NMC
Committed to Nuclear Excellence

Palisades Nuclear Plant
Operated by Nuclear Management Company, LLC

January 13, 2006

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

NMC Response to NRC Requests for Additional Information Dated November 30, 2005 Relating to License Renewal for the Palisades Nuclear Plant

In a letter dated November 30, 2005, the Nuclear Regulatory Commission (NRC) transmitted Requests for Additional Information (RAIs) regarding the License Renewal Application for the Palisades Nuclear Plant. Enclosure 1 provide the NMC response to each NRC request.

The timing of this response was discussed with the NRC License Renewal Project Manager. The Project Manager concurred that a mid-January response to the NRC letter, in lieu of 30 days from the date of the letter, would be acceptable.

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

Summary of Commitments

This letter revises one Preliminary Commitment (i.e., subject to acceptance in the NRC SER for the renewed operating license), identified in NMC letter dated March 22, 2005, as follows:

> NMC will re-evaluate effects of primary water stress corrosion cracking for all Alloy 600 components for which the current analyses found acceptable crack sizes at 40 years to identify those for which the analysis would predict unacceptable crack sizes at 60 years, and to identify appropriate additional inspections for them. NMC will complete these re-evaluations three years before the period of extended operation.

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I declare under penalty of perjury that the foregoing is true and correct. Executed on January 13, 2006.

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

Enclosures (2)

CC Administrator, Region III, USNRC
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ENCLOSURE 1

NMC Responses to NRC Requests for Additional Information
Dated November 30, 2005

(26 pages)

ENCLOSURE 1
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RAI 4.2-1

LRA Section 4.2.1 did not provide the effective full power years (EFPYs) for the proposed 60 calendar years of operation for the plant. Please provide the EFPY for 40 calendar years of operation and the EFPY for 60 calendar years of operation for the Palisades plant.

NMC Response to NRC RAI 4.2-1

The projected EFPY on March 24, 2011, as used for estimating 40-year reactor vessel fluence, is 24.63 EFPY. This value was established to be consistent with the NRC Safety Evaluation of November 14, 2000, as follows:

> The EFPY on December 14, 1999 was 14.62 EFPY. The additional effective full power operating time during the period from December 14, 1999 until March 24, 2011 was determined to be:
>
> EFPY = (2011.23 - 1999.95) x 0.887 (capacity factor)
> = 10.01
>
> Therefore the total on March 24, 2011 was projected to be 14.62 + 10.01 = 24.63 EFPY.

NMC currently projects that reactor operation will result in only 24.17 EFPY on March 24, 2011, primarily due to the extended maintenance outage in 2001. Assuming operation beyond March 24, 2011 continues at an operating capacity of 91% for 20 years (18.2 EFPY), accumulated operation on March 24, 2031 is projected to be 42.37 EFPY.

RAI 4.2-2

Separately, it appears that the LRA end of extended license Charpy upper shelf energy (USE) values for all beltline materials are based on Position 1.2 of Regulatory Guide (RG) 1.99, Revision 2. The staff's independent calculation indicated that using Position 1.2 of the RG is non-conservative for the lower shell axial weld 3-112A/C (fabricated with weld wire Heat No. 34B009), for which a surveillance weld of the same heat is available from Millstone, Unit 1. (See your submittal dated May 25, 1994 or the NRC's Reactor Vessel Integrity Database). Provide the end of extended license USE values based on your past approach, i.e., using surveillance data from Millstone, Unit 1, to meet the requirements of the RG and to demonstrate that this revision of USE values based on surveillance data will not change your conclusion in LRA Section 4.2.1.

NMC Response to NRC RAI 4.2-2

The Millstone Upper Shelf Energy values used for comparison should be those from NUREG/CR-6551, Improved Embrittlement Correlations for Reactor Pressure Vessel Steels (November 1998). NUREG/CR-6551 represents the latest technology for determining Charpy curves from surveillance results, and reevaluated the previously published surveillance data in a consistent way using this updated methodology.

Since the referenced 1994 submittal, this NUREG has been used by Palisades and accepted by the NRC. For example, in a letter dated September 8, 1998, in response to a request for additional information regarding reactor vessel integrity, Consumers Energy reviewed the available chemistry and surveillance data applicable to the Palisades reactor vessel welds. In that response, Consumers Energy utilized the Millstone 1 surveillance test results taken from draft NUREG/CR-6551. The final NUREG was issued in November 1998. NUREG/CR-6551 also reports the results of the reevaluation of the published upper shelf energy values. The September 8, 1998 Consumers Energy letter specifically addressed a change in reference temperature, but the upper shelf energy results in this document are also applicable.

The updated upper shelf energy surveillance results for Millstone 1 as reported in NUREG/CR-6551 are:

Capsule Identification	Fluence (10^{19} n/cm^2)	Measured Upper Shelf Energy (ft-lb)
Initial	--------	111
210°	0.033	110
300°	0.066	92

When the using RG 1.99 Position 1.2 methodology, with a copper concentration of 0.19%, corresponding Palisades' values of upper shelf energy can be predicted for the same fluence. Note that this value of copper is applicable to both the Millstone and Palisades specimens, and was used for the March 24, 2031 Palisades estimates reported in the License Renewal Application (LRA). If these USE values are added to the table to permit direct comparison, the table yields:

Capsule Identification	Fluence (10^{19} n/cm^2)	Measured Upper Shelf Energy (ft-lb) (Millstone 1)	Predicted Upper Shelf Energy (ft-lb) (Palisades)
Initial	--------	111	---------
210°	0.033	110	95
300°	0.066	92	92

As can be seen, the predicted values using RG 1.99 Position 1.2 methodology are equivalent to, or more conservative than, the measured values reported in NUREG/CR-6551. Since the 60-year information in LRA Table 4.2-1 is also based on the RG 1.99 Position 1.2 methodology, the reported information is considered conservative, and will not change.

RAI 4.2-3

(1) The applicant has provided a time limited aging analysis (TLAA) which determines that the limiting material for the Palisades reactor pressure vessel (RPV) will exceed the pressurized thermal shock (PTS) screening criterion in 2014. Describe any current or planned flux reduction program which is required to support the determination that the Palisades RPV limiting material will comply with requirements of 10 CFR 50.61 until 2014.

