July 7, 2006

10 CFR 50.91(a)(5)

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-0001

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

Response to Requests for Additional Information Regarding License Amendment Request: Removal of TSP from Palisades Containment

By letter dated March 20, 2006, pursuant to 10 CFR 50.91(a)(5), Nuclear Management Company, LLC (NMC) requested Nuclear Regulatory Commission (NRC) review and approval of a proposed license amendment for the Palisades Nuclear Plant (PNP). NMC is proposing to remove tri-sodium phosphate (TSP) from the containment building at PNP as an interim measure until the long term resolution of GSI-191 is implemented.

By letter dated June 22, 2006, the NRC sent a request for additional information (RAI) regarding the proposed license amendment request. Enclosure 1 provides responses to the RAIs for PNP.

A copy of this RAI response has been provided to the designated representative of the State of Michigan.

Summary of Commitments

This letter contains one new commitment and one revision to an existing commitment.

Commitment made by letter dated March 20, 2006:

2. NMC will inject sodium hydroxide as an alternate buffer within seven days post-LOCA with recirculation at PNP.

Revised commitment:

2. NMC will inject sodium hydroxide as an alternate buffer post-LOCA with recirculation within 20 hours following a loss of the fuel cladding barrier, as defined in the Emergency Action Levels, or within seven days, whichever occurs first, at PNP.

A001

New commitment:

> NMC will implement an alternate buffer program to achieve a pH of 7.0 – 8.0 post-LOCA with recirculation, during the 2007 fall refueling outage at PNP.

I declare under penalty of perjury that the foregoing is true and correct. Executed on July 7, 2006.

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

Enclosures (1)

cc: Administrator, Region III, USNRC
Project Manager, Palisades, USNRC
Resident Inspector, Palisades, USNRC

ENCLOSURE 1
RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION
PALISADES NUCLEAR PLANT

Part 1

NRC Request

1. *In Enclosure 1 of Nuclear Management Company's (NMC's) March 20, 2006, license amendment request regarding removal of tri-sodium phosphate from the Palisades' containment, NMC stated that the current licensing basis at PNP includes atmospheric dispersion factors (χ/Q values) that are based upon site-specific, wind tunnel modeling. When, and by what mechanism, did this methodology become part of the Palisades' licensing basis? Did the Nuclear Regulatory Commission approve this methodology for Palisades?*

NMC Response

1. The analysis of the maximum hypothetical accident, which incorporated the methodology for control room atmospheric dispersion factors based on site-specific wind tunnel testing, was submitted to the NRC by letter dated January 24, 1996. As this letter indicated, the analysis was to be incorporated into the Palisades FSAR. By letter dated August 11, 2000, the NRC was notified that the methodology was incorporated into the FSAR. Incorporation in Revision 19 was accomplished via the FSAR change request process.

   The site-specific wind tunnel data based on χ/Q methodology was not approved by the NRC. The methodology was withdrawn from review consideration by letter dated August 11, 2000, which cited the staff's concern that a sufficient body of wind tunnel test data was not available to permit a comprehensive review of the methods for translating wind tunnel test data to χ/Q values. In this letter, PNP committed to evaluate the control room habitability analyses in consideration of RG 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," July 2000, NEI 99-03, "Control Room Habitability Assessment Guidance," March 2003, and NUREG/CR-6331, "Atmospheric Relative Concentrations in Building Wakes," May 1997, and submit the analyses for review as appropriate.

NRC Request

2. *NMC further stated that Palisades has committed to submit a methodology change to a full scope implementation of the alternative source term (AST) methodology in July 2006, which will establish a conforming radiological design basis for control room habitability and offsite doses at Palisades. Implementation of the full scope AST methodology is currently expected to occur upon start-up from the fall 2007 refueling outage. Does this commitment specifically address if there will be a change in the methodology to be used to calculate χ/Q values?*

NMC Response

2. No. The commitment cited in Enclosure 1, to the March 20, 2006 letter, refers to the first new commitment in the letter to the staff dated July 7, 2005, regarding control room habitability, and does not specifically refer to a change in χ/Q methodology for calculating on site χ/Q. However, Enclosure 1 of the July 7, 2005 letter notes that on-site χ/Q generated with ARCON96 is a conforming methodology, and that χ/Q based on site-specific wind tunnel testing is not. PNP intends to submit χ/Q values based on the ARCON96 and PAVAN analytical methodologies that do not rely on site-specific wind tunnel testing data, as part of the full scope alternate source term methodology submittal, which is due to be submitted by July 31, 2006.

