NMC
Committed to Nuclear Excellence

Palisades Nuclear Plant
Operated by Nuclear Management Company, LLC

December 16, 2005

10 CFR 54

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

Palisades Nuclear Plant
Docket 50-255
License No. DPR-20

Supplemental Responses to NRC Questions Relating to License Renewal

In a telephone conference call on November 15, 2005, the NRC raised follow up questions about previous Nuclear Management Company (NMC) responses to two NRC Requests for Additional Information (RAIs), and asked one new question concerning the License Renewal Application. This letter responds to those verbal questions.

In a letter dated October 31, 2005, NMC provided a description of the Palisades ASME Section XI IWB, IWC, IWD, IWF Inservice Inspection Program. This letter closed a commitment made in a letter dated August 25, 2005. During the November 15 call, additional guidance was received on the preferred definition and treatment of exceptions to NRC's description of the ASME IWB, IWC, IWD Inservice Inspection Program in NUREG 1801, Generic Aging Lessons Learned Report. The Palisades program description has been revised to be consistent with this updated guidance, and is provided in Enclosure 1 as an updated response to the August 25, 2005, commitment.

In addition, follow up questions were raised about the NMC response to RAI 3.5.2-4-1(b) in a letter dated July 28, 2005. An expanded response to RAI 3.5.2-4-1(b), which responds to the reviewer's questions, is provided in Enclosure 2.

A question was also raised about the License Renewal Application Section 4.3.12. Supplementary information concerning LRA Section 4.3.12, which responds to the reviewer's question, is provided in Enclosure 3.

Finally, NMC has noted an editorial error in its response to NRC RAI 4.5.2(b) contained in a letter dated October 28. 2005. Enclosure 1 of that letter provided Revision 1 of a vendor report entitled, "30th Year Tendon Surveillance At The Palisades Nuclear Plant." The vendor's summary comparison of the 30th year test results with the original installation data, provided on page 4 of Enclosure 1, contained two superseded numbers. In the third paragraph of this summary, the vertical tendon 40-year projection value of 666 kips should read 670 kips (correct value was provided on page 7 of

Enclosure 1), and the horizontal tendon 40-year projection value of 647 kips should read 646 kips (correct value was provided on page 9 of Enclosure 1). These corrections have no impact on the overall conclusions in the summary.

Please contact Mr. Robert Vincent, License Renewal Project Manager, at 269-764-2559, if you require additional information.

Summary of Commitments

This letter contains no new commitments or revisions to existing commitments

I declare under penalty of perjury that the foregoing is true and correct. Executed on December 16, 2005.

Paul A. Harden
Site Vice President, Palisades Nuclear Plant
Nuclear Management Company, LLC

Enclosures (3)

CC Administrator, Region III, USNRC
Project Manager, Palisades, USNRC
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ENCLOSURE 1

Updated NMC Response to August 25, 2005, Commitment Regarding ASME Section XI IWB, IWC, IWD, IWF Inservice Inspection Aging Management Program

(5 Pages)

August 25, 2005, Commitment Regarding ASME Section XI IWB, IWC, IWD, IWF Inservice Inspection Aging Management Program

NMC Letter Dated August 25, 2005 stated,

NMC will revise the ASME Section XI IWB, IWC, IWD, IWF Aging Management Program descriptions in LRA Appendices A and B to reflect the 2001 edition including the 2002 and 2003 addenda of ASME Section XI. The revised program descriptions will identify exceptions to this code taken by the program, if any, that impact aging management effectiveness. Appropriate justification will also be provided to show that the exceptions, if any, still provide an acceptable level of aging management. The revised program descriptions will be submitted for NRC review and approval by October 31, 2005. [NMC Tracking No. 82]

NMC provided the revised program description in a letter dated October 31, 2005.

Updated NMC Response to August 25, 2005, Commitment Regarding ASME Section XI IWB, IWC, IWD IWF Inservice Inspection Aging Management Program

The NMC response to this commitment provided in a letter dated October 31, 2005, is hereby replaced in its entirety with the following:

The NUREG 1801 Revision 1, Section XI.M1, XI.M3 and XI.S3 programs, reference the 2001 edition of ASME B&PV Code, Section XI, as providing an acceptable basis for an aging management program. NMC has concluded that the ASME Section XI IWB, IWC, IWD, IWF Inservice Inspection Program for aging management should be revised to reflect the 2001 edition through 2003 addenda as the Section XI code of record. Accordingly, the following changes are hereby made to the description of the ASME Section XI IWB, IWC, IWD, IWF Inservice Inspection Program provided in LRA Section B2.1.2 on pages B-17 through B-25.

