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Palisades Nuclear Plant
Operated by Nuclear Management Company, LLC

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10 CFR 50.59

U.S. Nuclear Regulatory Commission
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Palisades Nuclear Plant
Docket 50-255
License No. DPR-20

Report of Facility Changes, Tests and Experiments and Summary of Commitment Changes

Nuclear Management Company, LLC, is providing the Palisades Nuclear Plant (PNP) Report of Facility Changes, Tests and Experiments for the time period of September 30, 2003 through September 30, 2005. This report is being submitted in accordance with the requirements of 10 CFR 50.59(d)(2). Also included is a summary of revised regulatory commitments as required by Nuclear Energy Institute (NEI) Guideline NEI 99-04, "Guidelines for Managing NRC Commitment Changes," Revision 0, endorsed by the Nuclear Regulatory Commission (NRC) in Regulatory Issue Summary 2000-17, "Managing Regulatory Commitments Made by Power Reactor Licensees to the NRC Staff."

Attachment 1 contains a brief description of changes to the facility and a summary of the evaluation, performed in accordance with 10 CFR 50.59. There were no changes made under 10 CFR 72.48 during this period.

Attachment 2 contains summaries of regulatory commitment changes requiring NRC notification, including justification for the change.

Summary of Commitments

This letter contains no new commitments and no revisions to existing commitments.

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Enclosures (2)

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ENCLOSURE 1

NUCLEAR MANAGEMENT COMPANY, LLC

PALISADES NUCLEAR PLANT
DOCKET 50-255

OCTOBER 19, 2005

REPORT OF FACILITY CHANGES, TESTS AND EXPERIMENTS

7 Pages Follow

SDR Review Number: 04-1438 **Document Number:** EA-GOTHIC-04-08

Title: Containment Response to a LOCA Using GOTHIC 7.1 (PATCH 1)

Activity Description

This engineering analysis evaluates the containment response to a loss of coolant accident (LOCA). The analysis calculates maximum component cooling water temperatures downstream of the component cooling water heat exchangers that exceed the design temperature of the component cooling water system. In addition, some changes were made to the inputs for this analysis which are considered to be non-conservative. Both of these items are required to be reviewed in a 50.59 evaluation.

Summary of 50.59 Evaluation

The 50.59 evaluation concluded that prior NRC approval is not required for this activity. The component cooling water system temperatures calculated in this analysis and the non-conservative input parameter changes will not result in more than a minimal increase in the likelihood of occurrence of a malfunction of a structure, system, or component important to safety. The component cooling water system will continue to be capable of performing its functions. The design temperatures of individual system components are not exceeded and the elevated temperature is not a significant contributor to system pipe stress. The analysis concludes that containment remains within design limits, so the input parameter changes will not increase the consequences of an accident previously evaluated in the FSAR and will not result in a design basis limit for a fission product barrier being exceeded.

SDR Review Number: 04-1439 **Document Number:** EA-GOTHIC-04-09

Title: Containment Response to a MSLB Using GOTHIC 7.1 (PATCH 1)

Activity Description

This engineering analysis evaluates the containment response to a main steam line break. The analysis calculated a maximum containment liner temperature (268°F) that is slightly in excess of the maximum analyzed containment liner temperature (266°F) and some changes were made to the inputs for this analysis which are considered to be non-conservative. Both of these items are required to be reviewed in a 50.59 evaluation.

Summary of 50.59 Evaluation

The 50.59 evaluation concluded that prior NRC approval is not required for this activity. The slightly elevated calculated containment liner temperature does not impact the ability of the containment structure to perform its design function. This calculated temperature has a negligible impact because this is a secondary, local expansion stress that is self-limiting and it occurs in a component (the containment liner) that is not a pressure boundary. Moreover, the containment concrete structure will not be affected. Therefore, this will not increase the consequences of an accident previously evaluated in the FSAR and does not result in a design basis limit for a fission product barrier being exceeded. The analysis concludes that containment remains within design limits, so the input parameter changes will not increase the consequences of an accident previously evaluated in the FSAR and will not result in a design basis limit for a fission product barrier being exceeded.

SDR Review Number: 05-0099 **Document Number:** EA-GOTHIC-04-09

Title: Containment Response to a MSLB Using GOTHIC 7.1 (PATCH 1)

This 50.59 evaluation applies to EA-GOTHIC-04-09 and its associated changes to the UFSAR. EA-GOTHIC-04-09 evaluates the containment response to a main steam line break (MSLB) using the GOTHIC 7.1 (Patch 1) computer program. The changes to EA-GOTHIC-04-09 between Revisions 0 and 1 are summarized below:·

1. Containment spray flow rates and start times are changed. Some changes increase flow rates and initiate flow earlier. Some changes reduce flow rates and initiate flow later.