NRC Request

3. *Page 10 of Enclosure 4 states that the exclusion area boundary and low population zone χ/Q values used in the dose assessment for this license amendment request are those in the Palisades Final Safety Analysis Report. Were these χ/Q values previously reviewed and approved by the U.S. Nuclear Regulatory Commission staff as part of a prior license amendment request? If so, please provide an appropriate reference citation (e.g., safety evaluation report).*

NMC Response

3. No. The offsite χ/Q values have not been previously reviewed and approved by the NRC staff. PNP's original short term offsite χ/Q values were based on twelve months of meteorological (MET) data accumulated over the period of December 1, 1977, to November 30, 1978. This data was accepted by the NRC and incorporated into the FSAR. Revision 18 of the FSAR incorporated a requirement for review of the short term χ/Q if the annual average parameters varied by more than ±25% from the 1977/78 values.

   In 1996, MET data varied sufficiently from the FSAR values and a review was required. The review was performed in 1998, and combined the 1977/78 data with 14 years of data to assess any variations. It was concluded that the χ/Q based on 1977/78 data was a reasonable and adequate representation of atmospheric dispersion characteristics at the PNP site. No regulatory requirement to revise the short term offsite χ/Q was identified. However, since χ/Q based on 15 years worth of MET data is statistically more significant than χ/Q based on only one year of data, the FSAR was revised to include the new χ/Q values to ensure future dose analyses incorporated the more statistically significant values. The 10 CFR 50.59 process was used to revise the FSAR χ/Q. Since the revised χ/Q were bounding (i.e., higher) than the existing FSAR χ/Q based on the 1977/78 data, no prior NRC approval was required. The revised FSAR χ/Q values were used in the license amendment request.

PNP intends to submit χ/Q values based on the ARCON96 and PAVAN analytical methodologies that do not rely on site-specific wind tunnel testing data, as part of the full scope alternate source term methodology submittal, which is due to be submitted by July 31, 2006. The methodology for calculating offsite χ/Q is only changed in the sense that updated meteorological data is planned to be used along with a version of the PAVAN code that contains corrections to errors identified in the USNRC version.

NRC Request

4. *The loss of coolant accident radiological consequences analysis takes credit for wall deposition of elemental iodine as discussed in NUREG-0800, "Standard Review Plan" (SRP), Section 6.5.2, "Containment Spray as a Fission Product Cleanup System," Revision 2. The elemental iodine wall deposition removal coefficient calculated by the SRP 6.5.2 method is based on a mass transfer coefficient for the conditions in the containment when sprays are operating. Changes to containment spray operation do not appear to be proposed in this amendment request. Are changes being proposed to post-accident containment spray operation?*

NMC Response

4. No. Changes to spray operation are not being proposed in the TSP Removal LAR. It should be noted that in response to GSI-191 concerns, candidate operator action of early termination of containment sprays prior to recirculation have been implemented. However, the conditions under which sprays may be terminated require that sprays are no longer needed for iodine removal from containment atmosphere. Therefore, there are no changes to spray operation that impact wall deposition.

Part 2

NRC Request

1. *In the event of a loss-of-coolant accident (LOCA) with recirculation, sodium hydroxide (NaOH) will be injected via the containment spray system within 7 days to control the containment pool pH. Provide an overview of the injection method, including NaOH spray duration, pH of the spray, and the range of containment pool pH once all NaOH has been injected. Discuss any assumptions.*

NMC Response

1. PNP intends to use a portable injection assembly to transfer the required amount of NaOH into containment upon demand. Figure 1 below provides an overview of the injection method. The portable assembly transfers the NaOH from the tank of the radwaste chemical addition system to the injection port on an instrument flange of the containment spray system. The NaOH can be injected into either one of the two redundant containment spray headers. Two portable injection pump skids are planned to be available at the plant site. The pump is planned to be powered from a safety related source and is to deliver a flow of 2.5 gallons per minute.

