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Palisades Nuclear Plant
Operated by Nuclear Management Company, LLC

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Updated Response to Information Notice 2005-26, "Results of Chemical Effects Head Loss Tests in a Simulated PWR Sump Pool Environment," for Palisades Nuclear Plant

On September 16, 2005, the Nuclear Regulatory Commission (NRC) issued Information Notice (IN) 2005-26 to all holders of operating licenses for pressurized water reactors (PWRs). The NRC requested that recipients review the information contained in the notice for applicability to their facilities and consider taking actions, as appropriate, to avoid similar issues.

By letter dated November 30, 2005, NMC submitted a response to the IN for Palisades Nuclear Plant (PNP). On December 13, 2005, a teleconference was held between NMC and NRC staff to discuss the PNP IN response. A subsequent phone call held on December 14, 2005, with senior NMC management and NRC staff resulted in a commitment for Palisades to update the IN 2005-26 response by December 23, 2005. Enclosure 1 provides the update to the IN 2005-26 response.

As a result of this updated response, NMC requests a meeting with NRC staff to discuss the PNP Tri-Sodium Phosphate removal License Amendment Request strategy. NMC is prepared to meet with the NRC and desires to conduct this meeting prior to the end of January 2006.

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Summary of Commitments

This letter contains no new commitments and no revisions to existing commitments.

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Enclosure (1)

CC Administrator, Region III, USNRC
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ENCLOSURE 1
UPDATED RESPONSE TO INFORMATION NOTICE 2005-26
PALISADES NUCLEAR PLANT

Summary

The following actions have been taken for Palisades Nuclear Plant (PNP):

- The Nuclear Regulatory Commission (NRC) Information Notice (IN) 2005-26, "Results of Chemical Effects Head Loss Tests in a Simulated [Pressurized Water Reactor] PWR Sump Pool Environment," test results have been reviewed and several significant differences between PNP conditions and the Argonne National Laboratory (ANL) test conditions have been identified. These differences suggest that calcium phosphate precipitants would be significantly less than observed during the ANL tests.

- A preliminary evaluation has been completed supporting removal of tri-sodium phosphate (TSP) from the PNP containment through a License Amendment Request (LAR).

- The compensatory measures taken in response to NRC Bulletin (BL) 2003-01, "Potential Impact of Debris Blockage on Emergency Sump Recirculation at Pressurized-Water Reactors," have been reviewed and found similarly effective for chemical effects debris types. In addition, the relationship between compensatory measures and additional PNP safety features mitigating sump blockage effects has been developed.

- The potential for significant head loss due to the creation of calcium phosphate precipitation, as described in IN 2005-26, has been entered into the PNP Operating Experience Program.

The following paragraphs provide a detailed discussion concerning these actions.

Background

IN 2005-26 provided the initial results of the NRC sponsored head loss testing being performed at the ANL. The information is relevant to plants containing phosphate and calcium sources that may dissolve within the post loss-of-coolant accident (LOCA) containment pool with sufficient concentrations to form calcium phosphate precipitate. The test results indicate that substantial head loss can occur if sufficient calcium phosphate is produced in a sump pool and transported to a preexisting fiber bed on the sump screen. Consequently, the emergency core cooling system (ECCS) flow and containment spray system (CSS) flow could be reduced by the increased head loss across the sump screen. Although significant increases in head loss were observed in the ANL testing, the NRC noted that these head loss results were obtained in a recirculating test loop not intended to be prototypical of PWR plant containments.

PNP uses TSP as a buffering agent to increase the pH of the initially acidic post-LOCA containment water to a more neutral pH. PNP calcium sources include the containment concrete and two forms of calcium silicate (CalSil): 1) pipe insulation and 2) Marinite® fiber board.

Information Notice 2005-26 Applicability Review

By letter dated November 30, 2005, Nuclear Management Company (NMC) submitted a response to the IN for PNP. On December 13, 2005, a teleconference was held between NMC and NRC staff to discuss the PNP IN response. A subsequent phone call held on December 14, 2005, with senior NMC management and NRC staff resulted in a commitment for NMC to update the PNP IN 2005-26 response by December 23, 2005. This Enclosure provides the update to the PNP IN 2005-26 response.