2. An operator action is incorporated to reduce the number of operating containment spray pumps to one when containment pressure falls to 20 psig. This proposed operator action is intended to reduce the potential for containment sump screen blockage. (It should be noted that any actual change to operator actions would be addressed in the 50.59 screen associated with a procedure change.) The effect of the change is slower containment temperature and pressure reduction.·

3. The modeling of recirculation mode operation is eliminated. This is consistent with the past modeling of the Main Steam Line Break. Because of this change, containment spray is secured once the RAS setpoint is reached if containment pressure has not fallen to the criterion for securing containment spray. The effect is a potential reduction in post-RAS containment heat removal capability (containment air coolers only).

4. During a containment walkdown, the primary system drain tank was found to be covered with insulation. Because of this, the primary system drain tank heat conductor is removed from the main steam line break models.

While containment peak pressure and environmental qualification temperature profile limits are not exceeded, the analysis calculates a maximum containment liner plate temperature of 268°F, which exceeds the maximum analyzed surface temperature of 266°F (UFSAR Section 5.8.4.1.2). The containment liner plate surface temperature in excess of the maximum analyzed temperature of 266°F does not impact the ability of the containment structure to perform its design function. Consequently, the proposed activity does not result in more than a minimal increase in the likelihood of occurrence of a malfunction of an SSC important to safety, does not create the possibility for an accident of a different type or a malfunction of an SSC with a different result than that evaluated in the FSAR, and does not result in a design basis limit for a fission product barrier being exceeded or altered.

SDR Review Number: 04-1348 **Document Number:** EAR-2003-0035

Title: Modification Package for Cycle 18 Reload Design

The reactor core is being refueled for Cycle 18. The Cycle 18 core consists of 60 fresh batch "V" assemblies, 64 once-burnt batch "U" assemblies, 56 twice-burnt batch "T" assemblies, 16 thrice-burnt batch "S" assemblies, 4 batch "U" shield assemblies and 4 SAN (Shield Assembly - N) assemblies.

The replacement of fuel does not increase the frequency of occurrence of accidents previously evaluated in the FSAR.

The mechanical design of the Batch V fuel is basically the same as the previous reloads. Vendor analyses have determined that the mechanical, thermo hydraulic and seismic design of the new fuel is acceptable. The neutronic design has been analyzed and found acceptable for the new fuel in the reactor, fuel pool, new fuel storage, fuel elevator and the transfer carriage. The weight of the new Batch V fuel is essentially identical to that of the previous Batch U fuel and therefore it will properly interface with the fuel handling equipment. As a result, there is no more than a minimal increase in the likelihood of occurrence of a malfunction of the fuel or its interfacing equipment previously analyzed in the FSAR.

The consequences for all the accidents analyzed in the FSAR have been reviewed or reanalyzed and found to be acceptable. Cycle 18 does not result in more than a minimal increase in the consequences of an accident previously evaluated in the FSAR. The consequences of a malfunction in the fuel are bounded by the consequences of an accident previously evaluated in the FSAR. There is no more than a minimal increase in the consequences of a malfunction of the fuel.

The Cycle 18 fuel and loading is similar and consistent to that of Cycle 17. The possibility of an accident of a different type or of a malfunction of the fuel with a different result than that previously analyzed in the FSAR is not created.

During normal operation and anticipated operational occurrences (AOO), the Cycle 18 core will not cause a design basis limit for a fission product barrier to be exceeded or affected. Limiting faults such as the Control Rod Ejection (CRE) Accident and the LOCA do result in the failure of the fuel cladding, however, this is acceptable for a fault. The consequences of these failures are bounded by the consequences previously analyzed in the FSAR.

The methods of evaluations used for the new cycle are consistent with those described in the FSAR and previously approved methods. NRC approval has previously been received for the use of statistical rather than deterministic verification of the Thermal Margin/Low Pressure limiting safety system setting and the inlet temperature LCO functions.

Therefore, prior NRC approval is not required for the Cycle 18 reload design.