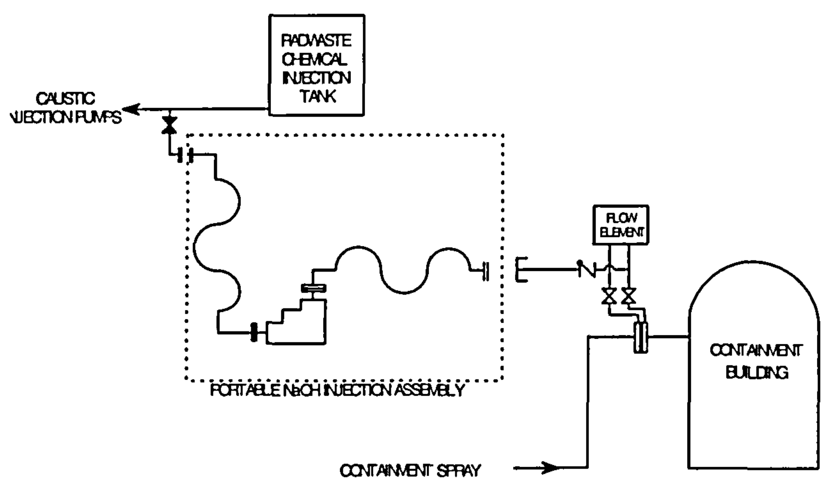

FIGURE 1

Under the design basis loss of coolant accident scenario, a nominal 380 gallons of 25% NaOH solution is to be injected into containment via the spray flow. The duration of the NaOH injection operation is estimated to be two and one-half hours. The design of the NaOH injection system is to raise the containment sump pH to between 7 and 8 as measured at 25ºC, based on an initial spray flow pH of 4.5.

NRC Request

2. Have any tests been conducted to evaluate potential interactions (i.e., chemical effects) between containment materials and a post-LOCA containment pool that does not contain a chemical to buffer pH? If so, please provide the results. If not, please discuss if there are any plans to investigate potential chemical effects in containments that do not have a buffering agent.

NMC Response

2. NMC has not conducted tests to evaluate potential interactions between containment materials and a post-LOCA containment pool that does not contain a chemical to buffer pH at PNP. The evaluations performed to support the TSP removal are based on the data and testing results from existing plant analyses, manufacturer specifications and technical literature in the industry. The evaluations address the chemical effects including hydrogen generation, corrosion of structural material, and the impact on the electrical equipment qualification program. There are no significant chemical effects identified from the evaluation. The current design with the use of TSP assumes a low pH spray occurring for a period of time during a loss of coolant accident until reaching the recirculation mode. Testing that has been performed with no buffer and low pH conditions has shown no significant chemical effects, as discussed in Westinghouse WCAP-16530-NP, "Evaluation of Post-Accident Chemical Effects in Containment Sump Fluids to Support GSI-191," dated February 2006. NMC continues to follow industry chemical effects testing efforts.

NRC Request

3. In relation to calculating release of the elemental iodine from the Safety Injection Refueling Water Tank (SIRWT), on page 5 of Enclosure 4 NMC said "... appropriately justified airborne fractions for SIRWT leakage addresses the impact of TSP removal on iodine airborne fraction of leak sump water." Page 6 of Enclosure 4 provides the value of SIRWT iodine-volatile fraction of 0.453, and the corresponding maximum fraction of airborne elemental iodine of 0.0456. These values were determined in Nuclear Management Company's (NMC's) analysis included in Attachment 1 to Enclosure 4(?). Since the analysis is difficult to follow without having more details, please provide the following information:

- *Is the current analysis performed using one iodine concentration (e.g., maximum) in the SIRWT, or does it consider time-dependant values for different amounts of iodine leaked from the containment sump?*

