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Palisades Nuclear Plant
Operated by Nuclear Management Company, LLC

August 19, 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 Requests for Additional Information Relating to License Renewal dated July 20 and 21, 2005

In letters dated July 20, 2005 (ML052010471) and July 21, 2005 (ML052080025), the Nuclear Regulatory Commission (NRC) requested additional information regarding the License Renewal Application for the Palisades Nuclear Plant. This letter responds to those requests.

Enclosures 1 and 2 provide 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 no new commitments or changes to previous commitments.

I declare under penalty of perjury that the foregoing is true and correct. Executed on August 19, 2005.

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

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Enclosures (2)

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 (ML052010471)
Dated July 20, 2005

(2 Pages)

RAI 4.6-1

In Section 4.6.1, provide the missing information in the second paragraph of the summary.

NMC Response to NRC RAI 4.6-1

The missing text in the summary description on page 4-46 is shown underlined below:

The Palisades containment design relies on the liner only to maintain a leak-tight containment. There are no design conditions under which the liner plate is relied upon to assist the concrete in maintaining the integrity of the structure <u>even though the liner will</u>, at times, provide assistance in order to maintain deformation compatibility.

RAI 4.6-2

The Analysis subsection in Section 4.6.2 reproduces the corresponding paragraphs of the Analysis subsection in Section 4.6.1. Provide confirmation that this conforms with the corresponding section in FSAR 5.8.6.4.1.

NMC Response to NRC RAI 4.6-2

The Analysis subsection in Section 4.6-2 has been revised to focus on to containment penetrations, deleting the inadvertent reference to the containment liner. The revised text conforms with FSAR 5.8.6.4.1.

Revise the first two paragraphs to read as follows:

The allowable stress was conservatively based on the ASME B&PV Code, Section III, Article 4, 1965. Specifically, the following sections were adopted as guides in establishing allowable stress limits:

1. Paragraph N-412(m) - Thermal Stress, Subparagraph 2
2. Paragraph N-412(n) - Operational Cycle
3. Paragraph N-414.5, Table N-413, Figures N-414 and N-415(a) - Peak Stress Intensity
4. Paragraph N-415.1 - Vessels Not Requiring Analysis for Cyclic Operation

The number of design load cycles is relatively insignificant when compared to the allowable number of cycles on the fatigue curve in code Figure N-415 (a) for the peak design stress in the penetrations, and when compared to the allowable number of cycles for 3 times the S_m value of code Table N-421 at the operating temperature. The results of the analysis confirm that the design of the containment penetrations complies with the provisions of code paragraph N-415.1, and a fatigue analysis for design load cycles is not required.

ENCLOSURE 2

NMC Responses to NRC Requests for Additional Information (ML052080025)
Dated July 21, 2005

(15 Pages)

RAI 4.3-1

Provide an explanation for the difference in plant heat-ups (134) and plant cooldowns (119) in Table 4.3.1-1.

NMC Response to NRC RAI 4.3-1

The numerical difference between heatups and cooldowns was found in plant records, primarily between 1978 and 1982. It is assumed the difference is associated with either differences in interpretation of the logging criteria, or some cooldowns not being logged. The numbers of heatups and cooldowns have coincided as expected since the end of 1982. In retrospect, the cool-down number reported in the LRA should have been conservatively increased to 134.

Regardless of the specific values, however, the estimated numbers of heat-ups and cool-downs do not approach the design value of 500 cycles during the 60 year operating period. As indicated in Table 4.3.1-1, the estimated total number of heatup and cooldown cycles at 60 years (240) would still be less than half the design value.

RAI 4.3-2

In Table 4.3.1-1, provide the basis for not counting the safety valve operation cycles and the assumption that the design value of 200 cycles will never be approached.

NMC Response to NRC RAI 4.3-2

The Palisades Primary Coolant System (PCS) code safety valves have not operated during the plant life to date, and are not expected to operate in the future. The 2005 and 60-year values for safety valve operation indicated in Table 4.3.1-1 should be zero (0) rather than NC.

PCS pressure increases during startup and other periods of low-temperature, water-solid-pressurizer operation, will cause the power-operated relief valves (PORVs) to open prior to the safety valves, precluding the need for safety valve operation. Pressure increases during plant operation at power have been managed by the pressurizer (heaters, sprays) and the chemical and volume control system (letdown and makeup); these controls have been adequate to control primary system volume and pressure below the PCS code safety valve setpoints.

RAI 4.3-3

In Section 4.3.2, does the CUF value of 0.4516 represent the highest 60-year CUF for all structures and/or components in the reactor vessel, such as those listed in Table IV, A2 "Reactor Vessel (PWR)" of NUREG-1801, Volume 2, where cumulative fatigue damage/fatigue is the aging effect/mechanism, and which require further evaluation as TLAAs for the period of extended operation? If not, provide justification for not considering these components.