NMC has reviewed the compensatory measures taken for PNP in response to BL 2003-01. Those compensatory measures address preventing or delaying the onset of sump recirculation flow or minimizing potential debris effects during sump recirculation flow. The measures address any debris type regardless of the cause. For this reason, NMC concluded the compensatory measures identified in NMC's response to BL 2003-01 are similarly effective for chemical effects debris types.

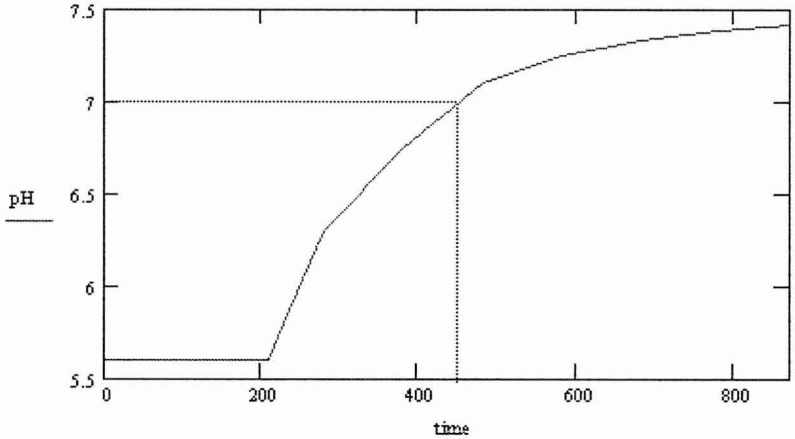
In addition, NMC reviewed the test conditions stated in the IN and the expected conditions for PNP and concluded that they are substantially different. The specific discussions and conclusions contained in this submittal are, in part, based on engineering judgment. Please note that NMC has been working on, and continues to work on, developing a long term resolution to the chemical effects issue. The following discussion provides an explanation of the PNP differences from the IN test conditions, mitigating design features, and basis for the selection of the potential actions and measures NMC proposes to fully address the issue described in IN 2005-26 for PNP.

<u>Pool pH vs. Time</u>

The IN described a strong correlation between the dissolution rate of CalSil and the initial pH of the solution. Dissolution of CalSil was more rapid in initially acidic solutions. Therefore, the containment sump pool pH time history is important since not all CalSil debris generated as a result of the LOCA arrives at the pool at the start of the break. The following points summarize the PNP pool pH vs. time during a large break LOCA.

- The minimum containment pool pH occurs in the time between the large break LOCA initiation and the time that containment water level reaches the bottom of the TSP baskets.
- The minimum initial pH value of the pool water at PNP is predicted to be approximately 4.6 with 0 ppm lithium.
- As a comparison with the third Integrated Chemical Effects Test (ICET-3) test concentration cited in the attachment to the IN (2800 ppm Boron and 3 ppm lithium), the PNP predicted pH values of the pool water are shown in the

following figure. Note that the addition of lithium, at the ICET-3 concentration, raises the minimum initial pH value of the pool water to approximately 5.6.



Post LOCA Sump pH Time Line in seconds

- The time the pool is predicted to reach pH greater than or equal to 7 is approximately eight minutes (440 seconds) after the large break LOCA initiation.
- The eight minute time period at a pH less than 7 is significantly less time than the thirty-five minute and four hour test durations cited in the IN for low initial pH solutions.

As shown above, the time following a large break LOCA that the containment pool solution is below a pH of 7 is approximately eight minutes. This time period is relatively short for CalSil to arrive at the sump pool and be exposed to an initially acidic solution. This period is anticipated to be much less than the time necessary to dissolve a significant proportion of CalSil debris to form the calcium ions necessary to precipitate with dissolved TSP. The quantity of CalSil expected to arrive at the sump pool, as well as a qualitative discussion of CalSil debris transport time history, is discussed in the next section.

Calcium Silicate Insulation

NMC previously reported that approximately 125 cubic feet of CalSil will transport to the PNP containment sump. NMC did not qualify the locations of this debris with respect to the LOCA Zone of Influence (ZOI) or describe any mitigating factors affecting the potentially delayed transport of this debris to the PNP containment sump, which would clarify the significant conservatisms in the previously reported CalSil transport quantity. This section describes the source of the previously reported CalSil debris generation quantities and clarifies the PNP CalSil debris generation and transport with respect to the issues cited in the IN.