NMC Response

- The current licensing basis methodology that was used in the Enclosure 4 analysis (EA-EC-976-01) considers time-dependent values for different amounts of iodine leaked from the containment sump. The methodology models the time-dependent iodine concentration in the SIRWT air volume due to iodine leaked from the containment sump as follows:

$$C_{air,i}(t) = \frac{A_{I_2,i}(t)}{V_{air}(t) + PF_{SRW} V_{liq}(t)}$$

Where:

$C_{air,i}(t)$	:	concentration of iodine isotope "i" in SIRWT air volume at time "t", Ci/ft^3
$A_{I_2,i}(t)$	:	total elemental iodine activity of iodine isotope "i" in SIRWT total (air plus water) volume at time "t", Ci
$V_{air}(t)$	:	SIRWT air volume at time "t", ft^3
PF_{SRW}	:	elemental iodine partition coefficient in SIRWT
$V_{liq}(t)$	:	SIRWT liquid volume at time "t", ft^3

As per Equation (25) of EA-PAH-91-05.

NRC Request

- *Justify using a concentration of 6.72E-05 g-atom/L of iodine in the SIRWT liquid. Since this value comes from Reference 11 in Attachment 1, why does it apply to the current analysis?*

NMC Response

- As indicated in Input 4.15, of EA-EC-976-01, the current licensing basis core inventories from NEDO-24782, "BWR Owners Group NUREG-0578 Implementation: Analyses and Positions for Plant Specific Submittals," dated August 1980, are used. As indicated in Assumption 6.5, of EA-EC-976-01, the current licensing basis release fraction of 50% of the core iodine inventory is assumed to be released to the containment sump. For the purposes of total iodine concentration determination, this includes both radioactive and non-radioactive iodine; NEDO-24782 gives the 50% iodine inventory as 0.03062 g-atom/MW_{th} (Pages B-2 and B-27). As indicated in Input 4.2, of EA-EC-976-01, the current licensing basis thermal power, including calorimetric uncertainty, is 2580.6 MW_{th}, resulting in 79.018 g-atom of iodine released to the containment sump. As indicated in Input 4.7, of EA-EC-976-01, the current licensing basis minimum sump volume is 41,498.2 ft^3 or 1,175,098 L, resulting in an iodine concentration of 6.72x10-5 g-atom/L. Note that a minimum sump volume results in a maximum iodine concentration, and a maximum iodine concentration results in a higher volatile fraction. Since the SIRWT initial water volume has an iodine concentration of zero, the sump iodine concentration bounds (i.e., is always higher than) the SIRWT concentration.

The calculation above mirrors the licensing basis calculation in EA-TAM-95-05, which is Reference 11 of EA-EC-976-01. The iodine core inventory, containment sump release fraction, thermal power level, and sump water volume are not impacted by the removal of TSP from containment. Therefore, since all inputs for the iodine concentration calculation remain valid under TSP removal, application of the calculated iodine concentration to the current analysis (EA-EC-976-01) is valid.

NRC Request

- *In your analysis, the pH used to determine the fraction of volatile iodine is 4.5, and it does not change with the in-leaking containment sump water. Justify this assumption.*

NMC Response

- A ph of 4.5 is lower than the expected pH of the sump water and the initial SIRWT water. Therefore, an assumption of a constant SIRWT pH of 4.5 bounds the expected transient SIRWT pH considering the in-leaking containment sump water.

  Expected pH of sump water: In an actual large break LOCA event with significant fuel failure, fission products (primarily cesium in the form of cesium hydroxide, cesium borate, and cesium iodide) and core-concrete interaction products are likely to control pH to pH>7, for time periods on the order of 24 hours in the absence of pH control additives such as TSP (NUREG/CR-5950, "Iodine Evolution and pH Control, Oak Ridge National Laboratory," December 1992). Additionally, regardless of the initial pH or the presence of TSP, the pH of the sump water is expected to rise to about 7, primarily because Cal–Sil contains sodium silicate as an impurity (ANL Report, Chemical Effects/Head–Loss Testing, Quick Look Report, Tests ICET-3-4 to 11, Argonne National Laboratory, ADAMS Ascension Number ML060190713, January 20, 2006). The sodium silicate is very soluble and as it dissolves, the dissolved sodium causes the pH of the initial boric acid/LiOH solution to increase. PNP specific testing indicates that post-accident sump water pH would be increased by the dissolved Cal-Sil (NWT Report 729, Part 1, Palisades Calcium Silicate Insulation Behavior, Laboratory Evaluations, October 2005). The assumption of a constant SIRWT pH of 4.5 conservatively ignores any beneficial increase in pH due to the higher pH of the backleakage.