NMC Response to NRC RAI 4.3-3

The CUF value of 0.4516 represents the highest 60-year CUF for all structures and/or components in the reactor vessel, with the exception of the vessel studs addressed in LRA Section 4.3.3, the replaced CRDM housings and appurtenances discussed in Section 4.3.4, and the safety injection-shutdown cooling nozzles discussed in Section 4.3.8. The 60-year CUFs for the reactor vessel are:

Reactor Vessel Subcomponent	60-Year CUF
Reactor Vessel Inlet Nozzle	0.10365
Reactor Vessel Outlet Nozzle	0.451
Reactor Vessel (Lower Head to Shell Transition)	0.00364
Reactor Vessel Surveillance Tube Brackets	0.17361
Reactor Vessel Flow Baffle	0.01448
Reactor Vessel Instrument Nozzle Shroud Tube	0.0005
Reactor Vessel Instrumentation Bolts (on the instrument nozzle on the vessel head)	0.1104
Reactor Vessel Head CRDM Nozzle J-Weld	0.00782

Reactor vessel supports (pads under the supporting vessel nozzles) were evaluated for 25,000 heatup/cooldown cycles (design = 500), 6,000 design seismic cycles (design = 1,000) and 800 accident occurrences (design = 1). The evaluation found that no fatigue analysis was required.

RAI 4.3-4

In Sections 4.3.2 and 4.3.3, provide the basis for the statement: "The number of design basis transient events is not expected to approach the number assumed by the analysis during the extended licensed operating period."

NMC Response to NRC RAI 4.3-4

As a component of the Primary Coolant System, the reactor vessel design basis transients and the maximum estimated numbers of actual transients are summarized in Table 4.3.1-1. The numbers of transients experienced to date and the maximum total numbers of transients estimated for 60 years are well below those allowed by design.

The reactor vessel head closure stud design basis assumes 50 cycles for 60 years. 17 actual refueling cycles have been experienced through 2005. Projecting this rate into the future results in an estimated 32 cycles to be experienced in 60 years.

RAI 4.3-5

Provide the highest 60-year CUFs and disposition for all structures and/or components in the reactor vessel internals, such as those listed in Table IV B3 "Reactor Vessel Internals (PWR) - Combustion Engineering", Items B3.2-f, B3.4-d and B3.5-g, of NUREG-1801, Volume 2, where cumulative fatigue damage/fatigue is the aging effect/mechanism, and which require further evaluation as TLAAs for the period of extended operation.

NMC Response to NRC RAI 4.3-5

Because they are not part of the primary coolant pressure boundary, a fatigue analysis was not performed on the core support barrel or the upper guide structure. These removable assemblies contain the Palisades' equivalent to the referenced GALL items (i.e., control rod shroud, core shroud, core support plate, fuel alignment pins). The reactor vessel evaluation discussed in Section 4.3.2 did include core stabilizing lugs, core stop lugs and the flow baffle, because they are integral parts of the reactor vessel. As mentioned in the response to RAI 4.3-3 regarding Section 4.3.2, the 60-year CUFs for the reactor vessel are:

Reactor Vessel Subcomponent	60-year CUF
Reactor Vessel Inlet Nozzle	0.10365
Reactor Vessel Outlet Nozzle	0.451
Reactor Vessel (Lower Head to Shell Transition)	0.00364
Reactor Vessel Surveillance Tube Brackets	0.17361
Reactor Vessel Flow Baffle	0.01448
Reactor Vessel Instrument Nozzle Shroud Tube	0.0005
Reactor Vessel Instrumentation Bolts (on the instrument nozzle on the vessel head)	0.1104
Reactor Vessel Head CRDM Nozzle J-Weld	0.00782

RAI 4.3-6

In Section 4.3.5, does the CUF value of 0.9158 represent the highest 60-year CUF for all structures and/or components in the steam generators, such as those listed in Table IV, D1 "Steam Generators (Recirculating)", Items D1.1 and D1.2 of NUREG-1801, Volume 2, where cumulative fatigue damage/fatigue is the aging effect/mechanism, and which require further evaluation as TLAAs for the period of extended operation? If not, provide justification for not considering these components.