SDR Review Number: 04-1628 **Document Number:** EAR-2001-0518

Title: Evaluate Current Mechanical Stop on New Fuel Elevator for Design Functions

Activity Description

The New Fuel Elevator (H-14A) has a mechanical stop whose sole function is to prevent raising an irradiated fuel bundle within 10 feet of the water surface. Besides the mechanical stop, the New Fuel Elevator is equipped with dual limit switches in the electrical control logic to perform this same function. Furthermore, the Fuel Elevator is manually raised and lowered and two manual controls must be actuated to move the elevator. The subject EAR eliminates the mechanical stop and relies on these other control methods to perform the function.

Background

An attempt at eliminating the mechanical stop was made during a previous major modification of the New Fuel Elevator. Specification Change SC-91-095 intended to eliminate the mechanical stop on the grounds that a redundant upper limit switch was being added and the elevator control console was being enhanced with state-of-the-art integrated circuits. A PRC meeting concerning the issue was conducted (PRC-92-024), which concluded that the mechanical stop should remain in place since there was not adequate redundancy in the dual limit switches because of shared mechanical devices including a cam. A single mechanical failure could render both switches inoperable.

The current 50.59 evaluation is based on the same design as that for the 50.59 evaluation for SC-91-095. The difference between the evaluations is that the new evaluation credits the manual actions of the operator and the associated administrative controls. To move the elevator, a foot switch and a joy stick must be actuated simultaneously. Bypassing the limit switch interlocks requires a key that is controlled by the control room. These controls together with the dual limit switches, which are tested daily during fuel handling operations, provide equivalent protection against an inadvertent lifting of a fuel bundle as a single limit switch and a mechanical stop.

Summary of 50.59 Evaluation

The subject change does not introduce the possibility of a change in the frequency of an accident because the removal of the mechanical stop is not an initiator of any accident and no new failure modes are introduced.

A review of the fuel elevator design has concluded that there is sufficient redundancy in the electrical interlock design that the level of protection provided by the switches and the manual operator actions involved in raising the elevator is similar to the level of protection provided by a single limit switch and the mechanical stop. There is no more than a minimal increase in the likelihood of occurrence of a malfunction of the equipment such that the function of submergence is not met.

The consequences of a fuel handling accident or the consequences of a malfunction of equipment on the new fuel elevator that would lead to a fuel handling accident are not increased. The drop of a fuel bundle is the bounding accident and is not affected by this change. This change has no effect on existing radiological barriers.

The fuel handling accident and the loss of the submergence design function are the only credible outcomes of a malfunction of equipment or misoperation of the elevator. The fuel handling accident has been analyzed in the FSAR. The loss of the submergence function has been found to have less than a minimal increase in the likelihood of occurrence. No new accidents or malfunctions with a different result than those stated in the FSAR are created. Furthermore, the design basis limits for the fuel cladding are not affected.

Finally, the change does not involve a new analysis methodology or change to an existing methodology described in the FSAR.

Based on this 50.59 evaluation, this change may be implemented without prior NRC approval.

SDR Review Number: 05-0568 **Document Number:** FC-978 50.59 EVAL

Title: Revision to 50.59 Evaluation for Closeout of FC-978, Cycle 17 Reload

This evaluation is an update to the FC-978 revision 0 evaluation contained in SDR-02-0536, the FC-978 revision 1 evaluation contained in SDR-03-0377, the FC-978 revision 2 evaluation contained in SDR-03-0702 and the FC-978 revision 3 evaluation contained in SDR-03-1002. The updated evaluation identifies that the reload is designed to operate at the Appendix K power uprate calculated for FC-977. The conclusion and the bases for the previous evaluations have not changed.

SDR Review Number: 04-0777 **Document Number:** SCD-PL-0080

Title: Software Classification for GOTHIC 7.1

This 50.59 evaluation approves the use of the computer code GOTHIC 7.1 (PATCH 1) in loss of coolant accident and main steam line break containment response analyses at Palisades. This computer code was previously approved by the NRC for these applications in an SER written for the Kewaunee Nuclear Power Plant. Guidance for the performance of 50.59 evaluations, provided in the NRC-endorsed NEI 96-07, Revision 1, allows plants to take credit for SERs that approve the use of methodologies at other plants as long as the terms, conditions, and limitations for its use are met. The 50.59 evaluation written for Palisades' use of the GOTHIC code documents that Palisades will use the GOTHIC code in a manner that complies with the terms, conditions, and limitations contained in the Kewaunee SER. Therefore, this code may be used at Palisades under 50.59 without prior NRC approval.