  Initial pH of SIRWT water: The initial SIRWT normal pH value range is 4.5-7.0, from the chemistry operating procedure for the engineered safeguards system. Note that a best estimate value for the minimum initial SIRWT pH is higher than 4.5, based on representative recent chemistry trend data.

  The pH value of 4.5 was chosen to conservatively bound post-LOCA SIRWT pH. Elemental iodine is not retained until the pool acidity begins to decline, i.e., when the pH is greater that 4.5.

NRC Request

- *In the analysis provided in Attachment 1 to Enclosure 4, please clarify the meaning of the numbers under the heading "SIRWT Iodine Volatile Fraction."*

NMC Response

- In Attachment 1 of EA-EC-976-01, the numbers under the heading "SIWRT Iodine Volatile Fraction" are identified by column headings, which have the following meanings.

$[I]_{aq}$: molar concentration of aqueous iodine in the sump (note that this bounds the concentration in the SIRWT since the SIRWT initial water volume has an iodine concentration of zero)

pH : pH of the water in the sump and SIRWT

$$Term\,1 = \frac{[I]_{aq}}{2}$$

$$Term\,2 = \frac{d + e10^{-pH}}{8*10^{-2pH}}$$

$$Term\,3 = \frac{1}{8*10^{-pH}}$$

$$Term\,4 = \frac{\left(d + e10^{-pH}\right)^2}{10^{-2pH}}$$

$$Term\,5 = 8[I]_{aq} * \left(d + e10^{-pH}\right)$$

Aqueous iodine will exist in pools/sumps in both I^- and I_2 species. Equation 12 in Section 3 of NUREG/CR-5950, "Iodine Evolution and pH Control," Oak Ridge National Laboratory, December 1992, derives the following relationship between the dissolved iodine ions $[I^-]$ and the elemental aqueous iodine $[I_2]$ concentration.

$$[I_2]_{aq} = \frac{[H^+]^2[I^-]^2_{aq}}{d + e[H^+]}$$

Where:

$[I_2]_{aq}$	=	concentration of elemental iodine (g-moles/liter)
d	=	6.05E-14 ± 1.83E-14
e	=	1.47E-09
$[H^+]$	=	concentration of H^+ ion (g-moles/liter)
$[I^-]$	=	concentration of ionic iodine (g-moles/liter)

In order to maximize the amount of I_2 in solution (and consequently the amount in the gas phase), the conservative value of the "*d*" parameter is used, i.e., the lower of the specified range or 4.22E-14. Although these values are based on experimental data at 25 °C, Appendix C of NUREG/CR-5950 indicates that this model conservatively over-predicts the conversion to I_2 at higher temperatures.

The total iodine concentration in the pool/sump is given by the following expression per Section 3.2 of NUREG/CR-5950 and includes the non-radioactive isotopes of iodine (e.g., I^{127}).

$$[I]_{aq} = 2*[I_2]_{aq} + [I^-]_{aq}$$

Where:

$[I]_{aq}$ = total iodine concentration (g-atoms/liter)

Eliminating the variable for the ionic iodine parameter $[I^-]$ and considering that $[H^+] = 10^{-pH}$, the following equation relates the aqueous I_2 concentration to the pool/sump pH and the total iodine concentration:

$$[I_2]_{aq} = \frac{[I]_{aq}}{2} + \frac{d + e10^{-pH}}{8*10^{-2pH}} - \frac{1}{8*10^{-pH}}\sqrt{\frac{(d + e10^{-pH})^2}{10^{-2pH}} + 8[I]_{aq}*(d + e10^{-pH})}$$

Note that applying the nominal value of the "*d*" parameter of 6.05E-14, the fraction of iodine in the I_2 species (i.e., $2[I_2]/[I]$) can be determined as a function of pH for various total iodine concentrations, as is done in Figure 3.1 of NUREG/CR-5950.

