, maintain a safe shutdown condition, or mitigate the consequences of an accident. The valves identified in this relief request provide thermal overpressure protection for such components. For Class 2 and 3 pressure relief valves, OM-1, paragraph 1.3.4, requires a minimum of 20% of the valves to be tested within any 48 months. A valve in a group of two valves is required to be tested at least once within any 48-month interval on an alternating basis. The licensee states that these thermal relief valves are needed only when the associated heat exchanger is isolated and proposes to test the relief valves in accordance with Section 8 of OM-1 when the associated heat exchanger is isolated for maintenance or inspection unless they have been tested within the last 48 months. RV-0402, RV-0403, RV-0954, RV-0955, and RV-2109 will be removed and setpoint tested at least once every 10 years.

The shutdown cooling heat exchangers cannot be isolated during power operation or when there is fuel in the reactor vessel. The licensee states that a plant core off-load would be required to perform the shutdown cooling heat exchanger relief valve setpoint testing. The spent fuel pool heat exchanger cannot be isolated without an alternate means of cooling the spent fuel pool. The licensee states that the hardships involved in using alternate means of cooling the spent fuel pool include (1) risk of a loss of shutdown cooling, (2) potential for primary coolant system, and/or spent fuel pool coolant leakage, (3) potential for transuranic contamination, and (4) increased radiation exposure to plant personnel.

Given the hardship associated with requiring a plant core off-load or an alternate means of cooling the spent fuel pool, the licensee's proposal provides a practicable alternative to the Code requirements. These relief valves are needed only when the associated heat exchanger is isolated. Imposing the Code requirement for setpoint testing of RV-0402, RV-0403, RV-0954, RV-0955, and RV-2109 would result in hardship without a compensating increase in the level of quality and safety. To resolve any potential safety issues relative to inservice testing of these valves, Palisades has procedures in place to evaluate and incorporate data from NPRDS, industry experience, and plant data. If the licensee determines through the evaluation of the performance data that the testing interval is too long to assure the operational readiness of the valves, the relief request would have to be reexamined. If a full core off-load or an alternate means of cooling the spent fuel pool is not feasible to accommodate the appropriate interval, some other method of assuring operational readiness must be provided, such as enhanced quality assurance and preventive maintenance. The use of other methods would require review and approval by the NRC staff prior to implementation.

5.4 Conclusion

The alternative testing frequency is authorized pursuant to 10 CFR 50.55a(a)(3)(ii) based on the hardship or unusual difficulty without a compensating increase in the level of quality and safety that would result if the requirements to perform setpoint testing at least once within any 48-month interval on an alternating basis were imposed.

6.0 REVISED RELIEF REQUEST VRR NO. 29

Revised VRR No. 29 requests relief from OM-10, paragraphs 4.2.1.4 and 4.2.1.8, regarding the stroke timing requirements for the main steam atmospheric dump valves. OM Part 10, 4.2.1 requires Category A and B power-operated valves to be individually full stroke exercised and stroked timed nominally every 3 months except as provided by paragraphs 4.2.1.2, 4.2.1.5, and 4.2.1.7. This relief request pertains to CV-0779, 0780, 0781, and 0782. The NRC safety evaluation dated August 30, 1996, stated that, by August 30, 1997, the licensee should revise the relief request to discuss whether these valves are within the scope of the IST program. In response to the safety evaluation, this relief request has been revised to provide a better description of the safety function of the valves in question.

6.1 Licensee's Basis for Request

The licensee provided the following basis for the relief request:

> The four steam generator atmospheric dump valves have an active safety function in the open position to remove decay heat from the primary system by discharging steam from the steam generators in order to bring the plant to shutdown cooling entry conditions following a steam generator tube rupture. These valves have an active safety function in the closed position to isolate the affected steam generator during a steam generator tube rupture event. The affected steam generator is isolated after the associated hot leg temperature falls below 525°F in order to prevent reopening the secondary side safety valves.
>
> Relief is requested in accordance with 10 CFR 50.55a(f)(5)(iii) on the basis that stroke timing the atmospheric steam dump valves is impractical. These valves were not originally designed to be a safety-related method of decay heat removal and, therefore, were not equipped with a control system that is suitable for performing stroke timing in accordance with OMa-1988, Part 10. These valves have a position indicating light in the closed position only, therefore stroke timing in the open position is not practicable. Subsequent to the original plant design, it was determined that these valves play an important role in the removal of PCS decay heat following a steam generator tube rupture accident. Therefore, atmospheric steam dump valves have been included in the scope of the IST program. However, at the time these valves were upgraded, no specific testing commitments were made to the NRC. There is no specific requirement for stroke time in either the open or closed direction.
>
> The preventive maintenance program includes three Periodic Preplanned Activity Controls (PPACs). The backup nitrogen supply instruments are calibrated biannually per PPAC MSS006. The valves and valve operators are inspected, repaired, and tested at an eight-year interval per PPAC MSS078. The atmospheric dump valve instrument is calibrated during T-207 per PPAC MSS096.

6.2 Proposed Alternate Testing

The licensee proposed the following alternative:

> These valves will be full-stroke exercised in both the open and closed directions each cold shutdown per Surveillance Procedure QO-6. Stroke Time will not be measured. The valves will be observed locally during stroke testing to ensure the valves stroke promptly and do not exhibit any abnormal or erratic behavior.

6.3 Evaluation

OM Part 10, paragraph 4.2.1 requires that Category A and B power-operated valves be stroke tested every three months, except as provided by paragraphs 4.2.1.2, 4.2.1.5, and 4.2.1.7. The licensee has proposed to exercise the atmospheric steam dump valves CV-0779, 0780, 0781, and 0782 during cold shutdowns by local observation, without measuring stroke times.
It is not evident based on the information presented that the licensee has considered non-intrusive techniques or considered assigning reasonable, objective acceptance criteria to an observable parameter, such as valve stem movement, flow rate, or ΔP, to measure stroke times and assess degradation. Also, the basis does not include justification for deferral of testing to cold shutdowns. Therefore, long-term relief cannot be granted. An interim relief would provide the time necessary for the licensee to address these issues. If the Code requirements cannot be met, the information presented should be detailed enough so that it is evident that the quarterly testing frequency and stroke timing are impractical or would result in hardship without a compensating increase in the level of quality and safety. Immediate compliance would be an undue burden since the proposed alternative testing should provide a reasonable assurance of operational readiness during the interim.

