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Palisades Nuclear Plant
Operated by Nuclear Management Company, LLC

July 1, 2005

10 CFR 54

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

Response to NRC Request for Additional Information Relating to License Renewal dated June 3, 2005

In a letter dated June 3 , 2005, the Nuclear Regulatory Commission (NRC) requested additional information regarding the License Renewal Application for the Palisades Nuclear Plant. This letter responds to those requests.

Enclosure 1 provides the text of, and the NMC response to, each NRC request.

Please contact Mr. Darrel Turner, License Renewal Project Manager, at 269-764-2412, or Mr. Robert Vincent, License Renewal Licensing Lead, at 269-764-2559, if you require additional information.

Summary of Commitments

This letter contains two new commitments, as follows:

The final text and schedule of licensee commitments that are confirmed by NRC in the final SER for the Palisades Renewed Operating License will be incorporated into appropriate locations of the FSAR in the first regular FSAR update under 10 CFR 50.71(e) following NRC issuance of the renewed operating license.

Visual inspections of a sample of buried carbon, low-alloy, and stainless steel components will be performed within ten years after entering the period of extended operation, unless opportunistic inspections have occurred within this ten-year period. Prior to the tenth year, NMC will perform an evaluation of available data to determine if sufficient inspections have been performed to assess the condition of the components. If insufficient data exists, focused inspection(s) will be performed as needed.



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I declare under penalty of perjury that the foregoing is true and correct. Executed on July 1, 2005.

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

Enclosure (1)

CC Administrator, Region III, USNRC
Project Manager, Palisades, USNRC
Resident Inspector, Palisades, USNRC
License Renewal Project Manager, Palisades, USNRC

## ENCLOSURE 1

## NMC Responses to NRC Requests for Additional Information dated June 3, 2005

**NRC RAI-A1.0-1**

Please include a schedule of implementation and description of any commitments as part of the Aging management Program (AMP) and time-limited aging analysis (TLAA) descriptions in the USAR Supplement in Appendix A of the license renewal application (LRA).

**NMC Response to NRC RAI-A1.0-1**

The NMC letter dated March 22, 2005, transmitting the Application for Renewed Operating License to the NRC, in Enclosure 2, provides the preliminary commitments related to license renewal for the Palisades Nuclear Plant. This list also specifies the section of the application in which the commitment was discussed. As noted in the letter, these commitments are subject to change during the NRC review. NRC practice is to include the final text of all commitments that are proposed by NMC, and found to be acceptable by the NRC, in the final NRC Safety Evaluation Report (SER).

The final text and schedule of the licensee commitments, that are confirmed by NRC in the final SER for the Palisades Renewed Operating License, will be incorporated into appropriate locations of the FSAR in the first regular FSAR update under 10 CFR 50.71(e) following NRC issuance of the renewed operating license.

Several of the preliminary commitments contained in the March 22, 2005 letter are not applicable to incorporation into the FSAR. A specific response for each preliminary commitment is provided as follows:

Commitment 1, which pertains to the annual update of the LRA during the NRC review period, is not applicable to incorporation into the FSAR.

Commitments 2 through 7, which relate to TLAAs, will be implemented according to the schedule provided in the text of each commitment. The text and schedule of each final commitment will be incorporated into the appropriate location of the FSAR as stated above.

Commitment 8, which pertains to the addition of LRA Appendix A information into the FSAR in accordance with 10 CFR 50.71(e) following issuance of the renewed license, is not, in itself, applicable to incorporation into the FSAR.

Commitments 9 through 37 itemize the program development and enhancement actions that are identified in LRA Appendix B. Except when otherwise noted in an individual program discussion, new programs and enhancements are scheduled to be implemented prior to the period of extended operation. The text and schedule of each final commitment will be incorporated into the appropriate location of the FSAR as stated above.

**NRC RAI-B2.1.2-1**

Please define the ASME Code Edition for Inservice Inspection (ISI) Program B2.1.2.

**NMC Response to NRC RAI-B2.1.2-1**

During the Aging Management Programs (AMP) audit conducted June 20 - 23, 2005, a similar question was asked by the audit team. An extensive discussion ensued about which version of the ASME Section XI code could be used as the basis of the license renewal aging management program that is reviewed by the NRC, and what impacts an ASME code update under 10 CFR 50.55a, and the existing NRC-approved Risk-Informed Inservice Inspection Program, would have on the review. Possible options, including decoupling the Inservice Inspection Program under 10 CFR 50.55a from the license renewal aging management program under 10 CFR 54, were reviewed. The audit question was left open, and additional management discussions are planned.