(2) LRA Section 4.2.2 states that you will select the optimum alternative to manage PTS in accordance with 10 CFR 50.61 and provide applicable submittals for NRC review and approval, prior to exceeding the PTS screening criteria during the period of extended operation. Please provide additional information regarding specific plant equipment modifications, operational modifications, revised PTS analysis, or thermal annealing which could be implemented to allow the Palisades RPV to comply with the requirements of 10 CFR 50.61 through the end of the period of extended operation.

NMC Response to NRC RAI 4.2-3

(1) The determination that Palisades' RPV material will remain below the 10 CFR 50.61 screening criteria until 2014 is based on the Ultra Low Leakage core design that has already been implemented. Over a number of years, flux reductions have been accomplished through a series of evolutionary changes in core reload design. These changes, which have culminated in the current Ultra Low Leakage core design, have been communicated to the NRC in various submittals related to core reload designs and PTS. Most recently the Consumers Energy Company letter of August 31, 2000, on page 2, summarized these reductions in neutron fluence caused by core design changes, as follows:

> "...the current actual rate of reactor vessel fluence accumulation has been reduced significantly below the rate that existed earlier in plant life. A series of physical changes in core design have been implemented to reduce core peripheral power levels and associated neutron leakage. Core design changes to reduce neutron leakage were first instituted at Palisades more than a decade ago, and have evolved over the years to provide a progressively lower neutron flux at the reactor vessel wall. In the current analysis the fluence accumulation rate for future cycles assumes calculated flux values at the reactor vessel wall which are based on the core design of the current cycle, Cycle 15. The lower neutron leakage of the cycle 15 core design results in a predicted fluence accumulation rate which is significantly below the rate derived from the Cycle 11 core design used for future fluence projections in our previous analysis submitted on April 4, 1996. The latest NRC SER on Palisades' fluence analysis and PTS projections dated December 20, 1996, is based on this 1996 submittal. This factor alone extends the projected date for reaching the PTS screening criteria by several years beyond the date reflected in the latest NRC SER. When this fact is coupled with the various other changes that improve the accuracy of the fluence

calculation methodology, we remain confident that the reactor vessel will not approach the PTS screening criteria until after the proposed license expiration date of March 24, 2011."

Among other subjects, the discussions in Attachments 1 and 2 of the August 31 letter highlight some of the effects of core design changes. For example, the Summary Response to Concern 1 on page 1 of Attachment 1 states,

> "...peripheral assembly powers in the high leakage cores used early in plant life are three to four times higher than the peripheral assembly powers in later low leakage cores."

The NRC acknowledged the effects of these core design changes to reduce vessel fluence in its letter and Safety Evaluation dated November 14, 2000. Section 2.5 of the SER specifically discusses the current Palisades Ultra Low Leakage Core Design. The conclusion in the SER states,

> "...the planned operation with the Ultra-Low Leakage core design will provide a substantial reduction in the reactor vessel fluence during the remaining fuel cycles (Cycles 16-22) of the current operating license with its proposed extension, ..."

The projection that the Palisades reactor vessel would reach the PTS screening criteria in 2014 was reported to the NRC in the Consumers Energy Company letter dated February 21, 2000. The NRC Safety Evaluation of November 14, 2000, acknowledged this date on page 8, as follows:

> "In its February 21, 2000 letter, the licensee requested that the NRC consider the information that has been discussed above regarding the licensee's proposed methodology for establishing reactor vessel fluence values and reevaluate the status of the Palisades Plant's reactor vessel with respect to the requirements in 10 CFR Part 50.61 on PTS. Specifically, the licensee requested that the NRC staff 'endorse the new date at which the reactor vessel is estimated to reach the PTS screening criteria.' The new date cited by the licensee is the year 2014."

While the NRC letter of November 14, 2000, did not explicitly endorse the 2014 date since it was beyond the license expiration date, it did approve the underlying methodology and the calculations of vessel fluence that were used to predict that date.

(2) Section 4.2.2 of the LRA states, "The licensee can submit a safety analysis pursuant to §50.61(b)(4) to determine what, if any, modifications to equipment, systems and plant operation are necessary to prevent failure of the reactor vessel from a postulated PTS event." 10 CFR 50.61 requires this analysis to be submitted at least three years prior to exceeding the screening criteria. This safety analysis determines "...what, if any, modifications to equipment, systems, and operation are necessary to prevent potential failure of the reactor vessel as a result of postulated PTS events if continued operation beyond the screening criterion is allowed."[1] Until such an analysis is performed, NMC

[1] 10 CFR 50.61(b)(4)

can not specify with certainty what the modifications might be.

One example of a potential operational modification that could eliminate the PTS event as a concern would be to preheat the safety injection water to reduce the differential temperature between the cooling water and the reactor vessel. This would reduce the resulting stress and the associated potential for propagating an assumed flaw. Another example of a potential equipment modification would involve the installation of circular bands placed around the circumference of the reactor vessel that would be tightened to provide a compressive prestress in the vessel beltline region. This would offset the stress induced during a pressurized thermal shock scenario, making an assumed flaw unable to propagate. These and/or other physical and operational changes would be evaluated for practicality and effectiveness during development of this safety analysis. The proposed change(s) would be submitted for NRC review and approval prior to plant operation beyond the PTS screening criteria in accordance with the rule.

Thermal annealing restores the ductility of the reactor vessel, but, because of an annealing project's scope and occupational radiation exposure, it is likely to be less desirable than other options. A thermal annealing plan for the Palisades reactor vessel was prepared and submitted to the NRC in a series of letters beginning on October 12, 1995. This application was later withdrawn when a refined fluence analysis showed an increased margin to the PTS screening criteria because the accumulated fluence from early plant operating cycles was lower than previously calculated. Annealing was determined not to be necessary for the plant to operate for the remainder of the license term. If, in the future, it is concluded that a thermal annealing treatment is the best method for showing reactor vessel compliance with 10 CFR 50 requirements through the extended operating period, then a thermal annealing plan will be submitted for NRC review and approval in accordance with 10 CFR 50.66.

It is possible that a revised PTS evaluation methodology will be available by the time action is required under 10 CFR 50.61. Technology that makes reanalysis of the PTS event a viable option for confirming reactor vessel acceptability through the extended operating period will be considered if available.

RAI 4.7.2-1

LRA Section 4.7.2 Table 4.7.2-1 provides a summary of the fracture mechanics assessment of the most susceptible pressurizer and primary coolant system Alloy 600 locations. Please:

(1) Confirm that all the components in Table 4.7.2-1 are inspection items under Palisades' Alloy 600 Aging Management Program (AMP).