6.4 Conclusion

Interim relief is granted from the Code power-operated valve stroke timing and testing requirements for the atmospheric steam dump valves CV-0779, 0780, 0781, and 0782 pursuant to 10 CFR 50.55a(a)(3)(ii) based on the determination that immediate compliance with the specified requirements results in a hardship without a compensating increase in the level of quality and safety. Relief is granted for an interim period of 120 days from the date of the safety evaluation to allow the licensee time to (1) provide basis for deferral of testing to cold shutdowns, (2) address non-intrusive techniques, and (3) address assigning reasonable, objective acceptance criteria to an observable parameter, such as valve stem movement, flow rate, or ΔP, to measure stroke time and assess degradation.

7.0 REVISED RELIEF REQUEST VRR NO. 30

Revised VRR No. 30 requests relief from OM-1, paragraph 1.3.3.1(b), regarding the test frequency for Class 1 pressure relief devices. This relief request pertains to valve RV-0401 in the Shutdown Cooling System.

7.1 Licensee's Basis for Request

The licensee provided the following basis for the relief request:

> This valve has a safety function to provide overpressure protection to the Shutdown Cooling return header between valves MO-3015 and MO-3016.
>
> Relief is requested in accordance with 10 CFR 50.55a(f)(5)(iii) from the requirement to perform set point verification on either a 5-year basis or a 24-month basis because such testing is impractical. Relief valve RV-0401 is welded in place and is located in the letdown from the Primary Coolant System to the Shutdown Cooling System. Testing cannot be performed with the Primary Coolant System greater than cold shutdown because RV-0401 provides the second isolation barrier for the PCS. Failure of the first isolation barrier (MO-3015) would result in uncontrollable and highly contaminated PCS leakage.
>
> Testing cannot be performed during cold shutdown with Shutdown Cooling in service because Palisades has no alternate letdown path for Shutdown Cooling. Shutdown Cooling cannot be isolated unless there is a full core off load. Based on this fact, RV-0401 can only be tested during full core off loads.
>
> A historical review of set point testing for Teledyne Farris Engineering Safety-Relief Valve Model 2741 PKD/S4 indicates maximum test intervals are dependent on system service conditions. Similar valves in other systems were also reviewed. However, those valves were subjected to more dynamic system pressure fluctuations and vibration caused by reciprocating positive displacement pumps than RV-0401. Relief valves subjected to dynamic loads would typically require shorter set point intervals than in static systems.
>
> RV-0401 provides overpressure protection to the Shutdown Cooling return header between valves MO-3015 and MO-3016. This portion of the Shutdown Cooling System is static except when providing core cooling during plant Cold Shutdowns. With more than 10 years of service, historical test data shows RV-0401 set-point to have drifted from 2482 to 2450 psig. Even with set point drift, system operability and safety requirements were maintained with no reduction of Safety Margin. Therefore, a maximum set point test interval of 10 years is acceptable for RV-0401.
>
> Due to the critical timing and sequencing of proceduralized test activities, the incorporation into surveillance test procedures of plant and Nuclear Plant Reliability Data System (NPRDS) valve performance data reviews to make adjustments to testing frequency or method is not considered practicable. Rather, Palisades, on a much broader spectrum than plant data and NPRDS, reviews industry and operating experiences applicable to valve testing per its Engineering Manual and Administrative Procedures. These procedures ensure significant events that would affect valve testing frequencies and test methods are evaluated by appropriate plant personnel. The responsible engineer would then make any necessary revisions to test procedures.

A performance review of plant data and the NPRDS to recommend adjustments in testing frequencies and/or methods is not practicable within the scope of Palisades Technical Specification Surveillance Procedures. Technical Specification Surveillance Procedures provide plant personnel with step by step instructions to test plant equipment, and include test result acceptance criteria. Often, the timing and sequencing of proceduralized test activities are critical, with testing posing a potential impact on the operational status of other plant systems.

Completed Technical Specification Surveillance Procedures are reviewed by the Inservice Testing (IST) valve coordinator in accordance with the requirements of Palisades Nuclear Plant Engineering Manual Procedure EM-09-02, "Inservice Testing of Plant Valves." A Condition Report is initiated for any valve testing failure, and is evaluated in accordance with Palisades Administrative Procedure 3.03, "Corrective Action Process." The results of Corrective Action Evaluations may include adjustments to valve testing frequencies and/or test methods.

Palisades also has procedures in place to evaluate and incorporate data from NPRDS, Industry Experience, and plant data. Palisades Engineering Manual Procedure EM-20, "System Performance Monitoring," is intended to provide indications of system performance, system operations, component maintenance, and structure conditions that could contribute to inadequate system and structure performance. This procedure requires System Engineers to perform industry experience reviews for systems, structures, and components assigned to them. During the review, System Engineers determine if a Condition Report is needed to resolve potential safety issues. Any such finding is documented in a System Health Report and includes a discussion of any further evaluation and action taken.

An evaluation of a Condition Report requires a review of industry data to learn from previous similar instances or to identify causes which may not be apparent when considering information from one event. Industry data available includes the Nuclear Network, NPRDS, and the Palisades Industry Experience Data Base. In the future, Palisades will be utilizing INPO's successor to NPRDS, the "Equipment Performance Information Exchange (EPIX)" data base. EPIX is the data base used for industry failure reporting.

The likelihood of RV-0401... opening with Shutdown Cooling in Service due to setpoint drift is low during initial alignment of SDC to the PCS, and becomes extremely low as PCS pressure, temperature, and inventory are reduced.

The greatest probability for the relief valves actuating due to setpoint drift is when SDC is initially aligned to the Primary Coolant System (PCS). During this plant operating condition, PCS applied system pressure to SDC is procedurally allowed to be as high as 270 psia (255.3 psig).

It is highly unlikely that the setpoint of RV-0401 would drift from 2485 psig to 273.3 psig, which is the calculated SDC system pressure at RV-0401 for the above described operating condition.

After initial alignment of the PCS to SDC, PCS temperature and pressure drop significantly below 270 psia and 300°F, and the likelihood of a relief valve lifting due to setpoint drift is further reduced. When the PCS is at reduced inventory, system [pressure at the relief valve is] < 5 psig.