Therefore, the specific answer to this RAI question will not necessarily define the version of the ASME Code that NRC should review as the Palisades AMP. A final response to that question must be deferred until after additional discussion with the NRC staff.

The specific information requested for RAI-B2.1.2-1 is as follows:

The Code of Federal Regulations, 10 CFR 50.55a, 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. The Palisades Inservice Inspection Program for the current (3$^{rd}$) ten-year interval, which began on May 12, 1995, implements the 1989 edition, no addenda, of ASME Section XI as modified by 10 CFR 50.55a and approved Relief Requests and Code Cases. One of the NRC-approved relief requests authorizes implementation of a risk-informed inservice inspection program. This program will be in effect until the end of the current ten-year interval, which ends December 12, 2006.

In 2006 NMC expects to update the inservice inspection code edition and addenda to those required for the fourth ten-year interval. It is anticipated that the Section XI version which will be incorporated into the Palisades program will be the 2001 edition through the 2003 addendum. This edition and addendum are the latest currently incorporated by reference in 10 CFR 50.55a(b).

**NRC RAI-B2.1.5-1**

Please identify the inspection frequency for the buried piping aging management program B2.1.5.

**NMC Response to NRC RAI-B2.1.5-1**

The Palisades Buried Services Corrosion Monitoring Program includes inspection activities that are designed to detect degradation due to aging effects prior to loss of intended function. Visual inspections of a sample of buried carbon, low-alloy, and stainless steel components will be performed within ten years after entering the period of extended operation, unless opportunistic inspections have occurred within this ten-year period. Prior to the tenth year, NMC will perform an evaluation of available data to determine if sufficient inspections have been performed to assess the condition of the components. If insufficient data exists, focused inspection(s) will be performed as needed.

This is very similar to the position contained in the "Safety Evaluation Report related to the License Renewal of the Joseph M. Farley Nuclear Plant, Units 1 and 2," issued as NUREG-1825.

**NRC RAI-2.1.1-1**

No license renewal (LR) system boundary flags or other means of identifying LR system boundaries appear on the scoping boundary drawings. Please identify the LR system boundaries in these drawings.

**NMC Response to NRC RAI-2.1.1-1**

Several sets of multi-colored drawings were sent to the License Renewal Project Manager on May 25, 2005. Each drawing has a separate color for each system that has components on that drawing in scope for license renewal. The color changes denote the license renewal system boundaries.

**NRC RAI-2.2-1**

For the various "component group/intended function AMR tables" listed in Section 2 of the LRA, please clearly identify the specific components that comprise the listed component types and groups.

**NMC Response to NRC RAI-2.2-1**

Major components that are in scope for license renewal are shown on the scoping boundary drawings that have been provided to the NRC. Information regarding specific components and subcomponents is available on site for review.

**NRC RAI-2.3.1-1**

In the initial review of the LRA it was noted that flange leak detection lines are not presented. The staff believes that these components have intended functions. Please provide your technical basis if they are not in scope for license renewal.

**NMC Response to NRC RAI-2.3.1-1**

NMC has treated the between-the-seals portion of the reactor flange leak detection tap and the downstream piping as having intended functions and being in-scope for license renewal. The in-scope head leak detection piping is depicted on drawing LR-M-201-1. LRA Table 2.3.1-2 for the reactor vessel provides a specific line item for reactor vessel head O-ring leakage monitoring, with an intended function of pressure boundary/fission product retention. The outer flange seal leak detection portion is not in scope.

The reactor vessel head leak detection tube (or tap) was evaluated with the reactor vessel and included in the summary of reactor vessel aging management evaluations provided in LRA Table 3.1.2-2. As indicated in the table, this component of head leak detection is managed using the Alloy 600 Program.

The stainless steel leak detection piping downstream of the integral Alloy 600 tube (or tap) was evaluated with the Primary Coolant System (not the reactor pressure vessel). Therefore, the stainless steel flange leak detection lines are included in LRA Table 3.1.2-1 under the small-bore stainless steel pipe line item. NMC grouped the leak detection piping in this manner to prevent having an incongruous "piping" component in the reactor vessel group.