On page B-17 under the heading **Program Description**, replace the second paragraph in its entirety with the following,

> "The Palisades ASME Section XI IWB, IWC, IWD, IWF Inservice Inspection aging management program is based on the ASME B&PV Code, Section XI, 2001 edition, including 2002 and 2003 addenda. The IWB-2500 Category B-Q requirements to perform volumetric examinations of steam generator tubes is addressed by the Steam Generator Tube Integrity Program.
>
> During the period of extended operation, the program will be maintained consistent with the requirements of 10 CFR 50.55a. 10 CFR 50.55a currently requires that inservice inspection of Class 1, 2, and 3 pressure retaining components, their integral attachments, and supports, be conducted in

accordance with the latest edition of ASME Section XI approved by the NRC twelve months prior to the start of a ten-year interval. 10 CFR 50.55a also provides for the use of NRC-approved relief requests. Therefore, any relief requests or other alternatives to the ISI code of record would be submitted for NRC review and approval at least twelve months prior to the start of each inspection interval in accordance with existing regulations."

On page B-19, under the heading **NUREG-1801 Consistency**, replace the existing paragraph in its entirety with,

"The ASME Section XI IWB, IWC, IWF Inservice Inspection Program is consistent with, but contains an exception to, NUREG-1801, Section XI.M1, "ASME Section XI Inservice Inspection, Subsections IWB, IWC and IWD." This program is consistent with NUREG-1801, Section XI.M3, "Reactor Vessel Closure Studs," and Section XI.S3, "ASME Section XI, Subsection IWF."

On page B-19, under the heading **Exceptions to NUREG-1801**, replace the existing section in its entirety with the following,

"One alternative to the ASME Section XI, 2001 edition through the 2003 addenda, is expected to be implemented as part of the aging management program in effect at the time Palisades enters the period of extended operation. This alternative is identified as an exception to NUREG-1801, and is justified as acceptable from an aging management point of view, in accordance with 10 CFR 54.

The specific exception identified to the NUREG-1801 program description is listed below with a justification for its acceptability. It has been determined that this exception only applies to the Detection of Aging Effects element.

1) Risk-Informed Inservice Inspection Program

   This alternative implements Risk-Informed Examination of Class 1, Class 2, Class 3 and Non Class Piping Butt Welds using Westinghouse Owners Group WCAP-14572, Revision 1-NP-A, "Westinghouse Owners Group Application of Risk-Informed Methods to Piping Inservice Inspection Topical Report" and WCAP-14572, Revision 1-NP-A, Supplement 1, "Westinghouse Structural Reliability and Risk Assessment (SRRA) Model for Piping Risk-Informed Inservice Inspection"

   The Risk-Informed Inservice Inspection (RI-ISI) program provides an acceptable alternative to the piping ISI requirements with regards to (1) the number of locations, (2) the locations of inspections, and (3) the method of inspection. The RI-ISI program maintains the fundamental requirements of ASME Section XI, such as the examination technique, examination frequency, and acceptance criteria. Although the RI-ISI program reduces the number of required examination locations in some cases, it maintains an acceptable level of quality and safety by focusing inspections on the most safety significant welds with nondestructive examination (NDE) techniques that are more focused towards finding the type of expected degradation as well as the types of flaws and degradation found during traditional inspections.

A systematic approach was used to identify component susceptibility to common degradation mechanisms and to categorize these degradation mechanisms into the appropriate degradation categories with respect to their potential to result in a postulated leak or rupture in the pressure boundary. An evaluation to determine the susceptibility of components to a particular degradation mechanism that may be a precursor to a leak or rupture in the pressure boundary, and an independent assessment of the consequences of a failure at that location were performed. Industry and plant-specific piping failure information (i.e., operating experience) was used to identify piping degradation mechanisms and failure modes, and consequence evaluations were performed using PRAs to establish safety ranking of piping segments for selecting new inspection locations. The degradation mechanisms identified in the RI-ISI program include thermal fatigue, mechanical fatigue, flow accelerated corrosion (FAC), microbiologically influenced corrosion (MIC), intergranular stress-corrosion cracking (IGSCC), and primary water stress-corrosion cracking (PWSCC). The consequences of pressure boundary failures were evaluated and ranked on their impact on core damage and early release. Therefore, redistributing the welds to be inspected with consideration of the safety significance of the segments provides assurance that segments whose failures have a significant impact on plant risk receive an acceptable and improved level of inspection.