The equation developed above is used to calculate the aqueous I_2 concentration, and can be restated as indicated below.

$$[I_2]_{aq} = Term\,1 + Term\,2 - Term\,3\sqrt{Term\,4 + Term\,5}$$

Part 3

NRC Request

NMC's letter of March 20, 2006, contained the following three new commitments:

1. *NMC will implement a potassium iodide (KI) program for control room personnel at PNP upon approval of the license amendment request. The KI program will be implemented per the guidance provided in Nuclear Energy Institute 99-03, "Control Room Habitability Assessment Guidance"...*

2. *NMC will inject sodium hydroxide as an alternate buffer within seven days post-LOCA with recirculation at PNP.*

3. *NMC will submit a license amendment request to implement an alternate buffer program after the Westinghouse Owners Group (WOG) Alternate Buffer Project is concluded.*

The following additional information is needed to complete our review:

1. *This item pertains to the timeliness of the proposed interim measure for addition of NaOH to control pH within 7 days following a LOCA.*

   The analysis of risk in the application appears flawed in that it solely addresses an unquantified avoidance of an increase in core damage frequency caused by postulated chemical effects blockage of the recirculation flow path. Recent NRC testing indicates a potential for incremental increase in blockage due to potential calcium silicate insulation and TSP chemical effects (above a debris contribution alone, which is considered the more likely cause of blockage). However, removal of the pH control, following both postulated and hypothetical accidents, increases the consequences of the entire set of such accidents when considering iodine reevolution from the containment pool. For design basis accidents, NMC assessed that, considering this increase coincident with certain revised assumptions, the calculated doses remained within regulatory limits. Although this aspect is being reviewed by the staff, there remains, nonetheless, an unquantified increase in consequence associated with removal of any buffer which could offset any decreased risk achieved by removing the chemical effect blockage. Without a quantifiable evacuation of these offsetting risks, NMC's statement that there is a net increase in plant safety appears unfounded.

   Regarding the impact of TSP removal on radiological consequences, NMC stated that in the event of a large-break LOCA with significant fuel damage, fission products released will likely control pH. The staff notes that the purpose of the proposed amendment is to avoid such fuel damage following an accident. Nonetheless, it would seem prudent to have pH control available in the event of significant core damage, versus relying on fission products to control pH. Also, NMC assumes that an additional source of conservatism is the dissolution of calcium silicate insulation following a LOCA. However there is no assurance that any significant quantity of calcium silicate insulation will be dislodged and *dissolved following a LOCA, nor will any be necessarily present for other hypothetical accidents, therefore considering its effects appears to be non-conservative.*

On this basis, the staff considers it prudent to have a readily available method to control pH (in hours upon detection of significant core damage, versus the 7 days proposed in the amendment). Additionally, on this basis, the staff should review the method and system aspects as a part of the amendment review and approval, versus the current proposal to accomplish this task in the 60-days following approval of the amendment.

In response to this request, either address the points made by the staff in the above paragraphs, or describe how you will achieve buffer addition in a more timely manner than the proposed 7-days to mitigate iodine reevolution from the sump pool.

NMC Response

1. NMC will inject sodium hydroxide as an alternate buffer post-LOCA with recirculation within 20 hours following a loss of the fuel cladding barrier, as defined in the Emergency Action Levels, or within seven days, whichever occurs first, at PNP. Primary coolant activity level, core exit thermocouple readings, and containment radiation monitoring are specifically used to determine when loss of fuel cladding barrier has occurred. Values for these parameters are established that would indicate potentially significant core damage. In the event that there is no indication of fuel cladding barrier failure, the NaOH will be injected into containment within 7 days post-LOCA with recirculation to protect the SSCs from corrosion under acidic environment.