**NRC RAI-2.3.2-1**

Please provide separate system description, system function listing, FSAR reference, scoping boundary drawings, and components subject to an aging management review (AMR) for each of the 9 systems that have been integrated in the Engineered Safety Features (ESF) section of the LRA.

**NMC Response to NRC RAI-2.3.2-1**

Section 2.3.2 of the LRA describes Palisades' Engineered Safeguards System (ESS) in a manner consistent with the current licensing basis (CLB) and Final Safety Analysis Report (FSAR). The LRA statement, "The system is divided functionally into seven mechanical subsystems: ..." was not intended to imply that ESS consists of, or can be subdivided into, discrete subsystems. The statement was intended to show that the ESS System, with one exception, is a single mechanical system (not a compilation of separate systems), which provides multiple functions. The exception is the Reactor Cavity Flood subsystem. Section 2.3.2 goes on to state, "Except for Reactor Cavity Flood, the mechanical subsystems use most of the same system components for the various subsystem functions, and those components are hydraulically interconnected. Therefore, ESS is most accurately presented as a single system, and the groups of components that provide each major function are characterized as subsystems for license renewal purposes."

NMC's use of the term "subsystem" in the descriptions of the ESS functions, and the inclusion of electrical items in the mechanical ESS description, may have caused some confusion. The following discussion will address the listed "subsystems" in three parts: ESS Mechanical (High Pressure Safety Injection (HPSI), Low Pressure Safety Injection (LPSI), Containment Spray (CSS), Safety Injection Tanks (SIT), Safety Injection and Refueling Water Tank and Containment Sump Suction (SIRW), and Shutdown Cooling (SDC)); Reactor Cavity Flood (RCF); and ESS Electrical.

<u>ESS Mechanical (HPSI, LPSI, CSS, SIT, SIRW, SDC)</u>
These mechanical functions are provided by a single system using shared components, as described in FSAR Chapter 6, Engineered Safety Features, Sections 6.1 and 6.2. In practice, these in-scope mechanical functions of ESS are not independent of each other, but are provided by changing the operating equipment lineups. Since the mechanical components supporting these ESS functions are nominally made of the same materials and subject to the same environments, the aging management results do not vary by function. It was judged most logical for license renewal, therefore, to review the components providing these functions as a single system.

To facilitate NRC reviewer understanding of ESS design and operation for these functions, and to facilitate assignment of a separate reviewer for CSS, additional descriptive information and training materials were provided to the Project Manager on April 12, April 15, and April 27, 2005. The information provided did not revise the application, but supported improved understanding and a suggested boundary definition for reviewer assignments.

For these functions, therefore, the requested information is provided in LRA Section 2.3.2.

Reactor Cavity Flood
The Reactor Cavity Flooding (RCF) piping is described in FSAR Chapter 6, Engineered Safety Features, Section 6.8. This piping was included for completeness of the ESS description, but the functions of this piping are not in scope for license renewal. Only two of the RCF components, reactor cavity drain plugs M-983A and M-983B, are in-scope because they are designated as safety-related for their seismic mounting (See function RCF-03). M-983A and M-983B are not addressed with ESS in LRA Section 3.2, but are, instead, addressed as part of a Non-ASME Piping & Mechanical Component Support commodity in the civil/structural area. As stainless steel components in air, there are no aging effects requiring management for these drain plugs.

ESS Electrical
The two electrical "subsystems" mentioned in LRA Section 2.3.2, Engineered Safety Features (ESF) Actuation, and Normal Shutdown (NSD) and Design Basis Accident (DBA) Sequencers, should not have been listed in this mechanical system description. These electrical "subsystems" are addressed in LRA Section 2.5.3 (Safety Injection System control circuits and Normal shutdown and Design Basis Accident Sequencer) in conjunction with the electrical Containment Isolation and Penetration System.

As discussed in LRA Section 2.5.3, there were no assets in these in-scope electrical "subsystems" that required aging management review. Any electrical components, cables, etc. that required aging management review were assigned to commodity groups as described in LRA Sections 2.5 and 2.5.1.

**NRC RAI-2.3.4-1**

In the initial review of the LRA it was noted that the steam generator feedwater inlet ring is not presented. The staff believes that this component has intended functions. Please provide your technical basis if it is not in scope for license renewal.