In response to Generic Letter 2004-02, “Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized Water Reactors,” the CalSil insulation quantities in the PNP debris generation calculation were conservatively computed for the sump strainer head loss calculation. Chemical debris and the resulting head loss were not well understood when these calculations were performed. For the debris head loss calculation, the amount of CalSil that can be transported to the screen over the mission time for the ECCS system operation (thirty days to one year) is required. Due to this time duration, it was conservatively assumed that all the CalSil that is in the LOCA ZOI (double-ended guillotine break of the Primary Coolant System (PCS)), as well as all the unjacketed CalSil insulation outside the ZOI that may be subjected to containment spray, will erode into fines and be transported to the sump strainer. The computed CalSil quantities for the worst case break (hot leg between Steam Generator E-50A and wall) were:

CalSil in ZOI	61 cubic feet
Unjacketed CalSil outside ZOI	51 cubic feet
Marinite® Fiber	13 cubic feet
Total CalSil at sump screen	125 cubic feet

The CalSil quantities in the ZOI for the other breaks were computed to be less than 36 cubic feet.

The quantity of CalSil computed in the debris generation calculation is appropriately conservative for the sump screen head loss evaluations due to fibrous and particulate debris. The computed quantity, however, is overly conservative when used to evaluate the chemical precipitant quantity or the resulting chemical effects head loss at the sump strainer. For chemical effects evaluation, the above CalSil quantities calculated in the debris generation calculation are overly conservative because:

- Only a small fraction of the CalSil will dissolve into solution in the short time the pH is low and then be available for precipitation as chemical precipitant. The CalSil dissolution rate after the pH is buffered to neutral is expected to decrease significantly as shown by the IN 2005-26 Small Scale Dissolution Test Results.

- The CalSil debris generated in the ZOI will likely have varying size distributions and various dissolution rates. NEI 04-07, "Pressurized Water Reactor Sump Performance Evaluation Methodology," generalized the CalSil debris distribution as 50% for small fines and 50% for large pieces. For conservatism, a 100% small fines distribution was applied in the PNP debris generation calculation. The larger pieces of CalSil will dissolve more slowly when compared to the fines and thus not fully contribute to the calcium in the pool water that reacts with TSP to produce calcium phosphate precipitant.

- The unjacketed CalSil outside the ZOI will slowly erode, dissolve and contribute to calcium in the pool water only when subjected to the containment spray. Due to the physical configuration of the containment building, not all the unjacketed CalSil outside the ZOI will be subjected to the spray and thus potentially contribute to calcium in the pool water available to react with TSP and produce chemical precipitation. In the debris calculation, it was conservatively assumed that all the unjacketed CalSil will erode over the one-year maximum mission time and will be transported to the screen. The quantity of unjacketed CalSil that will not erode because it is protected from containment spray (due to wall and slab or due to the containment spray coverage) was conservatively ignored in the debris generation calculation.

- The debris generation calculations also conservatively ignored the fact that the containment spray fluid would only be at a lower pH value prior to recirculation, and would have much less of an effect on unjacketed CalSil once recirculation of the neutral pH sump water begins. In addition, emergency procedure actions to reduce and stop containment spray, once containment pressure is reduced, were not considered.

- CalSil debris generated from within the ZOI must be transported to the sump through a tortuous path with many features that could restrict the flow of this debris to the containment sump, thus delaying the time of arrival of some of the CalSil at the containment sump while also reducing the actual amount of CalSil that would be expected to arrive at the sump.

The PNP volume of CalSil debris generated and transported to the sump would be significantly less than (approximately 1-2% of) the 4000 cubic feet equivalent volume of CalSil used as the basis for the ANL test. The expected PNP volume would be reduced further due to the conservatisms described above. In addition, PNP CalSil debris must be transported to the sump through a tortuous path, thereby preventing or delaying arrival at the sump. Therefore, the amount of available CalSil for conversion to calcium ions would be expected to be significantly less than that used in the ANL test.