(2) Clarify how this fracture mechanics assessment has been used to support the inspection methods and intervals of the pressurizer and primary coolant system Alloy 600 locations and how the fracture mechanics assessment is used to support the inspection methods and intervals of the Palisades Alloy 600 AMP.

(3) Provide justification for selecting 0.01 inch (LRA Page 4-54) as the postulated initial flaw depth in your fracture mechanics analyses and provide the flaw depth that you consider to be detectable by approved methods (LRA Page 4-56) and the basis for your determination.

Further, it appears that the fracture mechanics assessment summary is based on Report 32-1238965-00, "Fracture Mechanics Assessment of Palisades Alloy 600 Components, "...which was approved by the NRC in a safety evaluation (SE) dated June 27, 1995. In recent years, the industry started a systematic approach to manage reactor vessel head and pressurizer Alloy 600 penetration nozzles, e.g., the work related to crack growth rate in MRP-55 (July 18, 2002), "Crack Growth Rates for Evaluating Primary Water Stress Corrosion Cracking (PWSCC) of Thick-Wall Alloy 600 Material," and the fatigue growth rate of Appendix O, "Evaluation of Flaws in PWR Reactor Vessel Upper Head Penetration Nozzles," to Section XI of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code). Please assess the impact of this new information on the fracture mechanics assessment of Palisades Alloy 600 components summarized in LRA Table 4.7.2-1.

NMC Response to NRC RAI 4.7.2-1

(1) All the components in Table 4.7.2-1 are inspection items under Palisades' Alloy 600 aging management program (AMP). These items were described in the Alloy 600 project plan approved by NRC on June 27, 1995.

(2) The subject fracture mechanics assessment is discussed in the NRC's June 27, 1995 safety evaluation. Significant items affected by the analysis are that the pressurizer spray nozzle safe end is inspected every other refueling outage and pressurizer temperature element nozzles are inspected each refueling outage.

(3) A value of 0.01 inch was selected as the postulated initial flaw depth in the fracture mechanics analyses. This is smaller than detectable by approved NDE methods. The NRC's June 27, 1995 SER, on page 8, estimates that the ultrasonic examination

detection limit would be 2 mm (0.079 in.). Because of this, the inspection frequencies have not been extended based on some predicted propagation rate. The pressurizer spray nozzle safe end is inspected every other refueling outage and pressurizer temperature element nozzles are inspected each refueling outage.

In 2004, the Combustion Engineering Owners Group performed a generic fatigue crack growth evaluation associated with small diameter nozzles. This evaluation looked at limiting pressurizer and hot leg nozzle locations. Only two transients were determined to contribute to fatigue crack growth. The transients and the numbers of cycles assumed in the analysis for each are:

Start-up/Shut-down (Normal)	500 Events
OBE (Hot Leg only) (Upset)	200 Events

The initial crack depths evaluated were 0.938" for the hot leg, 0.719" for the pressurizer lower side shell and 1.125" for the pressurizer lower head. None of these cracks propagated substantially for the entire operating period as defined by the above transients. The disposition at the bottom of page 4-60 reads, "The Palisades plant-specific bounding fracture mechanics analysis demonstrates the validity of the cycle-dependent aspects of the generic bounding fracture mechanics analysis (WCAP-15973-P) by demonstrating that the plant-specific load and thermal events are within those assumed by the generic bounding analysis. The basis for the safety determination of the fracture mechanics evaluation calculation will therefore remain valid so long as the numbers of these events do not exceed the design basis values."

Note: There is a typographical error in Table 4.7.2-1 line 12. The Limiting Allowable Flaw Depth for an 18-Month Cycle for the Cold Leg Pressure and Sampling Nozzle is hereby corrected from 1.180 in. to 0.180 in.

RAI 4.7.2-2

On Page 4-57, concerning the pressurizer temperature element nozzles which were repaired in 1993, LRA Section 4.7.2 states that, "[t]he fatigue analysis of the weld pad repairs and the ASME XI Appendix A crack growth evaluation are therefore TLAAs." Please:

(1) Provide the actual cycle count recorded in the fatigue monitoring program for the pressurizer events from 1993 to date as a fraction (e.g., 7/10) of the 20-year pressurizer design basis event cycles that were assumed in the weld pad fatigue analysis to determine whether the action levels for cycles will be reached by major events early in the extended period of operation. If this is likely to happen, describe the actions to be taken.

(2) Provide information about the pad material and the pad weld material, and confirm whether the repair was in accordance with Section XI of the ASME Code and whether the welded pad was analyzed in accordance with Section III of the ASME Code.

(3) Provide information about the inspection results for the repair since 1993 and discuss the consistency of the inspections for the repair with those for pressurizer temperature element nozzles in the Alloy 600 AMP for the extended period of operation.

(4) Identify the submittal and the NRC SE regarding the corrosion evaluation (Page 4-58) for the extended 60-year licensed operating period and provide a consequence assessment of the pressurizer temperature element nozzle having a predicted bore diameter increase of 0.28 inch due to corrosion for the extended 60-year licensed operating period.

NMC Response to NRC RAI 4.7.2-2

(1) The license renewal application did not discuss the fatigue crack growth analysis of the weld pad repairs that had been prepared in 2003. The analysis determined that only pressurizer cool-down and system leak tests contribute to fatigue crack growth at these locations. The analysis assumed 500 cool-downs and 320 leak tests. In LRA Table 4.3.1-1, the maximum number of cool-downs for 60 years is determined to be 240 (240/500 = 0.48) and the maximum number of leak tests for 60 years is 10 (10/320 = 0.03125). Fatigue failure during the extended operating period of the weld pad repairs to the pressurizer temperature element nozzles is judged not to be a concern.

(2) The weld pad is deposited with SFA 5.11 ENiCrFe-3 welding electrodes as specified in ASME Section II, Part C. This filler material is compatible with both the alloy 600 (P43) nozzle and the vessel base material, SA-533 Grade B Class 1 for the top head nozzle and the A-10 composition weld pad of the side shell nozzle. The welding filler

material, SFA 5.11 ENiCrFe-3, conforms to the requirements of Sections II and III 1986 Edition and is compatible with the materials being joined. This weld material (ENiCrFe-3) provides corrosion resistance in the primary water environment and meets the strength requirements for design under ASME Section III. Repair of the temperature element nozzles was performed in accordance with the requirements of ASME Section XI 1983 with Summer 1983 addenda. The welded pad was analyzed in accordance with ASME Section III, 1965 with Winter 1966 addenda.