**NMC Response to NRC RAI-2.3.4-1**

The methodology for identifying the SSCs within the scope of license renewal and their intended functions is provided in Section 2 of the LRA, and is consistent with that contained in Section 2.1.2.1 of NUREG-1800 and in NEI 95-10. As indicated in NEI 95-10 and NUREG-1800, the fact that a system or component serves an intended license renewal function in accordance with 10CFR 54.4(b) does not mean that all of its subcomponents also support that function. In this case, as shown on drawing LR-M-201, the replacement steam generators (RSG) are components within the scope of license renewal; however, the feedwater inlet ring subcomponents do not support the RSG intended functions.

The intended functions of the RSG itself are listed in Section 2.3.1.4 of the LRA. The feed rings are not discussed in that section because the feed rings do not support any of the listed RSG functions, and, therefore, do not require an AMR.

The forged carbon steel feedwater nozzle is welded into the upper shell of the RSG. Feedwater is introduced into the secondary side of the RSG through the feedwater nozzle, the welded thermal sleeve assembly and fittings, and the feedwater distribution ring. Since the distribution function is performed within the steam generator, the feed ring does not support the RSG secondary side fluid pressure boundary function, nor does it perform any other license renewal intended function.

The feedwater ring support is included in GALL, with the ring itself, as Volume 2 Line Item IVD.1.3.1. The aging mechanism of concern is flow-accelerated corrosion (FAC). The Palisades response to that issue is provided in LRA Section 3.1.2.2.14. At Palisades, the carbon steel feed ring support is considered an integral part of the RSG carbon steel upper shell. Accordingly, the feed ring support function is subsumed into and managed as indicated under GALL Volume 2 line item IVD.1.1-c in LRA Table 3.1.2-4 for the RSG shell.

**NRC RAI-2.4-1**

Crane rail supports, hoists, and lifting devices are not explicitly identified as in scope for license renewal in the LRA. They are usually in the scope of license renewal. Please confirm that they are in scope and identify their location in the LRA, or provide technical justification if they are not in scope.

**NMC Response to NRC RAI-2.4-1**

Cranes, crane rail supports, hoists and one lifting device are in scope for license renewal and are identified in LRA Section 2.4.8, "Miscellaneous Structural and Bulk Commodities" of the License Renewal Application (LRA). Their functions are discussed in the system function description for "Cranes (BLC-NSAS)" and "Cranes (BLC-SR) "of the LRA. They are specifically identified in Tables 2.4.8-1 and 3.5.2-8.

Crane lifting devices, with one exception, are not in scope of license renewal since they are categorized as tools. Tools are not attached, or are only temporarily attached, to the building structure. Accordingly, they perform no license renewal intended functions. This is consistent with other applications as described in the license renewal Safety Evaluation Reports for Millstone (section 2.4A.6) and Arkansas Nuclear One Unit 2 (section 2.4). The one exception is a portion of the Reactor Vessel Head Lifting Device that remains attached to the reactor vessel head during normal plant operation. This component is in-scope for license renewal and is listed in the above referenced tables.

**NRC RAI-3.1-1**

During an initial review of the AMRs and the associated evaluations in the LRA, it was not apparent how you intend to manage thermal sleeve aging effects. Please provide information on how you plan to manage aging of thermal sleeves.

**NMC Response to NRC RAI-3.1-1**

The methodology for identifying the SSCs within the scope of license renewal and their intended functions is provided in Section 2 of the LRA, and is consistent with that contained in Section 2.1.2.1 of NUREG-1800 and in NEI 95-10. As indicated in NEI 95-10 and NUREG-1800, the fact that a system or component serves an intended LR function (e.g. fluid pressure boundary) in accordance with 10CFR 54.4(b) does not mean that all of its subcomponents support that LR function.

In this case, the thermal sleeves do not provide a fluid pressure boundary function and, as piping subcomponents, do not have an intended license renewal function. Therefore, thermal sleeves should not have been included in the LRA.

Although thermal sleeves do not serve a license renewal intended function, NMC believes that they are adequately managed by the Water Chemistry Program as the only practical management method.

**NRC RAI-3.1.2-1**

AMPS that note "one-time inspections" should identify an acceptable form of inspection method for various types of situations. The LRA does not identify any specific methods of inspection. The LRA simply provides a general statement that examination techniques will be visual, volumetric, or other appropriately established NDE methods. Please identify the inspection methods for each "one time inspection" listed.