At reduced PCS inventory the probability of a SDC relief valve lifting is extremely low for the following reasons:

1. For the plant to get to reduced PCS inventory conditions, the SDC relief valves must have remained seated during higher system operating pressures, such as system pressure experienced when SDC is initially aligned to the PCS.

2. It is overly conservative to postulate a setpoint drift greater than 60% of normal setpoint.

An emergency procedure is in place to mitigate the consequences of a Shutdown Cooling System relief valve opening due to setpoint drift with Shutdown Cooling in service. This is a low probability event, and the risk for this scenario is adequately addressed by the actions of Palisades Nuclear Plant Off Normal Procedure ONP-17, "Loss of Shutdown Cooling."

Palisades Nuclear Plant Off Normal Procedure ONP-17, "Loss of Shutdown Cooling," specifically addresses loss of Primary Coolant inventory in the unlikely event of RV-0401 opening with Shutdown Cooling in service. If the Primary System Drain Tank (PSDT) level is rising, ONP-17 directs Operators to verify Shutdown Cooling Relief Valve RV-0401 has not lifted. Assuming...the relief valve lift is due to excessive setpoint drift, simple gagging would successfully stop the loss of PCS inventory. However, if the diversion is not stopped, ONP-17 directs alternate PCS heat removal by using the Containment Spray Pumps, Spent Fuel Pool Cooling, or Steam Generators.

When Shutdown Cooling is in service, the greater potential for a stuck open Shutdown Cooling system relief valve is when SDC is initially aligned to the PCS. In this plant condition, the consequences of a stuck open Shutdown Cooling system relief valve do not include loss-of-cooling-accident. At other SDC operating conditions the potential of a stuck open SDC relief valve is significantly lower. However, if a SDC relief valve did lift and stick open, PCS make up from the Safety Injection and Refueling Water (SIRW) Tank would allow adequate time to implement Off Normal Operating Procedures and correct the problem as previously described.

The greatest probability of relief [valve] RV-0401...actuating and sticking open due to Shutdown Cooling (SDC) System pressure occurs when SDC is initially aligned to the Primary Cooling System (PCS). During this plant operating condition, PCS applied system pressure to SDC is procedurally allowed to be as high as 270 psia (255.3 psig). Assuming the relief valve(s) lift due to setpoint drift, premature actuation will result in a PCS loss-of-coolant until the PCS can be isolated from SDC and realigned to the Steam Generators per procedures ONP-17.

After initial alignment of the PCS to SDC, PCS temperature and pressure drop significantly below 270 psia and 300°F, and the likelihood of a relief valve lifting due to setpoint drift is further reduced. Should the unlikely event of a SDC system relief valve lifting and sticking open occur at these plant conditions, a loss-of-coolant accident would result. Assuming simultaneous full flow actuation of all three relief valves [RV-0401, RV-3162, and RV-3164], make up water would be available from the SIRW Tank for approximately 15 hours. Also, ONP-17 effectively deals with a loss of Primary Coolant inventory conditions, by directing operators to determine the cause of the loss of Primary Coolant, and reseat the stuck open relief valve.

Recovery from a failure of RV-0401...is detailed in Palisades Nuclear Plant Off Normal Procedure ONP-17, "Loss of Shutdown Cooling." The event ends by either reseating or gagging the affected relief valve.

RV-0401 is a Teledyne Farris Engineering 2741 PKD/S4 safety relief valve with a 0.110 Sq. In. orifice. It is located on the SDC suction line to the Low Pressure Safety Injection (LPSI) pumps between valves MO-3015 and MO-3016 at approximately the 609 foot elevation in containment. RV-0401 is installed in the SDC System to protect the section of piping between MO-3015 and MO-3016 from overpressure due to thermal expansion of water with MO-3015 and MO-3016 closed. Its setpoint is set at 2485 psig to accommodate slight seat leakage from MO-3015 with the PCS at normal operating pressure. RV-0401 discharges to the Primary System Drain Tank (PSDT) T-74.

It is highly unlikely that the setpoint of RV-0401 would drift from 2485 psig down to 273.3 psig. However, should RV-0401 setpoint drift to 273.3 psig, and the SDC system experience an upset of 10% above 273.3 psig, then based solely on orifice size and not considering restrictions in the relief valve discharge piping, a stuck open RV-0401 could divert 30 gpm of water from the Low Pressure Safety Injection (LPSI) Pumps suction piping to the PSDT. This condition would result in entering Palisades Nuclear Plant Off Normal Procedure ONP-17, "Loss of Shutdown Cooling." This procedure directs operators to determine if the Primary System Drain Tank (PSDT) level is rising. If the PSDT level is rising, then it is determined if SDC relief valve RV-0401 has lifted. Once it has been determined RV-0401 has lifted, simple gagging of the relief valve would end the event.

7.2 Proposed Alternate Testing

The licensee proposed the following alternative:

Consumers Energy will verify RV-0401 set points at least once every 10 years during full core off loads unless testing has been performed in the previous 24 months as required by Technical Specification Surveillance Procedure RT-116, "Miscellaneous Safety Systems Safety Valve Set point Testing."

7.3 Evaluation

OM-10, which references OM-1, requires testing of pressure relief devices that provide overpressure protection to components that function to shut down the reactor, maintain a safe shutdown condition, or mitigate the consequences of an accident. The Class 1 valve in question, RV-0401, provides overpressure protection for the shutdown cooling return header. For Class 1 pressure relief valves, OM-1, paragraph 1.3.3, requires a minimum of 20% of the valves to be tested within any 24 months. A valve in a group by itself is required to be tested at least once within any 24-month interval. In lieu of the Code requirements, the licensee proposes to setpoint test the relief valve at least once every 10 years during full core off-loads unless testing has been performed in the previous 24 months.

RV-0401 provides the second isolation barrier for the PCS. The [piping and instrumentation diagrams] P&IDs do not show a pressure indicator in the piping between the two barriers that could indicate the failure of the first barrier. Because a failure of the first isolation barrier could cause a loss-of-coolant (LOCA) during testing of RV-0401, testing during power operation is impractical. This valve cannot be tested when the shutdown cooling system is in service because Palisades has no alternate letdown path for shutdown cooling; therefore, the valve can be tested only when shutdown cooling can be isolated during a full core off-load.