SDR Review Number: 05-0239 **Document Number:** COP-1

Title: Primary Coolant System Chemistry

This evaluation documents the justification for the conclusion that increasing the silica concentration limit below which zinc injection is permitted (from 1.5 ppm to 1.79 ppm) will result in a less than minimal increase in the (1) frequency of an accident, (2) likelihood of a malfunction of a System, Structure or Component (SSC) important to safety, (3) consequences of an accident, and (4) consequences of a malfunction of a SSC important to safety. Likewise, this activity will not create or result in the (1) possibility of a different type of accident, (2) possibility of a SSC malfunction with a different result, (3) Design Basis Limit for a Fission Product Barrier being altered or exceeded, and (4) departure from a method of evaluation in design bases or safety analyses. The rational for the above conclusion was based on correspondence from Framatome ANP (our fuel vendor) and the Siemens Safety Evaluation used to support the installation of the zinc injection system. Specifically, these references are:1) Framatome Letter entitled, "Approval to Proceed with Zinc Addition at Palisades", dated February 17, 2005, and 2) Framatome letter entitled, "Transmittal of Document 51-5037175-00, Evaluation of Palisades EOC15 Fuel CDUD Sample Analysis Results", dated February 13, 2004.3) Siemens Work Report KWU NW-C/99/E004, "Safety Evaluation for Primary System Zn Injection at Palisades NPP". The conclusion drawn from the referenced documents is that there will not be a more than minimal increase in deposition of material on the fuel (CRUD and/or Zeolites) and that there will not be a more than minimal impact on the performance of the fuel cladding.

SDR Review Number: 05-0559 **Document Number:** TM-2005-013

Title: Disable Steam Generator "B" Level Transmitter LT-004

This evaluation applies to disabling of the high level override feature of LT-0704, Steam Generator E-50B Level Transmitter, per Temporary Modification TM-2005-013. It also applies to the related changes being made to the following procedures:
1) SOP-34 rev 18, Palisades Plant Computer (PPC) System;
2) ONP-10 rev 6, Excessive Feedwater Increase;
3) ONP-3 rev 20, Loss of Main Feedwater; and
4) ARP-5 rev 67, Primary Coolant Pump Steam Generator and Rod Drives Scheme EK-09 (C-12).

The condition which initiated this modification and associated procedure changes is described in CAP048578, Work Request Submitted For LT-0704 Steam Generator E-50B Level Transmitter. That document describes a condition where Palisades Plant Computer (PPC) indication for Steam Generator "B" level was low by 4% compared to other indications of 'B' Steam Generator level and the level in "A" Steam Generator. The cause was determined to be a failure of transmitter LT-0704 on the "B" Steam Generator.

The modification will disconnect level transmitter LT-0704 and associated instruments from their power supply. A high and low level alarm as well as a high level override signal will be lost. Measures will be taken to restore lost functions by addition of an alarm on the PPC to annunciate "B" Steam Generator high or low level. The new alarm from the PPC will use as its source "B" Steam Generator level transmitter LT-0752A. Plant procedures will be revised to incorporate the new alarm from the PPC and to delete the presence of the high level override signal from the "B" Steam Generator.

The proposed activity involves SSCs that are very similar in form, fit and function to the previously used SSCs. They will be able to deal with the accidents or transients associated with the liquid inventory in the steam generators. These are the Loss of Main Feedwater and the Excessive Feedwater Increase events. The major difference between the proposed activity and the existing situation is that there will be no high level override action available for the "B" steam generator. However, there is no credit taken for this action in the safety analyses. Moreover, the procedures that exist prior to installation of the Temporary Modification, and the procedures that will exist after installation, both contain the same manual actions to cope with a high level condition for the Excessive Feedwater Increase event. As a result, the proposed activity will not result in more than a minimal increase in the likelihood of occurrence or the consequences of an accident or malfunction of an SSC important to safety previously evaluated in the UFSAR. A new type of accident is not created. A malfunction of a SSC with a different result is not created.

The presence or lack of presence of the high level override signal will not affect the performance of a fission product barrier during an accident, and will not result in exceeding or altering a design basis limit for a fission product barrier as described in the UFSAR.