**NMC Response to NRC RAI-3.1.2-1**

Specific methods of inspection for individual components will be identified as part of implementation procedure development. NMC will begin working on implementation later this year and has plans to complete draft aging management programs and implementing procedures in 2006. The One-Time inspection program will be included in this effort. It is expected that Palisades' One Time Inspection methods will be generally in accordance with the table provided in NUREG-1801, Rev. 1, Vol. 2, as follows:

**Examples of Parameters Monitored or Inspected and Aging Effect for Specific Structure or Component**

| Aging Effect | Aging Mechanism | Parameter Monitored | Inspection Method |
|---|---|---|---|
| Loss of Material | Crevice corrosion | Wall Thickness | Visual (VT-1) and/or Volumetric (RT or UT) |
| Loss of Material | Galvanic corrosion | Wall Thickness | Visual (VT-3) and/or Volumetric (RT or UT) |
| Loss of Material | General Corrosion | Wall Thickness | Visual (VT-3) and/or Volumetric (RT or UT) |
| Loss of Material | MIC | Wall Thickness | Visual (VT-3) and/or Volumetric (RT or UT) |
| Loss of Material | Pitting corrosion | Wall Thickness | Visual (VT-1) and/or Volumetric (RT or UT) |
| Loss of Material | Selective Leaching | Wall Thickness | Hardness Test |
| Loss of Material | Erosion | Wall Thickness | Visual (VT-3) and/or Volumetric (RT or UT) |
| Loss of Heat Transfer | Fouling | Tube Fouling | Visual (VT-3) or remote visual |
| Cracking | SCC, thermal stratification and turbulent penetration | Cracks | Volumetric (RT or UT) |
| Loss of preload | Stress Relaxation | Various | Visual (VT-3) |

**NRC RAI-3.1.2.2.2-1**

The LRA states that an augmented inspection for steam generator shell assemblies for loss of material due to pitting/crevice corrosion is not applicable for Palisades. NUREG-1801, "Generic Aging Lessons Learned (GALL) Report" recommends such augmented inspection based on industry experience and extended exposure of the shell material to the water environment. Please provide technical justification for your determination.

**NMC Response to NRC RAI-3.1.2.2.2-1**

Section 3.1.2.2.2 of the SRP-LR states that the loss of material due to pitting and crevice corrosion could occur in the steam generator shell assembly. The existing program relies on control of water chemistry to mitigate corrosion, and inservice inspection to detect the loss of material. The extent and schedule of the existing steam generator inspections ensure that flaws cannot attain a depth sufficient to threaten the integrity of the welds. However, the NRC states in NRC IN 90-04, "Cracking of the Upper Shell-to-Transition Cone Girth Welds in Steam Generators," dated January 26, 1990, if pitting and crevice corrosion of the shell exists, the program may not be sufficient to detect pitting and corrosion. The GALL Report recommends augmented inspections to manage this aging effect.

In Section 3.1.2.2.2.1 of the LRA, Palisades states that pitting/crevice corrosion is not known to exist in the steam generator shells, and, therefore, augmented inspections are not necessary. This statement was based upon the following operating experience: (1) In February of 2000 the steam generator program engineer, who was also a certified welding inspector, completed a 360 degree walk down of the secondary side internal wall, at the elevation of the main feedwater ring. During that walk down no evidence of ID pitting was identified; and (2) During the 2003 refueling outage, 2 complete steam generator shell circumferential welds were examined from the OD using volumetric inspection techniques. These weld inspections did not identify evidence of internal pitting in the associated steam generator shell area.

The original Palisades steam generators were replaced in 1990 with Combustion Engineering Model 2530 steam generators. Since then, Palisades has maintained secondary water chemistry in accordance with EPRI guidelines. The combination of these factors, coupled with continued water chemistry maintenance and ISI inspections provides reasonable assurance that pitting/crevice corrosion will not threaten the steam generator shell pressure boundary function during the period of extended operation. This is consistent with the staff's conclusions in past SERs, including the SER for Plant Farley (NUREG 1825).

**NRC RAI-3.1.2.2.7.2-1**

The AMP for Cast Austenitic Stainless Steel (CASS) thermal embrittlement in the LRA does not include a flaw tolerance evaluation or enhanced volumetric inspection as recommended in the GALL report. Please clarify and discuss your basis.