The objective of ISI, required by ASME Section XI, is to identify conditions (e.g., flaw indications) that are precursors to leaks and ruptures in the pressure boundary that may impact plant safety. The RI-ISI program meets this objective. The risk-informed selection process not only identifies the risk-important areas of the piping systems but also defines appropriate examination methods, examination volumes, procedures, and evaluation standards necessary to address the degradation mechanism(s) of concern and the ones most likely to occur at each location to be inspected. Therefore, the examination methods of the RI-ISI program are acceptable since they are selected based on specific degradation mechanisms, pipe sizes, and materials of concern. The risk significance of piping segments is taken into account in defining the inspection scope of the RI-ISI program. The RI-ISI program methodology provides reasonable assurance that any reduction in inspections will not lead to degraded piping performance when compared to the existing performance levels. Inspections are focused on locations with active degradation mechanisms as well as selected locations that monitor the performance of system piping. Inspection strategies ensure that failure mechanisms of concern have been addressed and there is adequate assurance of detecting damage before structural integrity is affected.

The RI-ISI program is a living program that includes performance monitoring and feedback provisions to confirm the assumptions and analyses used in the development of the program. Feedback of relevant information is used to ensure the appropriate identification of safety-significant piping locations. As a minimum, risk-ranking of piping segments is reviewed and adjusted on an ASME-period basis. Significant changes may require more frequent adjustment of the risk-ranking of piping segments as directed by NRC bulletin or generic letter requirements, or industry and plant-specific feedback (i.e., operating experience).

In conclusion, the RI-ISI program is a full scope program that includes ASME Class 1, 2, 3, and non-class piping systems. The proposed alternative program provides an acceptable level of quality and safety. Additionally, the alternative program will not be limited to ASME Class 1 or Class 2 piping, but will encompass the high safety significant piping segments regardless of ASME Class Other non-related portions of the ASME Section XI Code are unaffected. WCAP-14572 defines the relationship between the risk-informed examination program and the remaining unaffected portions of ASME Section XI. This alternative provides an acceptable aging management program.

On page B-21, third paragraph, revise the first sentence to read as follows,

"The ISI Program meets the requirements of ASME Section XI, in accordance with applicable provisions and requirements of 10 CFR 50.55a.

On page B-22, replace the last paragraph under heading **Detection of Aging Effects** in its entirety with the following,

"The ASME Section XI IWB, IWC, IWD, IWF Inservice Inspection Program implements one alternative to the requirements of the ASME B&PV Code, Section XI, 2001 edition including 2002 and 2003 addenda. This alternative is considered an exception to the NUREG 1801 descriptions of the programs.

The specific exception and justification of acceptability in an aging management program are described in the preceding section entitled Exceptions.

This element is consistent with, but contains an exception to, NUREG 1801, Section XI.M1, "ASME Section XI Inservice Inspection, Subsections IWB, IWC. and IWD." This element is consistent with NUREG-1801, Sections XI.M3, "Reactor Head Closure Studs," and Section XI.S3, "ASME Section XI, Subsection IWF.""

On page B-25, under the heading **Conclusion**, replace the first paragraph in its entirety with the following,

> "The ASME Section XI IWB, IWC, IWD, IWF Inservice Inspection Program is an existing program that uses as its bases, various industry and NRC standards. This program is consistent with, but contains an exception to, NUREG-1801, Section XI.M1, "ASME Section XI Inservice Inspection, Subsections IWB, IWC and IWD." This program is consistent with NUREG-1801, Section XI.M3, "Reactor Vessel Closure Studs," and Section XI.S3, "ASME Section XI, Subsection IWF."

NMC has also concluded that no changes are necessary to the ASME Section XI, Subsections IWB, IWC, IWD, IWF Inservice Inspection Program description in LRA Section A2.2.

This completes NMC action in response to this commitment.

ENCLOSURE 2

Updated NMC Response to NRC Request for Additional Information 3.5.2-4-1(b)

(2 Pages)

NRC Follow Up Question from 11/15/05 Conference Call Concerning Response to RAI 3.5.2-4-1(b)

Based on the discussion of the elevated temperature condition in and around the primary shield wall in Section 3.5.2.2.2.1, the staff agrees with EPRI TR 103842, that the concrete properties will not be significantly affected, if the actual temperatures around the shield wall remain within the estimated limits. However, additional shrinkage and loss of moisture due to radiation could degrade the concrete on a long term basis. In this context, please provide a summary of the results of the last two inspections performed for:

(1) the primary shield wall.
(2) RPV supports,
(3) grouted anchorages, and
(4) masonry walls inside the containment.