(3) The most recent inspection of the upper temperature element nozzle was the bare metal visual performed during the 2004 refueling outage. This inspection was acceptable and is performed each refueling outage. There have been no corrective actions related to the condition of this nozzle since the 1993 repair. The inspection will be continued through the license renewal period.

(4) This corrosion analysis has been superseded. The revised analysis was submitted in a letter dated on August 9, 2004 (ML042240249) entitled "Request for Relief from ASME Section XI Code Requirements for Repair of Pressurizer Nozzle Penetrations." The supporting calculations determined the allowable maximum hole size in the vessel wall, under ASME code rules for the particular design of the lower TE-0101 nozzle, and the allowable corrosion as the difference between that value and the vessel wall bore for the nozzle. This, with an estimated corrosion rate (from WCAP-15973-P), resulted in an estimated repair lifetime of 52.3 years; that is, an estimated corrosion life of 52.3 years following initiation of leakage.

RAI 4.7.2-3

On Page 4-59, concerning the pressurizer spray and surge nozzle service time assessment, you concluded that the predicted service time for the surge nozzle is 40 years. Provide your disposition for the surge line nozzle TLAA, because it's not clear that it is included in "Disposition for all Alloy 600 Heater Sleeves, Nozzles, Safe Ends, and Flanges: 10 CFR 54.21(c)(1)(iii)." For the spray nozzle which has a predicted service time of 5.36 years at 640°F, provide information regarding the inspection results for the spray nozzle since 1995 and discuss the consistency of the current inspections for the spray nozzle with those for the spray nozzle in the Alloy 600 AMP for the extended period of operation.

NMC Response to NRC RAI 4.7.2-3

Disposition for the pressurizer surge line nozzle is in accordance with 10 CFR 54.21(c)(1)(iii). The Palisades Alloy 600 Program includes the surge line nozzle in the inspection program.

For the spray nozzle, the Alloy 600 Program requires a bare-metal VT-2, volumetric, or penetrant inspection every other refueling outage. Volumetric examination was performed during the most recent inspection during the 2004 refueling outage. No corrective action documents have been identified related to leakage or structural indication issues regarding this nozzle.

RAI 4.7.2-4

On Page 4-59, regarding the bounding fracture mechanics analysis of the hot leg, piping resistance temperature detector and sampling nozzles, pressurizer instrument nozzles, and pressurizer heater sleeves, LRA Section 4.7.2 states that, "[t]he bounding fracture mechanics portion of the analysis employs elastic-plastic methods with IWB 3600 and Regulatory Guide 1.161 acceptance criteria." The staff believes that this bounding fracture mechanics analysis refers to that of WCAP-15973, "Low-Alloy Steel Component Corrosion Analysis Supporting Small-Diameter Alloy 600/690 Nozzle Repair/Replacement Programs." Therefore, the quoted LRA Section 4.7.2 sentence must be modified by adding to its end, "as modified by the NRC SE dated January 12, 2005." The Regulatory Guide 1.161 acceptance criteria, especially the structural factors, are not acceptable for analyzing detected flaws, including flaws with indication of leakage.

NMC Response to NRC RAI 4.7.2-4

The first sentence of the last paragraph on page 4-59 is in error. The sentence is hereby revised to read, "The bounding fracture mechanics portion of the analysis employs elastic-plastic methods with IWB 3600 and Regulatory Guide 1.161 acceptance criteria, as modified by the NRC Safety Evaluation dated January 12, 2005."

RAI 4.7.2-5

On Page 4-61, concerning disposition for all Alloy 600 heater sleeves, nozzles, safe ends, and flanges, LRA Section 4.7.2 states that, "[l]ocations which are more susceptible to PWSCC, or whose failure could result in a more-significant safety hazard, are also subject to initial or periodic bare-metal VT-2, volumetric, or penetrant inspections." Please provide a list of the "more susceptible" locations mentioned above, the criteria for determining the more susceptible locations, and the development of these criteria based on the fracture mechanics assessment of Report 32-1238965-00.

NMC Response to NRC RAI 4.7.2-5

The locations most susceptible to PWSCC are:

- Surge nozzles safe ends
- Pressurizer temperature element nozzles
- Pressurizer relief valve safe ends
- Heater sleeves

The 251 Alloy 600 penetrations are ranked based on four main criteria: primary water stress corrosion cracking (PWSCC) susceptibility, failure consequence, leakage detection margin, and radiation dose rates.

PWSCC susceptibility is the most important criteria for ranking. The degree of susceptibility takes into account the material heat treatment, chemical composition and fabrication; it accounts for the operating environment; and it also takes into account tensile stress.

To a lesser degree, the following are also considered and may influence inspection timing:

- Failure consequence considers the implications of the potential failure. A failure that would cause a LOCA would be of greater concern than a leak requiring plant shutdown to repair.
- Leakage detection margin considers potential crack orientation, component inner diameter and material toughness to prioritize the likelihood of detecting a leak before break.
- Radiation dose rate takes into account the concept of ALARA.

The above criteria were developed in 1993, subsequent to the identification of cracking in the pressurizer power-operated relief valve (PORV) nozzle. The NRC has reviewed and accepted these criteria in the letter and Safety Evaluation of June 27, 1995.

RAI 4.7.2-6

On Page 4-61, LRA Section 4.7.2 states that, "NMC will re-evaluate effects of primary water stress corrosion cracking for all Alloy 600 components for which the current analyses found acceptable crack sizes at 40 years to identify those for which the analysis would predict unacceptable crack sizes at 60 years, and to identify appropriate additional inspections for them. NMC will complete these re-evaluations before the period of extended operation." Due to availability of new information on PWSCC crack growth rates and fatigue crack growth rates, the staff determines that this re-evaluation should be performed three years before the period of extended operation, instead of just before the period of extended operation. Please revise your commitment letter to reflect the change of the submittal date for this re-evaluation.

NMC Response to NRC RAI 4.7.2-6

The commitment in Enclosure 2 of the March 22, 2005 "Application for Renewed Operating License," and the last paragraph in Section 4.7.2 of the License Renewal Application, are hereby revised to read, "NMC will re-evaluate effects of primary water stress corrosion cracking for all Alloy 600 components for which the current analyses found acceptable crack sizes at 40 years to identify those for which the analysis would predict unacceptable crack sizes at 60 years, and to identify appropriate additional inspections for them. NMC will complete these re-evaluations <u>three years</u> before the period of extended operation."