The licensee states that a LOCA could result if RV-0401 lifts and sticks open when the shutdown cooling system is in service. Assuming simultaneous full flow actuation of the three relief valves (RV-0401, RV-3162, and RV-3164) in the line, makeup water would be available from the SIRW tank for approximately 15 hours. The Palisades Nuclear Plant Off Normal Procedure ONP-17, "Loss of Shutdown Cooling," directs the operators in this scenario to determine the cause of the loss of primary coolant and reseat the open relief valves.

The licensee states that, in order for RV-0401 to lift when the shutdown cooling system is in service, the setpoint would have to drift from 2485 psig down to 273.3 psig. If this happens, the licensee states that a stuck open RV-0401 would divert less than 30 gpm water from the LPSI pumps' suction piping to the primary system drain tank. This condition would result in entering Palisades Nuclear Plant Off Normal Procedure ONP-17, "Loss of Shutdown Cooling," which directs operators to determine if the PSDT level is rising. If RV-0401 is determined to have lifted, the licensee states that "simple gagging of the relief valve would end the event."

The shutdown cooling system cannot be isolated during power operation or when there is fuel in the reactor vessel. The licensee states that RV-0401 can be tested only during full core off-loads. Given the information provided by the licensee on the low likelihood of a RV-0401 failure and on the emergency procedures in place to mitigate the consequences of such a failure, requiring a plant core off-load solely to perform setpoint testing for this valve would not result in a compensating increase in the level of quality and safety. The licensee's proposal provides a reasonable alternative to the Code requirements. To resolve any potential safety issues relative to inservice testing of RV-0401, Palisades has procedures in place to evaluate and incorporate data from NPRDS, industry experience, and plant data. If the licensee determines through the evaluation of the performance data that the testing interval is too long to assure the operational readiness of the valves, the relief request would have to be reexamined. If a full core off-load or isolation of shutdown cooling is not feasible to accommodate the appropriate interval, some

other method of assuring operational readiness must be provided, such as enhanced quality assurance and preventive maintenance.

7.4 Conclusion

The alternative testing frequency is authorized pursuant to 10 CFR 50.55a(a)(3)(ii) based on the hardship or unusual difficulty without a compensating increase in the level of quality and safety that would result if the requirement to perform setpoint testing every 24 months were imposed.

8.0 REVISED RELIEF REQUEST VRR NO. 31

Revised VRR No. 31 requests relief from OM-1, paragraph 1.3.4.1(b), regarding the test frequency for Class 2 and 3 pressure relief devices. This relief request pertains to valves RV-3162 and RV-3164 in the Shutdown Cooling System.

8.1 Licensee's Basis for Request

The licensee provided the following basis for the relief request:

> Relief valve RV-3162 has a safety function to provide overpressure protection for the Shutdown Cooling discharge header. Overpressure protection is required due to small amounts of back leakage from the Primary Coolant System. Relief valve RV-3164 has a safety function to provide overpressure protection for the Shutdown Cooling supply line. Overpressure protection is required during Plant heatup and failure in this operating scenario could render the line inoperable during plant cooldown.
>
> Relief is requested in accordance with 10 CFR 50.55a(f)(5)(iii) from the requirement to perform set point verification on a 48-month basis because such testing is impractical.
>
> The Palisades configuration uses one system for shutdown cooling service and low pressure safety injection. Relief valves RV-3162 and RV-3164 are flanged in place and are located in the discharge and supply lines for the Shutdown Cooling System.
>
> Testing cannot be performed with the reactor critical because removal of these valves from service would render more Low Pressure Safety Injection System components inoperable than allowed by Plant Technical Specifications.
>
> Testing during the period between cold shutdown and reactor critical requires the draining of a safety system and the removal of relief valves for set point testing. Testing cannot be performed during cold shutdown with shutdown cooling in service because these valves are located in non-redundant portions of the Shutdown Cooling System. Palisades has no alternate discharge paths for shutdown cooling and the relief valves cannot be isolated unless there is a full core off load. Based on this fact, RV-3162 and RV-3164 can only be tested during full core off loads.
>
> A historical review of available data of set point testing for RV-3162 and RV-3164, covering 71 months, was conducted. This review shows RV-3162 has remained within

set point tolerance for the entire 71 month period. This review also shows the set point for RV-3164 drifted from 300 psig to 285 psig during the same 71 month period. Even with set point drift, system operability and safety requirements were maintained with no reduction in safety margin. This portion of the Shutdown Cooling System is static except when providing core cooling during plant cold shutdowns. Therefore, RV-3162 and RV-3164 are not exposed to excessive system vibration or pressure fluctuations which contribute to valve wear and set point fluctuation. A maximum set point interval of 10 years is acceptable.

Due to the critical timing and sequencing of proceduralized test activities, the incorporation into surveillance test procedures of plant and Nuclear Plant Reliability Data System (NPRDS) valve performance data reviews to make adjustments to testing frequency or methods is not considered practicable. Rather, Palisades, on a much broader spectrum than plant data and NPRDS, reviews industry and operating experiences applicable to valve testing per its Engineering Manual and Administrative Procedures. These procedures ensure significant events that would affect valve testing frequencies and test methods are evaluated by appropriate plant personnel. The responsible engineer would then make any necessary revisions to test procedures.

A performance review of plant data and the NPRDS to recommend adjustments in testing frequencies and/or methods is not practicable within the scope of Palisades Technical Specification Surveillance Procedures. Technical Specification Surveillance Procedures provide plant personnel with step by step instructions to test plant equipment, and include test result acceptance criteria. Often, the timing and sequencing of proceduralized test activities are critical, with testing posing a potential impact on the operational status of other plant systems.

Completed Technical Specification Surveillance Procedures are reviewed by the Inservice Testing (IST) valve coordinator in accordance with the requirements of Palisades Nuclear Plant Engineering Manual Procedure EM-09-02, "Inservice Testing of Plant Valves." A Condition Report is initiated for any valve testing failure, and is evaluated in accordance with Palisades Administrative Procedure 3.03, "Corrective Action Process." The results of Corrective Action Evaluations may include adjustments to valve testing frequencies and/or test methods.