Overall, the combined effects of a short amount of time at lower pH (<7) values and the small amount of CalSil debris likely to initially arrive at the containment pool would be expected to result in significantly lower calcium ion concentrations than used during the ANL tests, and consequently, significantly lower calcium phosphate precipitates. Even if calcium phosphate precipitates were to form, the PNP design provides an additional safety feature that would mitigate the effects of containment sump blockage as discussed below.

PNP Reactor Cavity Flooding Design Features

PNP was designed with a cavity surrounding the reactor vessel that will flood during containment spray events. The attached Figure 6-6 from the PNP Final Safety Analysis Report (FSAR) provides a representation of the PNP reactor cavity flooding system. During events requiring containment spray, the cavity directly outside of the reactor vessel fills with a significant amount of water. This water serves to provide an external cooling function for the reactor vessel. Even though this cooling capability is not credited in the PNP design bases for accident mitigation, it provides an additional safety feature during events requiring containment spray and serves as an additional barrier protecting the public health and safety by providing significant heat transfer from the reactor vessel.

In response to BL 2003-01, NMC has completed implementation of compensatory measures at PNP that provide alternative water sources to inject into the reactor core and spray into the containment atmosphere when needed. These supplemental water sources help ensure that the reactor cavity flooding system remains full of water even if the containment recirculation flow paths were to become blocked. The reactor cavity flooding system combined with the availability of supplemental water sources provides added assurance that public health and safety will be protected during and following events requiring containment spray even if containment sump blockage were to occur.

Further Actions to Mitigate Risk

NMC believes that the compensatory measures implemented at PNP in response to BL 2003-01 are adequate to address the issues raised by IN 2005-26. However, NMC realizes additional testing concerning chemical effects is on-going and that future test results may indicate additional actions are warranted. Therefore, NMC has been reviewing, and continues to evaluate, the feasibility of removing TSP from containment at PNP in order to completely eliminate this potential chemical effect. This action is in addition to the ongoing efforts to further reduce the amount of CalSil available for transport to the containment sump.

The following is a discussion of the impact on the design and licensing basis from the potential removal of all TSP currently present on the 590' level of containment (Ref. FSAR Section 6.4 and Technical Specification (TS) Limiting Condition for Operation (LCO) 3.5.5).

Design and Licensing Basis Considerations

TSP is intended to control post-accident sump pH to between 7 and 8. There are four primary issues associated with post-accident sump pH control:

1. Retention of radioiodine in sump water
2. Rates of corrosion
3. Rates of hydrogen generation
4. Qualification of Equipment Environmental Qualification (EEQ) components

Removal of TSP without the addition of alternative buffering or complexing agents results in the loss of pH control, resulting in post-accident sump pH values of less than 7. The impact on iodine retention and radiological dose consequences is discussed below. The impact on the rate of corrosion requires further investigation, but the increased rates do not appear to be sufficient to threaten performance of safety functions for the required mission time. Preliminary evaluation based on a post LOCA sump pool pH of 4.6 indicates that material degradation is not an issue for structural and pressure boundary materials. The impact of pH on hydrogen generation and EEQ components must be addressed. Note that post-accident hydrogen generation is believed to no longer be an issue (Ref. PNP License Amendment #221, which allowed removal of hydrogen recombiners and hydrogen monitors).

Impact on Radiological Design Basis

The removal of all TSP from containment is assumed to result in the loss of ability to control post-accident sump water pH between 7 and 8. In an actual large break LOCA event with significant fuel failure, fission products (primarily cesium in the form of cesium hydroxide and cesium borate) are likely to control pH to greater than 7 for time periods on the order of 24 hours in the absence of pH control additives such as TSP (Ref. NUREG/CR-5950, "Iodine Evolution and pH Control"). However, no credit for these basic materials will be assumed in evaluating the impact of TSP removal on the radiological design basis. Use of the bounding 10 CFR 50, Appendix K fuel failures precludes crediting these basic materials.

The impact of the loss of sump pH control on radiological consequences will be assessed through the maximum hypothetical accident.

PNP design basis for radiological consequence analysis is currently in a state of transition from existing FSAR analyses based on technical information document (TID)-14844 methodologies to alternative source term (AST) methodologies. The January 30, 2004, NRC letter in response to an NEI white paper establishes the acceptability of using AST for evaluations supporting operability evaluations regarding control room habitability following tracer gas testing. NMC believes it is also reasonable to use the AST methodology to assess the impact of TSP removal on the radiological design basis.