RAI B2.1.4-1

LRA AMP B2.1.4, "Boric Acid Corrosion Program," states that the program identifies components exhibiting boric acid leakage, evaluates the acceptability for the continued service of these components, performs trending and tracking, and recommends corrective actions. GALL XI.M10, "Boric Acid Corrosion," requires the determination of the principal location of leakage (prioritization). Discuss your program's classification of components based on their susceptibility to corrosion from boric acid leakage and your program's determination of the scope and frequency for visual and other nondestructive examination (NDE) inspections for these components. Please also provide information regarding provisions for managing potential boric acid leakage in inaccessible locations and areas covered by external insulation surfaces.

NMC Response to NRC RAI B2.1.4-1

Classification of components based on their susceptibility to corrosion from boric acid leakage

The Palisades program philosophy is that all leakage from systems containing boric acid should be minimized. Once a leak occurs our program addresses each leak by the same process with the underlying assumption that any components that may be exposed to the leakage could be susceptible to degradation from boric acid. The plant Boric Acid Corrosion Control Program procedure identifies the Primary Coolant System, Engineered Safeguards Systems, Chemical and Volume Control System and Liquid Radioactive Waste Systems as those with the highest probability of experiencing leakage of boric acid. The portions of these systems classified as ASME class 1, 2 or 3 are inspected at the frequencies specified in the ASME Section XI Inservice Inspection and Boric Acid Corrosion Control Program (BACCP) commitments. The ASME Section XI Code establishes the standards for inspection of code components based on their ASME classification. This includes criteria for visual or volumetric inspections as well as hydrostatic and leakage testing. Hydrostatic and leakage testing specified by the ASME Section XI Code establishes requirements to look for and determine the principle locations of leakage.

Those systems containing boric acid that are non-ASME Section XI class 1, 2 or 3 are inspected during system walkdowns. As a general practice the plant operations staff also looks for leakage from all plant systems on a continuing basis, and initiates corrective actions when appropriate to drive resolution of the leak.

Program determination of the scope and frequency for visual and other nondestructive examination (NDE) inspections

All in-scope systems receive periodic inspection in accordance with In-Service Inspection Pressure Testing, the Boric Acid Corrosion Control Program, and/or the System Monitoring Program. Generally, components are visually examined once per fuel cycle, except Primary Coolant System components, which may receive a mid cycle inspection (not necessarily VT-2) if conditions allow. Otherwise, indicators of primary

coolant system leakage, such as containment pressure, humidity, radiation, sump level, air cooler drainage, and visual observations during containment entries are monitored for leak indicating trends. Areas outside of containment are generally accessible by operations, maintenance and engineering personnel during routine activities. The Boric Acid Corrosion Program credits the System Monitoring Program walkdowns and inspections for signs of boric acid leakage, residue or degradation of mechanical systems/ components. Electrical connectors are specifically inspected on a periodic frequency for boric acid residue and degradation.

Provisions for managing potential boric acid leakage in inaccessible locations and areas covered by insulation surfaces.

Leaks from inaccessible locations are managed in accordance with the plant's corrective action program and in accordance with Technical Specifications and Inservice Inspection requirements. If evidence of leakage is indicated at an inaccessible location, that leakage will be located and corrective actions taken. If continued operation is allowed, justification will be provided based on approved industry standards.

A detailed discussion of insulated or inaccessible locations was provided in NMC's January 20, 2003, response to the NRC's November 18, 2002 request for additional information (RAI) concerning the 60 Day response to NRC Bulletin 2002-01,"Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure boundary Integrity". For convenience, the information requests and responses are repeated below.

NRC Request

Provide the technical basis for determining whether or not insulation is removed to examine all locations where conditions exist that could cause high concentrations of boric acid on pressure boundary surfaces or locations that are susceptible to primary water stress corrosion cracking (Alloy 600 base metal and dissimilar metal Alloy 821/82 welds). Identify the type of insulation for each component examined, as well as any limitations to removal of insulation. Also, include in your response actions involving removal of insulation required by your procedures to identify the source of leakage when relevant conditions (e.g., rust stains, boric acid stains, or boric acid deposits) are found.

Response

The technical basis for determining whether or not insulation is removed to examine all locations where conditions exist that could cause high concentrations of boric acid on pressure boundary surfaces or locations that are susceptible to PWSCC is provided in the ASME B&PV Code, Section XI, 1989 Edition. IWA-5242 requires that systems borated for the purpose of controlling reactivity shall have insulation removed from pressure retaining bolted connections in order to complete visual examination, VT-2. NMC considers portions of the Palisades engineered safeguards system (ESS) Chemical and Volume Control System (CVCS), the Primary Coolant System (PCS) and Spent Fuel Pool Cooling System

(SFP) as systems borated for the purpose of controlling reactivity. These VT-2 examinations are performed at the frequency specified in ASME B&PV Code, Section XI for system pressure tests. During regularly scheduled inservice inspection activities, insulation is removed as necessary to complete the specified inspection or examination technique.

In accordance with Palisades' procedures, new boric acid accumulations shall be documented by a maintenance work order request until the boric acid accumulation is removed and necessary repairs are completed.

In accordance with Palisades' corrective action process, an action request (AR) shall be initiated upon discovery of equipment malfunction, damage, or degradation that is considered sudden or unexpected. Per plant procedures, the following indications related to boric acid accumulations shall be documented by initiation of an AR:

a. Through-wall leakages identified from cracks or weld defects.
b. Any leakage from recirculation heat removal Systems (high pressure safety Injection (HPSI), low pressure safety injection (LPSI) and containment spray system (CSS)) outside of containment including leaks from seats, seals, valve stems, pump seals, vessel flange gaskets, and other mechanical joints which result in total leakage greater than 0.2 gpm (756 ml/min).
c. Degradation of fastener material, which may reduce cross sectional area greater than or equal to five percent.
d. Degradation of pressure boundaries, which may reduce wall thickness greater than or equal to 10 percent.
e. Other conditions adverse to quality not specifically described in the procedure.

Inspections conducted during corrective action planning and implementation activities include an assessment of condition for areas contacted by boric acid. This assessment includes the following:

a. Boric acid accumulation location(s).
b. Boric acid accumulation source(s).
c. If degradation OR corrosion is evident.
d. If boric acid leak is active (wet leakage) or inactive (minor dry residue).
e. If leakage has contacted other components.
f. If the source of boric acid accumulation is due to a component failure other than packing, flange OR threaded connection leaks (i.e., cracked fittings, welds, components).
g. Actions taken.