**NMC Response to NRC RAI-3.1.2.2.7.2-1**

As stated in Section B2.0 of the LRA, Palisades does not have a GALL XI.M12 AMP for Cast Austenitic Stainless Steel (CASS). Palisades has no CASS material in the primary coolant system other than valve bodies and pump casings which are managed under the ASME Section XI IWB, IWC, IWD, IWF Inservice Inspection Program discussed in Section B2.1.2.

**NRC RAI-3.2.1-1**

The flow assisted corrosion (FAC) program is not listed as an AMP in Sections 3.2.1 & 3.2.3 of the LRA. Such a program is typically necessary to manage the effects of FAC for license renewal. Please verify whether you intend to credit this AMP, B2.1.11, in these sections.

**NMC Response to NRC RAI-3.2.1-1**

Palisades does not credit the FAC AMP described in LRA Section B2.1.11 in LRA Sections 3.2.2.1 and 3.2.2.2.6 because there are no FAC susceptible material environment combinations in those ESF components. Most ESF components in contact with fluid are stainless steel, which is FAC resistant material. Further, except for some cast austenitic stainless steel valves, the Palisades ESF components were evaluated in their normal standby static condition of 120°F.

**NRC RAI-3.6.2-1**

In the notes area for Table 3.6.2-1 (and in other notes for other tables) various components are described; however, they are not specified in the associated table. for example, are neutron monitoring cables and uninsulated ground connectors in scope for license renewal? Where are they located in the associated LRA tables?

**NMC Response to NRC RAI-3.6.2-1**

The referenced plant specific notes are associated with a particular line item in the Table 3.6.2-1. The plant specific note applicable to neutron monitoring cables is note 602 (on page 3-418 of the LRA), which is referenced in the second row of Table 3.6.2-1 on page 3-415. Neutron monitoring cables are in-scope and are included in the listed commodity, "Electrical cables and connections used in instrumentation circuits not subject to 10 CFR 50.49 EQ requirements that are sensitive to reduction in conductor IR. (ISG-15) (Nuclear Instrumentation and Radiation monitoring systems)."

Uninsulated ground conductors are not referenced in Table 3.6.2-1, as they are not in scope for license renewal. The Palisades plant uninsulated grounding cables are installed to provide personnel safety and economic equipment protection, and are not associated with supporting any system or component License Renewal intended function. This is described in the fifth paragraph of FSAR section 8.3.1.2 which states "The 4,160 volt switchgear is provided with relay protection, grounding and the mechanical safeguards necessary to assure adequate personnel protection and to prevent or limit equipment damage during system fault conditions."

**NRC RAI-4.5-1**

Please provide the containment tendon prestress test data from past surveillance.

**NMC Response to NRC RAI-4.5-1**

Containment tendon prestress test data from past surveillances are provided in the following Table.

**Palisades Nuclear Plant Containment Tendon Prestress Test Data**

| Surveillance, Date | Hoop Tendon | Force (kips) | Vertical Tendon | Force (kips) | Dome Tendon | Force (kips) |
|---|---|---|---|---|---|---|
| 1-year, 1971 | BF-65 | 694.00 | V-84 | 700.00 | D1-38 | 668.50 |
| | BD-22 | 660.50 | V-104 | 633.00 | D2-53 | 687.00 |
| | DF-84 | 679.00 | V-200 | 685.00 | D3-11 | 690.00 |
| | | | V-324 | 700.00 | | |
| 3-year, 1973 | BF-65 | 663.50 | V-84 | 668.00 | | |
| | DB-22 | 656.00 | V-104 | 672.00 | | |
| | DF-84 | 673.50 | V-200 | 652.00 | | |
| | | | V-324 | 630.00 | | |
| | | | V-16 | 686.00 | | |
| | | | V-48 | 667.00 | | |
| | | | V-62 | 678.00 | | |
| | | | V-100 | 664.00 | | |
| | | | V-106 | 648.00 | | |
| | | | V-150 | 674.00 | | |
| | | | V-196 | 656.00 | | |
| | | | V-198 | 664.00 | | |
| | | | V-250 | 700.00 | | |
| | | | V-326 | 693.00 | | |
| 5-year, 1975 | 49AE | 679.00 | V-36 | 679.00 | D1-33 | 658.50 |
| | 59BD | 658.50 | V-86 | 669.00 | D1-51 | 679.50 |
| | 63BD | 685.50 | V-154 | 691.00 | D2-21 | 653.00 |
| | 80BD | 653.00 | V-202 | 662.00 | D2-49 | 644.00 |
| | 66BF | 671.50 | V-280 | 660.00 | D3-40 | 661.50 |
| | 71BF | 675.50 | | | D3-49 | 665.50 |
| | 79BF | 673.00 | | | | |
| | 56DF | 658.00 | | | | |
| | 68DF | 678.50 | | | | |
| | 76DF | 652.00 | | | | |
| 10-year, 1981 | 67BD | 682.00 | V-50 | 657.00 | D132 | 655.70 |
| | 67BF | 643.20 | V-176 | 651.80 | D245 | 624.30 |
| | 73DF | 651.60 | V-306 | 679.10 | D335 | 636.20 |
| 15-year, 1987 | 59AC | 638.00 | V-14 | 681.00 | D1-9 | 631.50 |
| | 77AC | 663.00 | V-124 | 657.00 | D1-38 | 667.00 |