Palisades also has procedures in place to evaluate and incorporate data from NPRDS, Industry Experience, and plant data. Palisades Engineering Manual Procedure EM-20, "System Performance Monitoring," is intended to provide indications of system performance, system operations, component maintenance, and structure conditions that could contribute to inadequate system and structure performance. This procedure requires System Engineers to perform industry experience reviews for systems, structures, and components assigned to them. During the review, System Engineers determine if a Condition Report is needed to resolve potential safety issues. Any such finding is documented in a System Health Report and includes a discussion of any further evaluation and action taken.

An evaluation of a Condition Report requires a review of industry data to learn from previous similar instances or to identify causes which may not be apparent when considering information from one event. Industry data available includes the Nuclear Network, NPRDS, and the Palisades Industry Experience Data Base. In the future, Palisades will be utilizing INPO's successor to NPRDS, the "Equipment Performance Information Exchange (EPIX)" data base. EPIX is the data base used for industry failure reporting.

The likelihood of...RV-3162 or RV-3164 opening with Shutdown Cooling in Service due to setpoint drift is low during initial alignment of SDC to the PCS, and becomes extremely low as PCS pressure, temperature, and inventory are reduced.

The greatest probability for the relief valves actuating due to setpoint drift is when SDC is initially aligned to the Primary Coolant System (PCS). During this plant operating condition, PCS applied system pressure to SDC is procedurally allowed to be as high as 270 psia (255.3 psig).

A historical review of available data for setpoint testing of RV-3162 over a 71 month period shows the setpoint for this valve remained within ± 3% for the entire period. Therefore, the probability of RV-3162 drifting low from a setpoint of 500 psig to the calculated SDC system pressure at the relief valve of 404.0 psig is low. For the setpoint of RV-3162 to drift to 404.0 psig requires a 19.2% change.

A historical review of available data for RV-3164 shows the setpoint of this valve drifted from 300 psig to 285 psig over a 71 month period (5% change). For RV-3164 to actuate with SDC in service as describe[d] in the above paragraph, its setpoint would have to drift to a calculated pressure of 271.8. This would require a setpoint drift of 9.4% and is considered a low probability.

After initial alignment of the PCS to SDC, PCS temperature and pressure drop significantly below 270 psia and 300°F, and the likelihood of a relief valve lifting due to setpoint drift is further reduced. When the PCS is at reduced inventory, system pressures at the relief valves are:

RV-3162 < 190 psig	RV-3164 < 3 psig

At reduced PCS inventory the probability of a SDC relief valve lifting is extremely low for the following reasons:

1. For the plant to get to reduced PCS inventory conditions, the SDC relief valves must have remained seated during higher system operating pressures, such as system pressures experienced when SDC is initially aligned to the PCS.

2. It is overly conservative to postulate a setpoint drift greater than 60% of normal setpoint.

An emergency procedure is in place to mitigate the consequences of a Shutdown Cooling System relief valve opening due to setpoint drift with Shutdown Cooling in service. This is a low probability event, and the risk for this scenario is adequately addressed by the actions of Palisades Nuclear Plant Off Normal Procedure ONP-17, "Loss of Shutdown Cooling."

Palisades Nuclear Plant Off Normal Procedure ONP-17, "Loss of Shutdown Cooling," specifically addresses loss of Primary Coolant inventory in the unlikely event of...RV-3162 or RV-3164 opening with Shutdown Cooling in service. If the Quench Tank level is rising, ONP-17 directs Operators to do the following:

a. Verify Shutdown Cooling Relief Valve RV-3164 has not lifted.

b. If RV-3164 is lifting, then attempt to reseat the relief valve.

c. Verify Low Pressure Safety Injection Relief Valve RV-3162 has not lifted.

d. If RV-3162 is lifting, then attempt to reseat the relief valve.

Assuming...the relief valve lift is due to excessive setpoint drift, simple gagging would successfully stop the loss of PCS inventory. However, if diversion is not stopped, ONP-17 directs alternate PCS heat removal by using the Containment Spray Pumps, Spent Fuel Pool Cooling, or Steam Generators.

When Shutdown Cooling is in service, the greatest potential for a stuck open Shutdown Cooling system relief valve is when SDC is initially aligned to the PCS. In this plant condition, the consequences of a stuck open Shutdown Cooling system relief valve do not include loss-of-cooling-accident. At other SDC operating conditions, the potential of a stuck open SDC relief valve is significantly lower. However, if a SDC relief valve did lift and stick open, PCS make up from the Safety Injection and Refueling Water (SIRW) Tank would allow adequate time to implement Off Normal Operating Procedures and correct the problem as previously described.

The greatest probability of relief valves...RV-3162 or RV-3164 actuating and sticking open due to Shutdown Cooling (SDC) System pressure occurs when SDC is initially aligned to the Primary Cooling System (PCS). During this plant operating condition, PCS applied system pressure to SDC is procedurally allowed to be as high as 270 psia (255.3 psig). Assuming the relief valve(s) lift due to setpoint drift, premature actuation will result in a PCS loss-of-coolant until the PCS can be isolated from SDC and realigned to the Steam Generators per procedure ONP-17.

After initial alignment of the PCS to SDC, PCS temperature and pressure drop significantly below 270 psia and 300°F, and the likelihood of a relief valve lifting due to set point drift is further reduced. Should the unlikely event of a SDC system relief valve lifting and sticking open occur at these plant conditions, a loss-of-coolant accident would result. Assuming simultaneous full flow actuation of all three relief valves [RV-0401,

RV-3162, and RV-3164], make up water would be available from the SIRW Tank for approximately 15 hours. Also, ONP-17 effectively deals with a loss of Primary Coolant inventory conditions, by directing operators to determine the cause of the loss of Primary Coolant, and reseat the stuck open relief valve.

Recovery from a failure of...RV-3162, and RV-3164 is detailed in Palisades Nuclear Plant Off Normal Procedure ONP-17, "Loss of Shutdown Cooling." In each scenario, the event ends by either reseating or gagging the affected relief valve.

RV-3162 is a Teledyne Farris Engineering 26FB12-141 safety relief valve with a size "F" nozzle. It provides overpressure protection to the LPSI pumps discharge header from small amounts of PCS leakage. RV-3162 discharges to the containment sump through the containment floor drain system and is set at 500 psig.