Based upon this assessment and in accordance with Palisades' work planning processes, corrective actions are planned and implemented to address all

equipment affected by boric acid, including removal of insulation and inspection of potentially affected carbon steel surfaces.

Insulation removal limitations are unique for each type of location and are dependent on the elevation of the location above floor level and proximity of the location to radiation sources, such as the PCS. These limitations are considered when planning examinations for specific locations. Due to the proximity of each of the Alloy 600 locations to the PCS, radiation dose is of primary concern.

NRC Request

Describe the technical basis for the extent and frequency of walkdowns and the method for evaluating the potential for leakage in inaccessible areas. In addition, describe the degree of inaccessibility, and identify any leakage detection systems that are being used to detect potential leakage from components in inaccessible areas.

Response

The technical basis for the extent and frequency of walkdowns and the method for evaluating the potential for leakage in inaccessible areas is provided in accordance with the safety evaluation (SE) dated June 28, 1996, entitled "Palisades Plant - Evaluation of the Third 10-Year Inspection Program Plan Requests for Relief NOS PR-02 and PR-04.'1 The SE specifies an alternative to ASME B&PV Code, Section XI, 1989 Edition, which requires VT-2 visual examination at nominal operating pressure and temperature for all portions of the PCS. In the SE, the NRC approved determination of leakage from piping and components under the RPV in accordance with paragraph IWA-5244, Buried Components, of ASME B&PV Code, Section XI, 1989 Edition, no Addenda. This requirement is satisfied by conducting PCS leak rate calculations in accordance with plant procedures. The detection systems used to conduct PCS leak rate calculations include the containment sump level and RPV flange leak off.

In addition to the leak rate calculation, and as a condition of relief request approval, the NRC invoked the performance of a VT-2 visual examination for evidence of leakage in the RPV cavity, during Mode 5 or 6, once per refueling cycle. The cylindrical portion of the RPV located within the reactor cavity is inaccessible for direct visual examination due to obstructions, heat stress and radiation hazards. Connections located in this area include four cold leg and two hot leg connections. The bottom head of the RPV has no penetrations and is accessible for visual examination. Procedures require that evidence of leakage be documented in accordance with the site's corrective action process and dispositioned prior to unit restart.

RAI B2.1.4-2

LRA AMP B2.1.4 lists a number of aging related issues which were addressed in various NRC and industry communications and indicates that the operating experience related to these issues have been incorporated into the AMP as applicable. For staff assessment of this incorporation, please provide information about the program improvements directly related to lessons learned from the Davis-Besse vessel head degradation and the control rod drive mechanism penetration cracking discussed in NRC Bulletin 2001-01, "Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles," NRC Bulletin 2002-01, "Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity," NRC Bulletin 2002-02, "Reactor Pressure Vessel Head and Vessel Head Penetration Nozzle Inspection Programs," and NRC Order EA-03-009. Also, provide a discussion using examples of implementation of corrective actions in the program to prevent the recurrence of degradation caused by boric acid leakage, as required by NRC Generic Letter 88-05, "Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components in PWR plants."

NMC Response to NRC RAI B2.1.4-2

This subject has been addressed extensively in a variety of docketed letters from NMC to the NRC. During a telephone discussion about this question, the NRC staff indicated that a listing of the various docketed letters that replied to these bulletins and orders would provide an acceptable response. Accordingly, a listing of the associated major NMC submittals, including their Accession numbers, is provided below. Each entry also includes a short description of the submittal or program improvements directly related to lessons learned from the Davis Besse vessel head degradation or Palisades-specific control rod drive mechanism penetration cracking.

When boric acid leakage is identified, the source is repaired to stop the leakage. Two examples are highlighted where the corrective actions included a design or material change to reduce the probability of future boric acid leakage. In both of these instances the identified leakage was from the primary coolant system, and the corrective actions included replacement of the leaking components with components less likely to degrade in the operating environment and cause recurring exposure of the surrounding area to boric acid. First, the plant's 45 stainless steel control rod drive seal housings were replaced with Alloy 600 seal housings as a better material choice in the operating environment. Second, the design of eccentric reducers in the 45 control rod drive upper housing assemblies was modified to move a weld farther from the reducer to eliminate sensitization from welding in an area where large stresses exist due to the configuration of the reducer.

<u>Correspondence Relating to NRC Bulletin 2001-01, "Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles, August 3, 2001</u>

- NEI Letter to the NRC Dated August 21, 2001 Contained EPRI MRP-48 "PWR Materials Reliability Program Response to NRC Bulletin 2001-01.

(ML012350150) -- This document provided the results of a study that ranked each participating reactor for the potential for PWSCC of reactor head top nozzles. For those plants without identified leaks from the reactor head top nozzles, the ranking was an estimate of how many effective full power years a plant could run before they would expect to see leakage. This report helped define how long it might be before reactor head top nozzle leaks might be experienced.

- NMC Letter to the NRC dated August 31, 2001 Responded to NRC Bulletin 2001-01 **(ML012550049)** -- This letter reported that NMC was participating in the Materials Reliability Program (MRP) integrated response to NRC Bulletin 2001-01, and provided the plant specific information related to the bulletin. This letter noted that a general visual inspection of the head would be completed in accordance with the plant Generic Letter 88-05 commitments.

- NRC Letter dated November 8, 2001, Bulletin Acknowledgement Receipt and Reminder **(ML013060186)** -- This letter acknowledged that our August 31, 2001 response provided the requested information and reminded NMC of the intended content of our post outage inspection report.

- NMC Letter Dated March 29, 2002 Updated the Response to Bulletin 2002-01 **(ML021050154)** – This letter revised a portion of the response to Bulletin 2002-01 based on a later re-evaluation of the Palisades Reactor Head temperature. This letter included a shorter predicted full power year prediction and also resulted in a commitment to perform a 100% effective visual examination of the reactor head upper metal surface in accordance with Bulletin 2001-01 during the next refueling outage. No lessons learned related to the boric acid program from this transfer of information, although the focus for confirmation for non-leakage of the reactor head penetrations is intensified.