| Surveillance, Date | Hoop Tendon | Force (kips) | Vertical Tendon | Force (kips) | Dome Tendon | Force (kips) |
|---|---|---|---|---|---|---|
| | 42DF | 646.00 | V-230 | 676.00 | D2-3 | 640.50 |
| | 65BF | 667.00 | V-250 | 682.00 | D3-18 | 661.50 |
| | 74BF | 633.50 | | | | |
| 20-year, 1992 | 29AE | 624.50 | V-20 | 659.00 | D1-42 | 653.50 |
| | 48AE | 702.00 | V-72 | 728.00 | D2-19 | 632.00 |
| | 52AE | 668.50 | V-128 | 680.00 | D2-23 | 637.00 |
| | 46BD | 652.50 | V-218 | 631.00 | D3-52 | 671.00 |
| | 70DF | 671.50 | | | | |
| | 77BF | 640.00 | | | | |
| 25-year, 1997 | H-68AC | 636.48 | V-26 | 691.03 | D1-29 | 648.77 |
| | | 655.34 | V-126 | 744.88 | | 646.57 |
| | H-69AE | 663.72 | V-248 | 664.86 | D2-47 | 642.37 |
| | | 642.48 | V-334 | 684.40 | | 624.95 |
| | H-26BD | 661.81 | | | D3-42 | 673.61 |
| | | 653.91 | | | | 672.71 |
| | H-72BF | 645.24 | | | D3-51 | 636.48 |
| | | 663.44 | | | | 667.21 |
| | H-28DF | 675.15 | | | | |
| | | 673.24 | | | | |
| 30-year, 2002 | H-22AE | 667.7 | V-14 | 695 | D1-18 | 657.5 |
| | | 633.7 | V-16 | 677.8 | | 656.9 |
| | H-23BD | 632.7 | V-30 | 664.7 | D1-38 | 666.0 |
| | | 625.3 | V-116 | 740.4 | | 685.9 |
| | H-24BD | 586.2 | V-302 | 669.1 | D2-43 | 653.9 |
| | | 634.4 | V-334 | 682.9 | | 655.6 |
| | H-25BD | 623.6 | | | D3-20 | 658.6 |
| | | 653.6 | | | | 659.9 |
| | H-37AC | 644.8 | | | | |
| | H-62BF | 644.8 | | | | |
| | | 676.8 | | | | |
| | H-78CE | 690 | | | | |
| | | 701 | | | | |
| | H-84DF | 642.8 | | | | |
| | | 682.2 | | | | |

**NRC RAI-4.7.5-1**

The LRA states that the reactor coolant pump fly wheel is not considered as a TLAA. However, based on past review experience, the staff believes that this is a TLAA. Please provide the TLAA evaluations in accordance with 54.21(c)(1) or provide further technical justification that it is not a TLAA.

**NMC Response to NRC RAI-4.7.5-1**

Nuclear Management Company acknowledges that the Primary Coolant Pump Flywheel Fatigue or Crack Growth Analysis, presented in LRA Section 4.7.5, can be viewed as a TLAA. Accordingly, Section 4.7.5 has been revised as follows:

**4.7.5 Primary Coolant Pump Flywheel Fatigue or Crack Growth Analysis**

The four original Palisades primary coolant pump motors were built by Allis-Chalmers. Westinghouse built an additional motor. One of the five is maintained as a spare, and any combination of four may be installed. All have flywheels at the top of the motor to provide additional rotational inertia for gradual coastdown and continued circulation, in case of a power supply loss or inadvertent trip.