ENCLOSURE 2

NUCLEAR MANAGEMENT COMPANY, LLC

PALISADES NUCLEAR PLANT
DOCKET 50-255

OCTOBER 19, 2005

SUMMARY OF COMMITMENT CHANGES

3 Pages Follow

COMMITMENT NUMBER	SOURCE DOCUMENT/DATE	COMMITMENT DESCRIPTION	REVISED COMMITMENT	JUSTIFICATION
2000319	Response to NRC Request for Additional Information on Proposed Administrative Changes	Implement an administrative program to assure an appropriate level of management approval is required when operating conditions are changed from cold shutdown to hot shutdown and from hot shutdown to critical in hot standby with inoperable equipment. The program will consist of requiring a Plant Review Committee (PRC) review and approval of all upward operational condition changes made under Technical Specification 3.0.4.	Commitment is being removed from ongoing status.	The commitment was made as an alternative to continuing with a proposed Technical Specification Change Request that had been previously submitted to add the requirement to the PRC responsibilities, which were included in the Technical Specifications (6.5.1.6) at the time. The commitment was made to address an NRC concern regarding limiting the use of Technical Specification 3.0.4. Since that time, Technical Specification 3.0.4 has been revised to now require a risk assessment before the technical specification may be applied. The risk assessment assures that entry into Technical Specification LCO 3.0.4 will not be made without appropriate administrative controls.
1012729	Response to confirmatory action letter and information request pursuant to 10CFR50.54(f) dated 5/21/86 and 11/20/86	Enhancements for handling consistency of design input. Clarify controls for engineering or design work done by outside plant groups.	Commitment is being removed from ongoing status.	The Facility Change (FC) process has been replaced by the Nuclear Management Company (NMC) fleet modification process. The NMC Quality Assurance Topical Report (NMC-1) commits to implementation of ASME NQA-1-1994, "Quality Assurance Requirements for Nuclear Facility Applications." NMC procedures for modifications comply with requirements of NQA-1, Basic requirement 3, Design Control.
1012727	Response to confirmatory action letter and information request pursuant to 10CFR50.54(f) dated 5/21/86 and 11/20/86	Enhancements for handling consistency of design input. More detailed identification of design input is needed. Rewrite admin section 9.	Commitment is being removed from ongoing status.	The FC process has been replaced by the NMC fleet modification process. The NMC Quality Assurance Topical Report (NMC-1) commits to implementation of ASME NQA-1-1994, "Quality Assurance Requirements for Nuclear Facility Applications." NMC procedures for modifications comply with requirements of NQA-1, Basic requirement 3, Design Control.
1012728	Response to confirmatory action letter and information request pursuant to 10CFR50.54(f) dated 5/21/86 and 11/20/86	Enhancements for handling consistency of design input. More detailed identification of design input is needed.	Commitment is being removed from ongoing status.	The FC process has been replaced by the NMC fleet modification process. The NMC Quality Assurance Topical Report (NMC-1) commits to implementation of ASME NQA-1-1994, "Quality Assurance Requirements for Nuclear Facility Applications." NMC procedures for modifications comply with requirements of NQA-1, Basic requirement 3, Design Control.
1012730	Response to confirmatory action letter and information request pursuant to 10CFR50.54(f) dated 5/21/86 and 11/20/86	Enhancements for handling consistency of design input. Review NODS to determine if documentation to support operability determination needs filing with facility change package.	Commitment is being removed from ongoing status.	Nuclear Operations Department Standards (NODS) have been deleted. The facility change process has been replaced by the NMC fleet modification process. The NMC Quality Assurance Topical Report (NMC-1) commits to implementation of ASME NQA-1-1994, "Quality Assurance Requirements for Nuclear Facility Applications." NMC procedures for modifications comply with requirements of NQA-1, Basic requirement 3, Design Control.