NMC Response to NRC RAI 4.3-6

As shown in the table below, the CUF value of 0.9158 represents the highest 60-year CUF for the structures and/or components in the steam generators. The 60-year CUFs for the steam generators are:

Steam Generator Subcomponent	60-year CUF
Steam Generator Primary Head Shell	0.12
Steam Generator Secondary Shell (in Tubesheet Area)	0.019
Steam Generator Tube-to-Tubesheet Weld	0.69
Steam Generator Secondary Shell	0.51
Steam Generator Primary Inlet Nozzle	0.052
Steam Generator Primary Outlet Nozzle	0.13
Steam Generator Main Steam Outlet Nozzle	0.0236
Steam Generator Auxiliary Feedwater Nozzle	0.2936
Steam Generator Main Feedwater Nozzle	0.9158
Steam Generator Secondary Manway	0.206
Steam Generator Secondary Handhole	0.16
Steam Generator Secondary Handhole Stud	0.50
Steam Generator Small Nozzles	0.028
Steam Generator Divider Plate	0.32
Steam Generator Miscellaneous Secondary Structures	0
Steam Generator Tubes and Tube Support	0
Steam Generator Upper Support Lugs	0.0033
Steam Generator Support Skirt	0.165

RAI 4.3-7

In Section 4.3.5, provide the date of Plant Heat-up Number 106 and the date when Plant Heat-up Number 306 is projected to occur.

NMC Response to NRC RAI 4.3-7

Heatup number 106 occurred on March 2, 1991. In Table 4.3.1-1, it is estimated that the primary system will be heated up approximately 240 times in 60 years. Assuming an average of 4 heatups per year, heatup number 306 would not occur until approximately year 2046. This is well past the expected expiration date of the renewed operating license, March 24, 2031.

RAI 4.3-8

In Section 4.3.6, does the CUF value of 0.7572 represent the highest 60-year CUF for all structures and/or components in the pressurizer, such as those listed in Table IV, C2.5 "Pressurizers" of NUREG-1801, Volume 2, where cumulative fatigue damage/fatigue is the aging effect/mechanism, and which require further evaluation as TLAAs for the period of extended operation? If not, provide justification for not considering these components.

NMC Response to NRC RAI 4.3-8

The CUF value of 0.7572 represents the highest 60-year CUF for all structures and/or components in the pressurizer, with the exception of the pressurizer surge nozzle (discussed in LRA Section 4.3.9), the pressurizer spray nozzle (discussed in Section 4.3.10), and the repaired pressurizer upper head temperature instrument nozzle for TE-101 (discussed in Section 4.7.2). The 60-year CUFs for the pressurizer are:

Pressurizer Subcomponent	60-Year CUF
Pressurized Bottom Head Support Skirt (lower end of tapered section)	0.36
Pressurizer Heater Sleeve-Head Junction	0.0226
Pressurizer Water Level Boundary	0.02094
Pressurizer Upper Level Nozzle	0.12499
Pressurizer Power-Operated Relief Valve Nozzle (Inside of Nozzle-Head Juncture)	0.7572
Replaced 316 SS Pressurizer Power-Operated Relief Valve Nozzle Safe End (Transition, Inside Wall)	0.084
Pressurizer Safety Valve Nozzles	0.122
Pressurizer Manway, Head and Studs	0.12429

RAI 4.3-9

Provide the highest 60-year CUFs and disposition for all structures and/or components in the reactor coolant system and connected lines, such as those listed in Table IV C2 "Reactor Coolant System and Connected Lines (PWR)", Items C2.2-a, C2.2-b and C2.2-c, "Connected Systems Piping and Fittings," of NUREG-1801, Volume 2, where cumulative fatigue damage/fatigue is the aging effect/mechanism, and which require further evaluation as TLAAs for the period of extended operation.

NMC Response to NRC RAI 4.3-9

Section 4.3.13 of the application summarizes the fatigue analysis results for the connected systems' piping and fittings. With the exception of the regenerative heat exchanger and the charging inlet nozzle, fatigue analysis has not been performed. As discussed on page 4-36, the allowable stress range is 1.0 S_A for 7,000 equivalent full-range thermal cycles or less. As long as the assumed number of the plant design basis event cycles is not exceeded, the secondary stress range reduction factors assumed for these B31.1 piping and components, and similar code designs, remain valid. The Fatigue Monitoring Program will ensure reanalysis or other appropriate corrective action in the unlikely event that a design basis cycle count limit is reached at any time during the extended licensed operating period.

The limiting component is the regenerative heat exchanger, discussed in section 4.3.7. The maximum calculated CUF for the 60 year period is 0.439 for the regenerative heat exchanger at the two most critical locations (tube sheet and tube sheet to shell junction).

RAI 4.3-10

In Section 4.3.8, provide the basis for the statement: "The number of each of the design basis events that affect the hot and cold legs is not expected to approach its design basis limit during the extended licensed operating period..."

NMC Response to NRC RAI 4.3-10

The numbers of design basis transients, estimated actual transients, and transients estimated to occur in 60 years, that affect the hot legs and cold legs, are summarized within lines 1 through 6 and 9 through 10 of LRA Table 4.3.1-1 Primary Coolant System Design Transients. The Primary Coolant System piping is limiting for Transients 5, 6 and 10. The number of transient cycles expected for 60 years remains less than the design basis cycles allowed. In those cases where Table 4.3.1-1 indicates NC as the estimated number, the event is not specifically counted because the allowed value will never be approached over the plant life.