Updated Response to NRC RAI 3.5.2-3-1(b)

The NMC response to this RAI provided in a letter dated July 28, 2005, is hereby replaced in its entirety with the following:

1 & 2) No inspection results are available. As discussed in Section 2.4.4 of the LRA, the entire interior concrete surface of the Palisades Primary Shield Wall is lined with welded carbon steel plate. This includes the area around the RPV supports. Accordingly, the shield wall concrete and the concrete surrounding the RPV supports are not accessible for inspection. In Palisades FSAR Figure 6-6, which shows the reactor vessel, reactor cavity, bioshield, reactor supports, insulation, etc., it can be seen that these areas are inaccessible and are high radiation dose locations.

With regard to the potential for additional shrinkage and loss of moisture due to radiation that could degrade concrete on a long term basis, the shield cooling system maintains structural concrete at temperatures that mitigate concrete thermal heating due to radiation or conduction (see LRA page 2-147 for shield cooling system description). These embedded cooling system coils are installed more densely around the steel components that support the reactor vessel to maintain the steel and concrete around the supports at the design temperature.

Loss of concrete strength due to cumulative radiation exposure, is addressed in the Palisades LRA Page 3-7, Table 3.01, Service Environments, which provides the following discussion:

> "Radiation - Plant radiation doses outside the reactor cavity are not of concern for aging management. Materials can be affected by cumulative radiation exposure. For concrete, neutron fluence above 10^{19} n/cm2 (>1 Mev) or gamma dose >10^{10} rads is required to cause degradation."

The Palisades' neutron fluence estimate through the end of the proposed extended operating period at the outside diameter (OD) of the reactor vessel, is 1.94 x 10^{18} n/cm^2. The fluence at the inside diameter (ID) of the biological shield can be assumed to be the same." This is less than the threshold value for degradation of 1.0 x 10^{19} n/cm^2 . At a depth of approximately 11 inches into the biological shield (the approximate depth of the non-structural, non-reinforced, sacrificial concrete – See LRA page 2-228 for cavity description), the neutron fluence is equivalent to 10^{17} n/cm^2 (E > 1 Mev). This is less than the threshold value of 1.0 x 10^{19} n/cm^2 by two orders of magnitude. Palisades estimates that gamma dose at a depth of approximately 11" behind the biological shield liner plate is 4.74 x 10^8 rads (less than 1.0 x 10^{10} threshold). These values of neutron fluence and gamma dose are even lower above or below the core centerline. Therefore, the sacrificial concrete portion of the bioshield is not subject to degradation due to loss of concrete strength from cumulative radiation exposure.

Based on the above discussion, no aging management program is required for the interior reactor cavity steel liner plate, interior concrete primary shield wall or the RPV supports. However, as discussed, the accessible external portions of the reinforced concrete bioshield are included in the Palisades Structures Monitoring Program.

3) The term "grouted anchorage" in the description of "Building Framing - Containment Cavity" in LRA Table 2.4.4-1 is used generically. The specific anchorage in the vicinity of the reactor shield wall is cast-in-place bolting or strap anchors, depending on elevation, for the liner plate. These anchors are not accessible for inspection.

4) There is one block wall inside containment on the 649' level, which is remote from the high temperature and radiation environment of the shield wall. Structural Monitoring Program inspections were performed inside containment in 1996 and 1999. The top five courses of masonry wall blocks were found spalled at the northern most tip of the wall. The existing condition was determined not damaging to the masonry wall integrity, which serves only as a partition wall. The condition was deemed acceptable as-is.

This masonry wall (with vertical and horizontal steel reinforcement) partially surrounds the shield cooling system surge tank and associated piping components which are not safety related. Palisades FSAR Figure 1-6 (E2) shows the location at Elevation 649' of the shield cooling surge tank, that is surrounded on three sides by the masonry block wall. FSAR Table 5.2-3 shows the component classification of the surge tank, piping and valves, as Class 3. On the West side, these two walls are "qualified", indicating that they were analyzed to ensure that failure would not occur due to a design basis earthquake load in accordance with NRC IEB 80-11 (See Palisades FSAR Section 5.10.3.2 Masonry Walls). The North and East side walls are identified as "unqualified", since their failure would not impact any safety related equipment or components, since that area is used for lay-down/storage purposes, as shown on FSAR Figure 1-6.