A primary coolant pump flywheel could theoretically burst because of centrifugal stresses, which could produce missiles inside containment and could also damage pump seals or other pressure boundary components. This concern is the subject of Regulatory Guide 1.14. The flywheels may therefore be subject to crack growth or fatigue.

Early technical specifications required periodic, relatively frequent, inspections of primary coolant pump motor flywheels. To justify a longer inspection frequency, the Combustion Engineering Owners Group prepared report SIR-94-080. This report used a crack growth analysis of Palisades' primary coolant pump flywheels to establish acceptable limits for the flywheel inspection interval. The evaluation determined that the primary coolant pump would be subject to approximately 500 startup/shutdown cycles, and the crack growth fatigue analysis assumed 4000 cycles. It was concluded that a ten year inspection interval was acceptable since an assumed preservice flaw would not grow to a critical flaw size during the period between inspections. (In fact, the report justified that the assumed preservice flaw would not grow to critical flaw size during the entire licensed operating period, or extended licensed operating period) The NRC approved the change to the inservice inspection requirements to extend the flywheel examination frequency to once each ten years.

**Analysis**

The primary coolant pump is assumed to experience approximately 500 startup/shutdown cycles, and the crack growth fatigue analysis assumed 4000 cycles. The expected number of cycles for the 60-year extended licensed operating period, as listed in Table 4.3.1-1, is substantially less than the 500 assumed cycles.

The Fatigue Monitoring Program will ensure reanalysis or other appropriate corrective action in the unlikely event that the design basis cycle count limit is reached at any time during the extended licensed operating period.

**Disposition: 10 CFR 54.21(c)(1)(i) and 10 CFR 54.21(c)(1)(iii)**

Therefore, this item is dispositioned under 10 CFR 54.21(c)(1)(i), the analysis remains valid through the period of extended operation, and 10 CFR 54 (c)(1)(iii), the effects of aging on the intended function will be adequately managed for the period of extended operation.

In addition to the change to LRA Section 4.7.5, Appendix A is revised to include the analysis of primary coolant pump flywheels as a TLAA discussion. Accordingly, the following is added as new LRA Section A4.5.5:

**A4.5.5 Primary Coolant Pump Flywheel Fatigue or Crack Growth Analysis**

A primary coolant pump flywheel could theoretically burst because of centrifugal stresses, which could produce missiles inside containment, and could also damage pump seals or other pressure boundary components. This concern is the subject of Regulatory Guide 1.14. The flywheels may therefore be subject to crack growth or fatigue.

Early technical specifications required periodic, relatively frequent, inspections of primary coolant pump motor flywheels. To justify a longer inspection frequency, the Combustion Engineering Owners Group prepared report SIR-94-080. This report used a crack growth analysis of Palisades' primary coolant pump flywheels to establish acceptable limits for the flywheel inspection interval. The evaluation determined that the primary coolant pump would be subject to approximately 500 startup/shutdown cycles, and the crack growth fatigue analysis assumed 4000 cycles. It was concluded that a ten year inspection interval was acceptable since an assumed preservice flaw would not grow to a critical flaw size during the period between inspections. (In fact, the report justified that the assumed preservice flaw would not grow to critical flaw size during the entire licensed operating period, or extended licensed operating period) The NRC approved the change to the inservice inspection requirements to extend the flywheel examination frequency to once each ten years.

**Analysis**

The primary coolant pump is assumed to experience approximately 500 startup/shutdown cycles, and the crack growth fatigue analysis assumed 4000 cycles. The expected number of cycles for the 60-year extended licensed operating period is substantially less than the 500 assumed cycles.

The Fatigue Monitoring Program will ensure reanalysis or other appropriate corrective action in the unlikely event that the design basis cycle count limit is reached at any time during the extended licensed operating period.

**Disposition: 10 CFR 54.21(c)(1)(i) and 10 CFR 54.21(c)(1)(iii)**

The analysis remains valid through the period of extended operation and the effects of aging on the intended function will be adequately managed for the period of extended operation.