- NMC Letter dated October 22, 2002 Updated the Response to Bulletin 2002-01 **(ML023030471)** -- This letter informed the NRC that the Effective Full power year evaluation number needed to be changed from the 22.5 number previously reported to 23.9 years. This change did not affect the susceptibility group categorization and resulted in no new commitments and no revisions to existing commitments.

<u>Correspondence Related to NRC Bulletin 2002-01, "Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity," March 18, 2002</u>

- NMC Letter Dated April 3, 2002 provided the 15 Day Response To Bulletin 2002-01 **(ML020990462)** -- This letter provided the information requested and included detailed descriptions of previous reactor vessel head inspection activities as well as a reference to the previous commitment to perform a 100% effective visual examination of the reactor head upper metal surface during the next refueling outage, currently planned for spring 2003.

- NMC Letter dated May 16, 2002 provided a 60 Day Response To Bulletin 2002-01 **(ML021570032)** – This letter provided a detailed response to the bulletin item 3.A "the basis for concluding that your boric acid inspection program is providing reasonable assurance of compliance with the applicable regulatory requirements discussed in generic letter 88-05 and this bulletin." This information confirmed that our boric acid inspection program provides reasonable assurance of compliance with the referenced regulatory requirements. The lesson learned here is that our boric acid program was properly focused.

- NRC Letter Dated May 28, 2002 Acknowledged Receipt of 15 Day Letter **(ML021370231)** – This letter provides the NRC acknowledgement of our 15 day response to the bulletin.

- NMC Letter dated June 25, 2002 provided the 15 Day Response Supplemental Information **(ML021980504)** – This letter provided additional information that the 1995 visual inspection of the upper reactor head surface was a VT-2 examination completed with the insulation removed.

- NRC Letter Dated November 18, 2002 Request For Additional Information **(ML023080098)** – This RAI requested information related to how our program meets the expectations defined in this letter.

- NMC Letter Dated January 20, 2003 provided the 60 Day Response to NRC's 11/18/02 letter **(ML030360244)** – This letter provides the detailed response to the NRC's November 18, 2002 RAI, which provides detailed information concerning our Boric Acid Corrosion Control (BACC) program and how it meets the expectations laid out in the NRC letter.

<u>Correspondence Related to NRC Bulletin 2002-02, "Reactor Pressure Vessel Head and Vessel Head Penetration Nozzle Inspection Programs," August 9, 2002</u>

- NMC Letter dated August 26, 2002 provided the 15 day response to Bulletin 2002-02 **(ML-022480317)** – This letter provided the requested information and included a commitment to continue to evaluate non-visual nondestructive examination (NDE) of the reactor pressure vessel head and vessel head penetration nozzles. NMC committed to complete this review and select appropriate NDE method(s), and to provide the remaining requested information as described in Bulletin 2002-02 at least nine months prior to the refueling outage that followed the next refueling outage.

<u>Correspondence Related to NRC Order EA-03-009, February 11, 2003</u>

- NMC Letter Dated March 3, 2003 Notice Of Future Relaxation Request **(ML030710460)** – This letter acknowledged the order and set a timetable and parameters for response.

- NMC Letter dated May 22, 2003 provided a Response For 2003 Refueling Outage **(ML031490263)** – This letter reports results of bare metal visual inspection of 100% of reactor vessel surface (including 360 degrees around each RPV head penetration nozzle). The inspection showed some carbon steel scaling, a boric acid stain from a previous external leak source, and no accumulation of boric acid in the vicinity of any of the 54 RPV head penetrations, and no leakage of boric acid through any of the 54 RPV head penetrations. The lesson learned is that this inspection validates the effectiveness of the Boric Acid Corrosion control program.

- NMC Letter dated September 18, 2003 Relaxation Request **(ML032731380)** – This letter requests relaxation of the NRC Order EA-03-009 requirement. This letter was later withdrawn by the March 8, 2004 letter.

- NRC Letter Dated February 20, 2004 First Revised Order (EA-03-009) **(ML040220181)** – This letter transmits the revised NRC order for EA-03-09 establishing interim inspection requirements for reactor pressure vessel heads at pressurized water reactors.

- NMC Letter Dated March 8, 2004 Withdrawal of Relaxation Request **(ML041610288)** – This letter withdraws the September 18, 2003 Relaxation request.

- NRC Letter Dated September 13, 2004 NRC Agrees With Our Withdrawal **(ML042530359)** – This NRC letter concurred with our withdrawal of the relaxation request.

- NMC Letter dated January 13, 2005 provided the 60 Day Report per Revised Order **(ML050960547)** – This letter conveys the results of the fall 2004 reactor head inspection results which included ultrasonic examination of each RPV head penetration. Based on results a bare metal head examination was also performed. Ultrasonic results identified two penetrations with leak path indications. Both penetrations were repaired. Due to the discovery of the leak path indications and subsequent repair, the Palisades RPV head is now in the high susceptibility category. The enhanced inspections have been successful at identifying leak paths.

<u>Correspondence Related to NRC Bulletin 2004-01, "Inspection of Alloy 82/182/600 Materials Used in the Fabrication of Pressurizer Penetrations and Steam Space Piping Connections at Pressurized Water Reactors," May 28, 2004</u>

- NMC Letter dated July 26, 2004 60 Day Response to Bulletin 2004-01,"Inspection of Alloy 82/182/600 Materials Used in the fabrication of Pressurizer Penetrations and Steam Space Piping Connections at Pressurized Water reactors" **(ML042100242)** – This letter provides the 60 day response to the bulletin and included four new commitments related to pressurizer inspections.

- NMC will perform a bare metal visual inspection of 100 percent of all pressurizer heater sleeve locations, in a manner that visual access to the bare metal 360 degrees around each sleeve can be attained each refueling outage at the Palisades Nuclear Plant.

- NMC will perform non-destructive examination (NDE) capable of characterizing crack orientation of all sleeves for which visual inspection shows evidence of leakage at Palisades Nuclear Plant. The NDE will be performed prior to any repairs.

- NMC will notify the NRC immediately if the NDE defines the flaw as potential circumferential primary water stress corrosion cracking (PWSCC) in either the pressure boundary or non-pressure boundary portions of any locations covered under the scope of Bulletin 2004-01 for the Palisades Nuclear Plant. An appropriate inspection plan will be developed, which will define additional sleeves to be inspected by NDE, sufficient to determine the extent of condition commensurate with the characterization of the flaw.

- NMC will perform bare metal visual inspections of all Alloy 82/182/600 primary system pressure boundary locations normally operated at greater than or equal to 350°F within the next two refueling outages for the Palisades Nuclear Plant.