Although the as found setpoint of RV-3162 has always been within tolerance, the question postulates a setpoint drift to 404.0 psig. If the setpoint of RV-3162 drifted to 404.0 psig, and the LPSI Header experienced an upset of 10% above 404.0 psig, then based solely on orifice size and not considering restrictions in the relief valve discharge piping, a stuck open RV-3162 could divert 101 gpm of water from the LPSI Header to the Containment Floor Drain. This condition would result in entering Palisades Nuclear Plant Off Normal Procedure ONP-17, "Loss of Shutdown Cooling." This procedure directs operators to determine if SDC relief valve RV-3162 has lifted. Once it has been determined RV-3162 has lifted, operators are directed to attempt to reseat the valve. If this fails, simple gagging of RV-3162 would end the event.

RV-3164 is a Teledyne Farris Engineering 26GB12-141 safety valve with a size "G" nozzle. It provides overpressure protection to the SDC suction line during plant heatup. RV-3164 discharges to the Primary System Drain Tank (PSDT), and is set at 300 psig.

Plant history shows RV-3164 has drifted over a 71 month period to 285 psig. Therefore, a setpoint drift to 271.8 psig for RV-3164 can be conservatively postulated. If the setpoint of RV-3164 drifted to 271.8 psig, and the SDC suction piping experienced an upset of 10% above 271.8 psig, a stuck open RV-3164 could divert 135 gpm of Primary Coolant from SDC suction piping to the PSDT. This is based solely on orifice size and not considering restrictions in the relief valve discharge piping. This condition would result in entering Palisades Nuclear Off Normal Procedure ONP-17, "Loss of Shutdown Cooling." This procedure directs operators to determine if the PSDT level is rising. If the PSDT is rising, then it is determined if SDC relief valve RV-3164 has lifted. Once it has been determined that RV-3164 has lifted, operators are directed to attempt to reseat the relief valve. If reseating is unsuccessful, simple gagging of the relief valve would end the event.

8.2 Proposed Alternate Testing

The licensee proposed the following alternative:

> Consumers Energy will verify RV-3162 and RV-3164 set points at least once every 10 years during full core off loads when testing has not been performed in the previous 48 months as required by Technical Specification Surveillance Procedure RT-116, "Miscellaneous Safety Systems Safety Valve Set point Testing."

8.3 Evaluation

OM-10, which references OM-1, requires testing of pressure relief devices that provide overpressure protection to components that function to shut down the reactor, maintain a safe shutdown condition, or mitigate the consequences of an accident. The valves identified in this relief request provide thermal overpressure protection for such components. For Class 2 and 3 pressure relief valves, OM-1, paragraph 1.3.4, requires a minimum of 20% of the valves to be tested within any 48 months. A valve in a group of two valves is required to be tested at least once within any 48-month interval on an alternating basis. In lieu of the Code requirements, the licensee proposes to setpoint test the relief valves at least once every 10 years during full core off-loads unless testing has been performed in the previous 48 months.

These relief valves are in a nonredundant portion of the shutdown cooling system. Testing these valves would result in a loss of system function; therefore, these valves can be tested only during a full core off-load. The licensee states that available information on the history of the valve over a 71-month period showed that system operability and safety requirements were maintained with no reduction in safety margins.

The licensee states that a LOCA could result if the valves in question lift and stick open when the shutdown cooling system is in service. Assuming simultaneous full flow actuation of the three relief valves (RV-0401, RV-3162, and RV-3164) in the line, makeup water would be available from the SIRW Tank for approximately 15 hours. The Palisades Nuclear Plant Off Normal Procedure ONP-17, "Loss of Shutdown Cooling," directs the operators in such a scenario to determine the cause of the loss of primary coolant and reseat the open relief valve.

The licensee states that in order for RV-3162 to lift when the shutdown cooling system is in service, the setpoint would have to drift from 500 psig down to 404 psig (a 19.2% change). For RV-3164, the setpoint would have to drift from 300 psig to 285 psig (a 9.4% change). The licensee states that less than 101 gpm of water would be diverted from the LPSI header to the containment floor drain following a stuck open RV-3162 and less than 135 gpm of water would be diverted from SDC suction piping to the PSDT following a stuck open RV-3164. These conditions would result in entering Palisades Nuclear Plant Off Normal Procedure ONP-17, "Loss of Shutdown Cooling." If RV-3162 or RV-3164 has lifted, the licensee states that "simple gagging of the relief valve would end the event" if attempts to reseat the valve are unsuccessful.

The shutdown cooling system cannot be isolated during power operation or when there is fuel in the reactor vessel. The licensee states that these valves can be tested only during full core off-loads. Given the information provided by the licensee on the low likelihood of failure and on the

emergency procedures in place to mitigate the consequences of such a failure, requiring a plant core off-load solely to perform setpoint testing for these valves would not result in a compensating increase in the level of quality and safety. The licensee's proposal provides a reasonable alternative to the Code requirements. To resolve any potential safety issues relative to inservice testing of RV-3162 and RV-3164, Palisades has procedures in place to evaluate and incorporate data from NPRDS, industry experience, and plant data. If the licensee determines through the evaluation of the performance data that the testing interval is too long to assure the operational readiness of the valves, the relief request would have to be reexamined. If a full core off-load or isolation of shutdown cooling can be isolated is not feasible to accommodate the appropriate interval, some other method of assuring operational readiness must be provided, such as enhanced quality assurance and preventive maintenance. The use of other methods would require review and approval by the NRC staff prior to implementation.

8.4 Conclusion

The alternative testing frequency is authorized pursuant to 10 CFR 50.55a(a)(3)(ii) based on the hardship or unusual difficulty without a compensating increase in the level of quality and safety that would result if the requirements to perform setpoint testing at least once within any 48-month intervals on an alternating basis were imposed.

9.0 RELIEF REQUEST VRR NO. 32

VRR No. 32 requests relief from OM-10, paragraph 4.3.2.1, which requires check valves to be individually exercised nominally every 3 months, except as provided by paragraphs 4.3.2.2, 4.3.2.3, 4.3.2.4, and 4.3.2.5. This relief request pertains to containment spray system check valves CK-ES3216 and CK-ES3226.