RAI 4.3-11

In Section 4.3.9, provide clarification indicating whether Palisades has operated with continuous pressurizer spray operation since the start of plant operation. If not, indicate when continuous pressurizer spray operation was implemented. Indicate whether the CUFs for the surge line elbow, the Hot leg and Pressurizer Surge nozzles include cycling due to intermittent pressurizer spray operation.

NMC Response to NRC RAI 4.3-11

Pressurizer proportional heaters were originally intended to compensate for steady state losses. During the hot functional testing performed prior to plant operation, it was discovered that there was insufficient heat input from the proportional heaters to control pressure effectively. This resulted in cycling of the backup heaters, which caused primary coolant system pressure oscillations. Operation was revised to keep the backup heaters on with continuous spray, using the proportional heaters for control. The change was implemented prior to the beginning of commercial operation in 1971.

Although the Palisades pressurizer spray is operated continuously, the CUFs reported for the surge line elbow (0.937), the hot leg to surge line nozzle (0.3818) and the pressurizer surge nozzle (0.9611) were determined by fatigue evaluation for a typical Combustion Engineering plant with intermittent pressurizer spray.

RAI 4.3-12

In Section 4.3.9, provide the basis for the statement: "The number of transient events which might be expected to initiate these thermal stratification events will not exceed their design basis limits for the extended licensed operating period."

NMC Response to NRC RAI 4.3-12

The thermal stratification events are those discussed in the subsection entitled "Design Basis Thermal Transients and Expected Thermal Transients." The transients addressed in the CEOG report included limitations for:

Surge line ΔT	Cycles allowed
90 °F	87,710
150 °F	500
200 °F	400
250 °F	375
320 °F	75

Due to the use of continuous spray, the ΔT of the metal at Palisades will not exceed 210 °F, well below the 320 °F assumed in the CEOG report. Only 47 events are recorded with a measured ΔT above 200 °F. As such, the limiting transient is the normal heat-up and cool-down transients which have been estimated to reach 240 cycles, well below the cycle limitations listed above that can be reached with a maximum ΔT of 210 °F.

RAI 4.3-13

In Section 4.3.10, provide the basis for the 60-year projected CUF of 0.517 for the pressurizer spray nozzle, when the first 20-year CUF is estimated as 0.353.

NMC Response to NRC RAI 4.3-13

The pressurizer spray nozzle fatigue evaluation, performed in 1991, estimated the cumulative usage factor to be 0.353 based on the number of transient cycles observed to date. That same evaluation estimated that the frequency of spray over the subsequent 20-year period (years 21 through 40) would increase the CUF by 0.082 for a total of 0.435 for the 40-year operating period. Assuming operation during years 41 through 60 will be the same as during years 21 through 40, the CUF again increases by 0.082 to the reported value of 0.517.

RAI 4.3-14

In Section 4.3.10, provide the basis for the statement: "The projected number of cycles of the design basis events does not exceed the design basis limit during the extended licensed operating period."

NMC Response to NRC RAI 4.3-14

The design basis events that affect the pressurizer spray piping and nozzles are the 200°F per hour cool-downs and the high-differential-temperature spray events.

The plant heatup and cooldown transients, listed as Transient No. 1 in Table 4.3.1-1, include pressurizer cool-down at 200°F per hour. 500 cool-down events are assumed in the design. The estimate of 240 plant cool-downs projected for 60 years is substantially less than the 500 cooldowns assumed in the design.

The high-differential-temperature spray events are listed below:

Pressurizer Spray Specified ΔT	Pressurizer Spray Counted Range ΔT	Cycles allowed	Counted to January 9, 2005	Most events expected in 60 years
303°F	201°F-300°F	300	26	50
403°F	301°F-400°F	150	17	32
503°F	401°F-500°F	10	3	7
613°F	501°F-600°F	10	1	3

Again, the numbers of each transient projected at 60 years are substantially less then the values assumed in the design.

RAI 4.3-15

Section 4.3.14 indicates that Palisades has no shutdown cooling line inlet transition, and that the safety injection and shutdown cooling functions share a common nozzle. As an alternate location to the shutdown cooling line inlet transition, provide the highest CUF at this location which includes the effect of the reactor coolant system environment, or select an alternative high CUF location equivalent to the shutdown cooling line inlet transition.

NMC Response to NRC RAI 4.3-15

Preliminary analysis results indicate that the limiting location in this area is at the end of the cladding near the safe end on the safety injection nozzle. This is the common nozzle that supports both safety injection and shutdown cooling. The fatigue usage factor at this location is 0.0308. After applying the environmental factor of 15.35 for stainless steel, the environmentally corrected usage factor is 0.472.

The analysis which supports these values is in the process of being finalized. If these values change in the final, approved analysis, an updated response will be provided.