NRC Request

2. *Provide the methods and procedural plans to inject NaOH into the containment spray system. Include aspects of the operation of containment spray system to achieve NaOH addition to the sump pool.*

NMC Response

2. Upon demand, a nominal amount of 380 gallons of 25% concentration NaOH is to be injected from the radwaste chemical injection tank into the containment spray flow pathway using a portable injection system. The injection system can take suction from either one of the two piping headers at the tank outlet and be pumped into one of the two containment spray headers. Manual connection of the NaOH injection system by operations staff would be required. Two NaOH injection pump skids are planned to be available for redundancy. The portable NaOH injection pump skid, powered from a safety related bus, has the capacity to deliver a 2.5 gpm NaOH into the containment spray flow pathway. The hose assemblies are made of sections of one-inch size flexible hoses, burst rated and pressure tested by the manufacturer prior to shipping. The pump skids, the connecting hose assemblies, electrical cord, and the NaOH are to be pre-staged

in a designated storage area. PNP intends to include the operation of the NaOH injection system in the Emergency Operating Procedure (EOP) associated with pre and post RAS actions.

PNP has three containment spray pumps to support the two redundant trains of the emergency core cooling system. During the post-RAS operation, the spray pumps provide both the motive force of the containment spray flow, and the subcooled cooling water to the high pressure safety injection system. At least one of the spray pumps and one spray header will be in service to support the recirculation operation. The NaOH injection operation would temporarily remove service from the containment spray flow meter on one of the two spray headers. However, the spray flow is not interrupted by the NaOH injection operation.

NRC Request

3. Describe the specifications of the buffer to be used, how it will be stored on site, and its availability for use.

NMC Response

3. A minimum of 380 gallons of 25% concentration NaOH will be pre-staged in a dedicated storage area. The NaOH will be handled as hazardous material in accordance with the applicable PNP procedures.

NRC Request

4. Provide calculations of the amount of NaOH to be added to complete the neutralization of the borated water added to the sump pool.

NMC Response

4. The amount of NaOH to be added to complete the neutralization of the borated water added to the sump pool is provided in calculation EA-EC976-09, "Determine Post-LOCA Sodium Hydroxide Amount." This calculation is provided as Attachment 1.

NRC Request

5. Describe testing and maintenance planned for the NaOH-injection capability.

NMC Response

5. PNP intends to test the portable NaOH injection system prior to making it available for emergency service. The testing plans include performance testing the NaOH injection pump skids, hydrostatic testing, NaOH injection pump relief valve set point verification, and pressure testing the connecting hose assemblies. Both the NaOH injection pump skids and the associated hose assemblies are pressure rated and therefore, the tests would be conducted at a pressure level in excess of the design requirements. Since this portable NaOH injection pump

system is a compensatory measure that will not be used following the 2007 refueling outage, preventive maintenance and surveillance testing are not planned.

NRC Request

6. *NMC's application stated that it would submit a license amendment request to implement an alternate buffer after the Westinghouse Owners Group Alternate Buffer Project is concluded. This project may not provide definitive guidance for selection of an alternate buffer. Describe your plans and schedule to implement an alternate buffer.*

NMC Response

6. The Westinghouse Owners Group (WOG) has identified several potential alternate buffer agents for investigation. Narrowing the recommendations of these potential buffer agents is in progress and a report of the findings scheduled to be completed shortly. NMC understands that the WOG will recommend sodium tetraborate and sodium metaborate as alternatives to the TSP buffering agent. NMC intends to review the WOG recommendations and select the appropriate buffer for PNP.

   PNP intends to incorporate the chemical effects of the selected buffer into the PCI passive strainer design and testing program, to verify that the strainer fully loaded pressure drop is acceptable. PCI is the passive strainer vendor for PNP. The testing plans include the chemical precipitation expected from alternate buffer interaction with containment materials.

   NMC intends to include the bypass debris from the strainer vendor program in a core analysis to ensure long term core cooling is maintained. The impact on core cooling due to chemical (including boron) precipitation and bypass debris accumulation is planned to be analyzed. NMC will implement an alternate buffer program to achieve a pH of 7.0 – 8.0 post-LOCA with recirculation, during the 2007 fall refueling outage at PNP.