9.1 Licensee's Basis for Request

The licensee provided the following basis for the relief request:

> These eight inch check valves perform an active safety function in the open position to admit containment spray flow to the spray headers. These valves open when containment spray is automatically initiated on a Containment High Pressure (CHP) signal.
>
> These valves have no safety function in the closed position. Since the containment spray system will be in operation during an accident condition requiring containment integrity, these valves are not considered to be required for containment integrity. These valves function to prevent backflow to the shutdown cooling system during system testing.
>
> Relief is requested in accordance with 10 CFR 50.55a(a)(3) from disassembling and inspecting a sample (one) valve each refueling outage on the basis that the alternative proposed will provide an acceptable level of quality and safety.
>
> The result of a full-stroke exercise of the containment spray check valves would be the initiation of the containment spray system and spraying down containment. The original

Plant design did not provide a test circuit for these valves. As a result, full stroke testing is not practical during any mode of operation. Therefore, because of this limitation, Palisades has implemented disassembly and inspection, and part stroke testing in accordance with Generic Letter 89-04. This program requires one valve of the group composed of CK-ES3216 and CK-ES3226 to be disassembled and inspected each refueling outage, and quarterly part stroke testing when the plant is in cold shutdown. During periods where shutdown cooling is required, the disassembly would require a single valve isolation from the Containment Spray System using a valve with a air operator closure. Loss of control air to this single valve could result in a loss of shutdown cooling or reduced shutdown cooling effectiveness. Leakage past the valve could also present difficulties in completing valve inspections. Disassembly of the containment spray check valves also results in increased dose to plant staff on the order of 200mR.

Valve CK-ES3216 was disassembled and inspected during the 1994 refueling outage in accordance with work order 24416316. This was the first disassembly and inspection performed since initial plant installation and testing. The inspection indicated the valve was in good condition with little or no degradation after 23 years of service.

Valve CK-ES3226 was disassembled and inspected during the 1996 refueling outage in accordance with work order 24613344. This was the first disassembly and inspection performed since initial plant installation and testing. Inspection indicated the valve was in good condition with little or no degradation after 25 years of service.

A review of industry experience for this type of valve does not indicate any adverse performance experience associated with valves in mild service similar to the containment spray header check valves. The containment spray system is composed of stainless steel piping components rated at 500 psig and 350°F, and transports clean borated water. The spray header check valves do not see service conditions at any time other than system surveillance testing. No degradation mechanisms associated with system service or testing have been identified.

A search of Palisades specific experience indicates no corrective actions or work orders associated with these valves.

Due to the critical timing and sequencing of proceduralized test activities, the incorporation into surveillance test procedures of plant and Nuclear Plant Reliability Data System (NPRDS) valve performance data reviews to make adjustments to testing frequency or methods is not considered practicable. Rather, Palisades, on a much broader spectrum than plant data and NPRDS, reviews industry and operating experiences applicable to valve testing per its Engineering Manual and Administrative Procedures. These procedures ensure significant events that would affect valve testing frequencies and test methods are evaluated by appropriate plant personnel. The responsible engineer would then make any necessary revisions to test procedures.

A performance review of plant data and the NPRDS to recommend adjustments in testing frequencies and/or methods is not practicable within the scope of Palisades Technical Specification Surveillance Procedures. Technical Specification Surveillance

Procedures provide plant personnel with step by step instructions to test plant equipment, and include test result acceptance criteria. Often, the timing and sequencing of proceduralized test activities are critical, with testing posing a potential impact on the operational status of other plant systems.

Completed Technical Specification Surveillance Procedures are reviewed by the Inservice Testing (IST) valve coordinator in accordance with the requirements of Palisades Nuclear Plant Engineering Manual Procedure EM-09-02, "Inservice Testing of Plant Valves." A Condition Report is initiated for any valve testing failure, and is evaluated in accordance with Palisades Administrative Procedure 3.03, "Corrective Action Process." The results of Corrective Action Evaluations may include adjustments to valve testing frequencies and/or test methods.

Palisades also has procedures in place to evaluate and incorporate data from NPRDS, Industry Experience, and plant data. Palisades Engineering Manual Procedure EM-20, "System Performance Monitoring," is intended to provide indications of system performance, system operations, component maintenance, and structure conditions that could contribute to inadequate system and structure performance. This procedure requires System Engineers to perform industry experience reviews for systems, structures, and components assigned to them. During the review, System Engineers determine if a Condition Report is needed to resolve potential safety issues. Any such finding is documented in a System Health Report and includes a discussion of any further evaluation and action taken.

An evaluation of a Condition Report requires a review of industry data to learn from previous similar instances or to identify causes which may not be apparent when considering information from one event. Industry data available includes the Nuclear Network, NPRDS, and the Palisades Industry Experience Data Base. In the future, Palisades will be utilizing INPO's successor to NPRDS, the "Equipment Performance Information Exchange (EPIX)" data base. EPIX is the data base used for industry failure reporting.

9.2 Proposed Alternate Testing

The licensee proposed the following alternative:

Valves CK-ES3216 and CK-ES3226 will be part-stroke exercised quarterly each cold shutdown per QO-10.

Palisades proposes to extend the disassembly and inspection interval for CK-ES3216 and CK-ES3226, such that, both valves will be disassembled and inspected at full core off-loads or when shutdown cooling can be isolated, provided the valves have not been disassembled and inspected in the previous 40 months.

At each disassembly, the valves will be manually exercised to verify full-stroke capability. Also, the disassembled valve will be inspected to ensure the internals are structurally sound (no loose, damaged, or corroded parts).

CK-ES3216 and CK-ES3226 will be disassembled and inspected at least once every ten years.

9.3 Evaluation

These 8-inch containment spray check valves have a safety function in the open position to allow containment spray flow into the containment and have no safety function in the closed position. Full-flow exercise of these valves is impractical during any mode of operation because the initiation of the containment spray system would result in spraying down the containment. In lieu of the Code requirements (specified in Section 5.0, above), the licensee proposes to (1) part-stroke exercise the two containment spray check valves during cold shutdowns and (2) disassemble and inspect the valves at full core off-loads or when shutdown cooling can be isolated, provided the valves have not been disassembled and inspected in the previous 40 months.