COMMITMENT NUMBER	SOURCE DOCUMENT/DATE	COMMITMENT DESCRIPTION	REVISED COMMITMENT	JUSTIFICATION
1012731	Response to confirmatory action letter and information request pursuant to 10CFR50.54(f) dated 5/21/86 and 11/20/86	Enhancements for handling consistency of design input. Review responsibilities of project engineer during installation phase.	Commitment is being removed from ongoing status.	The FC process has been replaced by the NMC fleet modification process. The NMC Quality Assurance Topical Report (NMC-1) commits to implementation of ASME NQA-1-1994, "Quality Assurance Requirements for Nuclear Facility Applications." NMC procedures for modifications comply with requirements of NQA-1, Basic requirement 3, Design Control.
1012366	Response to confirmatory action letter and information request pursuant to 10CFR50.54(f) dated 5/21/86 and 11/20/86	Continue to monitor V-2A fan performance prior to start up.	Commitment is being removed from ongoing status.	The original intent of this commitment was to monitor V-2A fan vibration and stator temperature utilizing temporary instrumentation during plant restart after shutdown mandated by NRC Confirmatory Action Letter issued on May 21, 1986. Fan performance was monitored during and after plant restart and performance was found to be acceptable. Monitoring is no longer required.
1012460	Response to confirmatory action letter and information request pursuant to 10CFR50.54(f) dated 5/21/86 and 11/20/86 – Additional Info Request	P-15B, P-15C, P-15D, P-15F, P-15G - chemical addition pumps. Maintenance on pumps will be monitored and diaphragm replaced yearly.	The chemical addition pumps will be replaced every 10 years.	Over the years, it was found that the reliability of the chemical addition pumps was very high and that performing preventive maintenance and replacing diaphragms annually caused the pumps to begin having problems. The action has been changed to replace the pumps every ten years. The interval was determined based on pump reliability. Each pump has a manual backup. On failure, system alarms notify personnel that a problem exists so that the backup pump may be started.
1012465	Response to confirmatory action letter and information request pursuant to 10CFR50.54(f) dated 5/21/86 and 11/20/86 – Additional Info Request	Generic Issue - Containment Isolation Valves (Cmt# 129): LLRT Program - Implement An Augmented LLRT Program Which Will Increase Frequency For Testing All Penetration Valves.	Commitment is being removed from ongoing status.	Technical Specification SR 3.6.1.1 requires that containment leakage rate testing of valves be performed in accordance with the containment leak rate testing program. Technical Specification ADMIN 5.5.14, Containment Leak Rate Testing Program, requires that a testing program be established in accordance with 10 CFR 50, Appendix J, Option B. 10 CFR 50, Appendix J, Option B is a performance based testing standard that required tracking of the valve's performance and establishes test frequencies based on the valve's leak rate performance. In accordance with the requirements each valve has an administrative leak rate limit associated with it which, when exceeded, requires an evaluation of the condition to ensure the appropriate action is taken. Also, in accordance with the requirements the current and historical test results for each valve are evaluated against the administrative limit to establish required test frequencies. Therefore, the requirements of Palisades' Technical Specification ensures that valves with significantly degraded leakage rates are identified, and that valve leak rate performance is used to establish the correct testing frequency.

COMMITMENT NUMBER	SOURCE DOCUMENT/DATE	COMMITMENT DESCRIPTION	REVISED COMMITMENT	JUSTIFICATION
1012466	Response to confirmatory action letter and information request pursuant to 10CFR50.54(f) dated 5/21/86 and 11/20/86 – Additional Info Request	Containment isolation valves: LLRT program - develop valve trending program to track valve performance for identification of degradation prior to valve failing leak rate.	Commitment is being removed from ongoing status.	Technical Specification SR 3.6.1.1 requires that containment leakage rate testing of valves be performed in accordance with the containment leak rate testing program. Technical Specification ADMIN 5.5.14, Containment Leak Rate Testing Program, requires that a testing program be established in accordance with 10 CFR 50, Appendix J, Option B. 10 CFR 50, Appendix J, Option B is a performance based testing standard that required tracking of the valve's performance and establishes test frequencies based on the valve's leak rate performance. In accordance with the requirements each valve has an administrative leak rate limit associated with it which, when exceeded, requires an evaluation of the condition to ensure the appropriate action is taken. Also, in accordance with the requirements the current and historical test results for each valve are evaluated against the administrative limit to establish required test frequencies. Therefore, the requirements of Palisades' Technical Specification ensures that valves with significantly degraded leakage rates are identified, and that valve leak rate performance is used to establish the correct testing frequency.
2010559	Response To NRC 10CFR50.54(f) Letter: Request fFor Information Pursuant to 10CFR50.54(f) Regarding Adequacy And Availability Of Design Bases Information	Update and re-institute use of Quality Assurance Requirements Matrix (QARM) database by the end of 1998.	Commitment is being removed from ongoing status.	The Quality Requirements Matrix is no longer applicable. NMC has developed a matrix that fleet sites may use to reference quality requirements. Additionally, qualified review and plant operations review committee ensure quality requirements are met.
2011097	IEB 80-10 Response - Contamination Of Non-Radioactive System And Resulting Potential For Release Of Radioactivity To Environment	The air receiver tanks (service air) T-8A, T-8B and T-8C will be sampled and gamma analysis performed weekly. This will be implemented by 10/1/80 due to potential complications in ability to obtain a sample.	The air receiver tanks (service air) T-8A, T-8B and T-8C will be sampled and gamma analysis performed quarterly.	All three air receiver tanks have been sampled/analyzed on a weekly basis since the early 1980's. Never in that time frame, has radioactivity been detected in the samples collected.