ENCLOSURE 3

Supplemental Information for LRA Section 4.3.12

(2 Pages)

NRC Follow Up Question from 11/15/05 Conference Call Concerning LRA Section 4.3.12

The applicant stated that breaks in Class 1 high energy piping were not postulated based on exceeding a fatigue cumulative usage factor (CUF) criterion. This criterion is specified in FSAR Section 5.6.3.1 and NRC Generic Letter 87-11 as CUF = 0.10. Therefore, the applicant claims that postulation of ASME III Class 1 HELB Locations and Leak Before Break analyses based on the fatigue CUF limit is not a TLAA.

The applicant is requested to provide justification for taking exception to the requirement in FSAR Section 5.6.3.1 and NRC Generic Letter 87-11 to apply the CUF>0.10 criterion for postulating HELB in Class 1 piping for the period of extended operation.

NMC Response to NRC Follow Up Question

Section 4.3.12 of the Palisades LRA states:

> **4.3.12 Absence of a TLAA for ASME III Class 1 HELB Locations and Leak-Before-Break Analyses Based on Fatigue Usage Factor**
>
> Review of the Palisades licensing basis and the associated HELB reports revealed that selected break locations, either inside or outside containment, were not dependent on aging factors. Therefore, HELB analyses at Palisades are not TLAAs.

This summary was based on an evaluation of the methodology used to perform High Energy Line Break (HELB) analyses under Palisades' Current Licensing Bases (CLB), as summarized in the Palisades FSAR. Additional information relevant to this subject can be found in FSAR Section 5.6, and in LRA Section 4.3.1 (pages 4-16 through 18), Table 4.3.1-2 (page 4-21), and Section 4.3.8 (pages 4-29 through 31). Also see Section 4.3.13 (pages 4-36 through 38) for additional discussion of which piping has been analyzed for fatigue.

When the Palisades Plant was designed, ASME Boiler & Pressure Vessel Code Section III did not address piping. FSAR Section 5.2 discusses the plant design classes of the Palisades CLB that exist in lieu of ASME design classes 1, 2 and 3. Most piping was designed and installed under ANSI B31.1 regardless of function. B31.1 did not require piping analyses for fatigue usage. As described in FSAR Section 5.6.7, Palisades' HELB analyses were not required as part of the original design, and were performed after the plant was licensed to operate. As described in FSAR Section 5.6.3.1, the HELB analysis criteria that governed break location selection in ASME Section III, Class 1, piping included a provision based on CUF where fatigue analyses had been done. However, that criterion was not applicable to most high energy lines in the plant since they were not designed to ASME Section III, and fatigue of piping had not been analyzed under the B31.1 design.

ASME class designators have since been assigned to piping and components based on function for Section XI inservice and preservice (post-repair or replacement) inspection purposes only. These designations are not applicable to piping design analyses, which are governed by the design requirements in FSAR Sections 5.2 and 5.7.

With the exception of a portion of the 42" hot leg and 30" cold leg primary coolant piping, which were analyzed using a Mechanistic Approach for HELB, all other high-energy systems were analyzed under the Effects Oriented Approach. The Mechanistic Approach selects line break locations based on either piping stress locations that exceed 2.4 S_m or a CUF that exceeds 0.1. The Effects Oriented Approach does not use fatigue stress to determine line break locations.

The fatigue usage factors for the hot and cold legs were calculated based on the enveloped stress ranges for locations in each of a typical hot leg and cold leg. Since the fatigue calculations were not location-specific, the CUF criteria cannot be used in the Mechanistic Approach to determine the line break location. Therefore, only piping stress criteria are available for use in the Mechanistic Approach for determining Palisades' line break locations in the hot leg and cold leg.

In summary, the Palisades HELB analysis methodology included one criterion for selection of break locations in ASME Section III Class 1 piping based on CUF (where analyzed), among others. In practice, as discussed in FSAR Section 5.6.7, however, Palisades break locations were postulated using other selection criteria, and the CUF criterion was not used. The approaches to Palisades' HELB analyses contained in the Palisades CLB are summarized below:

Inside Containment: HELB locations were determined from the Effects Oriented Approach for all high-energy piping, except for a portion of the Primary Coolant System hot leg and the cold leg where pipe stress criteria of the Mechanistic Approach were used. HELB locations were not determined based on a TLAA.

Outside Containment: The Effects Oriented Approach was used. HELB locations were not determined based on a TLAA.

Leak-Before-Break: Leak-before-break analyses at 13 locations inside containment were done to confirm that the effect of breaks are limited, and are not time-dependent.