The staff evaluated the licensee's proposal to perform inspection in accordance with GL 89-02, Position 2, which states that extension of the inspection interval beyond one valve every other refueling outage should only be considered in cases of extreme hardship where the extension is supported by actual in-plant data from previous testing. The licensee's proposed alternative is to perform inspection (1) during each full core off-load or during isolation of shutdown cooling, provided the valves have not been disassembled and inspected in the previous 40 months, and (2) at least every 10 years. Question Group 12 on GL 89-04 Position 2 (see NUREG-1482, Pp A-9 and 10) states that, in order to alter the inspection frequency, licensees should use the criteria in Position 2 to justify and to document the proposed disassembly schedule. The staff's response to Question Group 12 then states that the justification should address the significance of the loss of benefits of sampling in light of the condition, service history, and application of the valves. In response to a question on the definition of extreme hardship associated with the extension of a disassembly interval, the staff's response to Question Group 19 (see NUREG-1482, pp. 13 and 14) states that the existence of "extreme hardship" that would allow extension of the disassembly schedule is dependent on the particular circumstances at the plant. To determine whether extreme hardship exists, the staff's response to Question 19 states that the licensee should conduct a detailed evaluation of the various competing factors:

> First, the licensee should determine the effect on plant safety that would result for the proposed schedule extension. The maintenance history of the component and other information relevant to its reliability should be reviewed to determine whether the decrease in assurance of plant safety resulting from the schedule extension is justified. A need to off-load the reactor core, such as when testing the combined injection header check valves at some plants, or to operate at mid-level of the reactor coolant loops may be considered. The radiation exposure that would result from disassembly and inspection is a factor to be considered under the ALARA principle, but it should be judged in combination with all of the other factors.

With regard to the hardship associated with disassembly of these check valves, the licensee indicates that the disassembly would require a single valve isolation from the containment spray system using a valve with an air operator closure. During periods where shutdown cooling is

required, a failure of this air-operated valve during disassembly could result in a loss of shutdown cooling or reduced shutdown cooling effectiveness. The potential for a loss of shutdown cooling would constitute extreme hardship in this case. The licensee also stated that disassembly of the containment spray check valves results in an increased dose to plant staff on the order of 200mR. Further, a review of industry experience and actual in-plant data from previous testing identified no indication of adverse performance. To resolve any potential safety issues relative to inservice testing of CK-ES3216 and CK-ES3226, Palisades has procedures in place to evaluate and incorporate data from NPRDS, industry experience, and plant data.

On the basis of the above discussion, the staff finds that full-flow exercising of these check valves during any mode of operation is impractical. The licensee's proposed alternative is to (1) partial-stroke exercise during cold shutdowns, (2) perform inspection during each full core off-load or during isolation of shutdown cooling, provided the valves have not been disassembled and inspected in the previous 40 months, and (3) perform inspection at least every 10 years. The licensee's proposed deferral of testing on the basis of extreme hardship and actual plant data is consistent with GL 89-04, Position 2. Therefore the staff concludes that relief may be granted pursuant to 10 CFR 50.55a(f)(6)(I) because the proposed testing provides reasonable assurance of operational readiness and imposition of Code requirements would cause undue burden on the licensee. If the licensee determines through performance trends that the disassembly and inspection interval is too long to assure the operational readiness of the valves, the relief request would have to be reexamined. If a full core off-load or isolation of shutdown cooling is not feasible to accommodate the appropriate interval, some other method of assuring operational readiness must be provided, such as non-intrusive techniques and regular preventive maintenance. The use of other methods would require review and approval by the NRC staff prior to implementation.

9.4 Conclusion

Relief Request Number 32 is granted pursuant to 10 CFR 50.55a(f)(6)(I), based on the impracticality of performing IST on the individual components, and that the proposed alternative is authorized by law, will not endanger life or property, or the common defense or security, and is otherwise in the public interest, giving due consideration to the burden that would ensue if the Code requirements were imposed, and the adequacy of alternative testing for assessing the operational readiness of the valves.

10.0 REVISED RELIEF REQUEST PRR NO. 7

Relief Request PRR No. 7 was previously approved by the NRC safety evaluation dated August 30, 1996, pursuant to 10 CFR 50.55a(f)(6)(I). This request concerns the skid-mounted emergency diesel generators' diesel jacket water pumps. The safety evaluation recommended that relief request reference NUREG-1482, Section 3.4, which states that the testing of a major component is an acceptable means of monitoring the operational readiness of skid-mounted components if the licensee documents this approach in the IST program. In response to the safety evaluation, PRR No. 7 has been revised to include this recommendation. Therefore, the revised relief request remains approved as discussed in the referenced NRC safety evaluation.

11.0 CONCLUSION

The proposed alternatives in Revised Relief Requests VRR-7, 12, 28, 30, and 31 are authorized pursuant to 10 CFR 50.55a(a)(3)(ii) based on the hardship or unusual difficulty without a compensating increase in the level of quality and safety that would result if the Code requirements were imposed. Revised Relief Request PRR-7 and Relief Request VRR-32 are granted pursuant to 10 CFR 50.55a(f)(6)(I) based on the impracticality of performing inservice testing on the individual components, and that the proposed alternative is authorized by law, will not endanger life or property, or the common defense and security, and is otherwise in the public interest, giving due consideration to the burden that would ensue if the Code requirements were imposed, and the adequacy of alternative testing for assessing the operational readiness of the components. The proposed alternative in Revised Relief Request VRR-23 is authorized pursuant to 10 CFR 50.55a(a)(3)(I) based on the acceptable level of quality and safety provided by the alternative. For Revised Relief Request VRR-29, interim relief is authorized for a period of 120 days from the date of this safety evaluation pursuant to 10 CFR 50.55a(a)(3)(ii) based on the determination that immediate compliance with the specified requirements results in a hardship without a compensating increase in the level of quality and safety. This interim relief is to allow the licensee time to (1) provide basis for deferral of testing to cold shutdowns for the main steam atmospheric dump valves, (2) address non-intrusive techniques, and (3) address assigning reasonable, objective acceptance criteria to an observable parameter, such as a valve stem movement, flow rate or ΔP, to measure stroke time and assess degradation.

The staff concludes that the relief requests as evaluated and modified by this safety evaluation will not compromise the operational readiness of the pumps and valves to perform their safety functions. The staff has determined that granting of the relief requests and authorization of the proposed alternatives to the Code requirements pursuant to 10 CFR 50.55a are authorized by law and will not endanger life or property, or the common defense and security and are otherwise in the public interest. In making this determination, the staff has considered the impracticality of meeting the requirements and the burden on the licensee if the requirements were imposed.

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