- "NMC letter dated November 9, 2004. "Clarification of 60-Day Response to Bulletin 2004-01, "Inspection of Alloy 82/182/600 Materials Used in the Fabrication of Pressurizer Penetrations and Steam Space Piping Connections at Pressurized-Water Reactors" **(ML 043210174)** -- This letter clarified one of the commitments contained in our July 26, 2004 letter.

  - NMC will perform bare metal visual inspections of all Alloy 82/182/600 primary system pressure boundary locations normally operated at greater than or equal to 350°F within the next two refueling outages for the Palisades Nuclear Plant. *NMC will then reevaluate the inspection program for the Palisades Nuclear plant to determine the appropriate frequency for each Alloy 82/182/600 location.*

- NMC Letter dated January 13, 2005," 60-Day Report per Bulletin 2004-01" **(ML050190263)** -- This letter conveyed the results of the inspections completed under the Generic letter. During the 2004 refueling outage, a bare metal visual inspection of all 120 pressurizer heater sleeves (J-groove welds) was performed. This examination included 360° around each sleeve. There was no accumulation of boric acid in the vicinity of any of the penetrations. All visual examinations of the penetrations had acceptable results. The lesson learned is that this inspection validates the effectiveness of the Boric Acid Corrosion control program.

During the 2004 refueling outage, a bare metal visual examination of the following Alloy 82/182/600 primary system pressure boundary locations, normally operated at greater than or equal to 350°F, was performed:

* Eight pressurizer level taps (butt welds)
* Three pressurizer safety relief valve nozzles (butt welds)
* Two pressurizer temperature element penetrations (pad weld)
* One pressurizer PORV nozzle (butt welds)
* Ten hot leg temperature element penetrations (J-groove weld)
* Twelve cold leg temperature element penetrations (J-groove ` weld)

Qualified VT-2 examiners, using direct visual techniques, performed the examinations in accordance with a qualified NDE procedure. Results of the bare metal visual examination of the Alloy 600 penetrations listed above, including 360° around each penetration, were acceptable, with no accumulation of boric acid in the vicinity of any of the penetrations. The remaining Alloy 82/182/600 primary system pressure boundary locations normally operated at greater than or equal to 350°F will be examined during the next refueling outage in 2006.

RAI B2.1.16-1

LRA AMP B2.1.16, "Reactor Vessel Integrity Surveillance Program," states that Enhancement 1 to this AMP requires that Palisades pressure-temperature (P-T) limits and low temperature overpressure protection (LTOP) curves be updated and submitted to NRC for review and approval prior to the period of extended operation to reflect the additional neutron fluence accumulated during the extended operating period. Please provide the approximate date of withdrawal for the next surveillance capsule (Capsule W-280) and confirm that the updated P-T limits and LTOP curves to be submitted to the NRC prior to the period of extended operation will incorporate information from the surveillance report on irradiated specimens from Capsule W-280. Further, you stated that this program ensures the reactor vessel materials "have adequate margins against brittle fracture caused by PTS in accordance with 10 CFR 50.61." Please confirm that, like P-T limits and LTOP curves, PTS evaluation is also part of the Reactor Vessel Integrity Surveillance Program.

NMC Response to NRC RAI B2.1.16-1

Palisades reactor vessel surveillance capsule W-280 is scheduled for removal following completion of the 19th operating cycle, which is estimated to occur in September 2007. Results of subsequent testing are required by 10 CFR 50 Appendix H to be reported within one year from the date of removal, so they would be available as input for revision of P-T limits and LTOP curves by September 2008. It is anticipated that the revised P-T limits and LTOP curves would be submitted to the NRC in 2010 to satisfy the requirement for submittal three years prior to reaching the PTS screening criteria. Therefore the test results from capsule W-280 would be used in the development of the new curves. However, should plant operating cycle dates be such that the capsule is not removed in time to support this revision to the P-T limits and LTOP curves, this would not be a concern. The results from this capsule will follow those from capsule W-100 by only three cycles. Capsule W-280 would be unlikely to provide significant new information not already gleaned from W-100, and would be unlikely to alter the analysis for the new P-T limits and LTOP curves.

As stated on page 4-16 and B-120 of the LRA, monitoring the reactor for compliance with pressurized thermal shock requirements is an integral part of the reactor vessel surveillance program. In fact, the current Palisades P-T limits and LTOP curves are analyzed through 2014 when the reactor vessel is projected to reach the PTS screening criteria. Revisions to P-T limits and LTOP curves occur in conjunction with the PTS evaluation.

RAI B2.1.16-2

LRA AMP B2.1.16, "Reactor Vessel Integrity Surveillance Program," states that Enhancement 3 to this AMP will evaluate and revise the Palisades surveillance withdrawal and testing schedule of Final Safety Analysis Report (FSAR) Table 4-20 as necessary such that at least one capsule remains in the reactor vessel to be tested during the period of extended operation. GALL XI.M31, "Reactor Vessel Surveillance," Item 5 provides this guideline for applicants whose surveillance capsules' projected fluence (equivalent vessel fluence) at the end of 40 years is less than the 60-year fluence. Please confirm that the projected fluence at the end of 40 years for Palisades Capsules W-280, W-260, and W-80 is less than the projected 60-year reactor vessel fluence. Please also provide the projected fluence and EFPYs for Capsules W-280, W-260, and W-80 at the date of their scheduled withdrawals and identify the capsules that you intend to keep in the reactor vessel and to have them tested during the period of extended operation.

NMC Response to NRC RAI B2.1.16-2

Reactor vessel surveillance capsule W-280 is presently estimated to be removed in the fall of 2007. Capsules W-260 and W-80 are scheduled for removal during the extended operating period and are projected to have accumulated fluence of 2.57×10^{19} n/cm^2. The projected peak vessel fluence at 60 years (42.37 EFPY) is estimated to be 2.998×10^{19} n/cm^2.

Capsule	Capsule fluence at 40 years (24.63 EFPY)	Capsule fluence at removal	EFPY at removal	Refuel	Removal Date
W-280	2.33×10^{19} n/cm^2	2.33×10^{19} n/cm^2	20.99	19	Fall 2007
W-260	2.57×10^{19} n/cm^2	2.88×10^{19} n/cm^2	29.23	25	Fall 2016
W-80	2.57×10^{19} n/cm^2	3.06×10^{19} n/cm^2	31.96	27	Fall 2019