The impact of assuming a complete loss of sump pH control on the AST methodology described in Regulatory Guide (RG) 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents At Nuclear Power Reactors," is as follows:

Source term – chemical form (RG 1.183, Appendix A, Section 2)

When sump pH is controlled to values of 7 or greater, the chemical forms of radioactive iodine released to containment are specified (RG 1.183, Section 3.5 and Appendix A, Section 2). When sump pH is not controlled to values greater than 7, the chemical forms are evaluated by the NRC on a case-by-case basis (RG 1.183, Appendix A, Section 2). Guidance exists for the chemical forms of re-evolved iodine in similar representative conditions, for example, Boiling Water Reactor (BWR) suppression pools and common spent fuel pools (RG 1.183, Appendix A, Section 5.6 and Appendix B, Section 1.3). These chemical forms for re-evolved iodine increase the fraction of the elemental iodine species. However, assuming no credit for spray removal coefficients and the relative magnitude of the remaining elemental and particulate natural deposition coefficients (1.3/hr elemental, 0.1/hr particulate), chemical forms that increase the elemental iodine fraction would result in lower predicted doses. (The basis for not crediting spray removal coefficients is discussed in the next section.) Therefore, assuming the chemical forms specified when sump pH is controlled to values of 7 or greater conforms with the intent of the guidance for loss of pH control.

Iodine scrubbing – retention in pools (RG 1.183, Appendix A, Section 3.5)

Analyses should consider iodine re-evolution if pH is not maintained greater than 7 (the discussion is specifically in context of BWR suppression pools but applies to PWR sump pools as well). Re-evolution of iodine limits the effectiveness of the containment sprays. Therefore, assuming no credit for elemental and particulate spray removal conforms with the intent of the guidance for loss of pH control.

Iodine airborne fraction – leaked sump water (RG 1.183, Appendix A, Section 5.5)

The amount of iodine that becomes airborne should be assumed to be 10% of the total iodine activity in the leaked fluid, provided the flashing fraction of the leaked fluid is less than 10%, unless a smaller amount can be justified based on the actual sump pH history. If sump pH control is lost and sump pH remains appreciably below 7, aqueous iodine will convert to elemental iodine and re-evolve. Re-evolution could occur both inside containment and outside containment after leakage. Re-evolution in containment will reduce, to some extent, the iodine concentration available for release via leaked sump fluid, whereas re-evolution outside of containment could potentially increase the airborne fraction of the remaining iodine in the leaked sump water. A mechanistic model would be required and the worst case re-evolution scenario would have to be determined in order to evaluate this situation. To avoid the complexity and associated uncertainties, sump pH is conservatively assumed to not affect the iodine concentration in the leaked sump fluid and an airborne fraction of 10% is assumed for the fluid that is leaked to the Engineered Safeguards Features (ESF) rooms. Therefore, assuming sump water leaked to the ESF room does not have reduced iodine concentration and assuming an ESF room leakage iodine airborne fraction of 10% conforms with the intent of the guidance for loss of pH control.

The Safety Injection and Refueling Water Tank (SIRWT) back-leakage justifies a lower airborne fraction based on the SIRWT pH history, which increases with time if sump pH control is maintained. Loss of sump pH control would lead to a more constant, acidic pH in the SIRWT and must be factored into the analysis. Therefore, assuming no increase in pH for sump back-leakage to SIRWT conforms with the intent of the regulation for loss of pH control.

Considering the above methodology impacts, the maximum hypothetical accident will be re-analyzed, based on the following assumptions:

- Loss of sump pH control results in complete and instantaneous re-evolution of iodine removed from the containment atmosphere due to sprays

- Loss of sump pH control results in no pH increase for post-injection residual SIRWT water upon sump back-leakage

In order to accommodate the effect of the loss of sump pH control, deviations from design basis analysis assumptions would be necessary, including exceptions to RG 1.183. These exceptions are discussed in the following paragraphs. NRC approval may be required to implement these exceptions.

For offsite dose analyses, the only expected exception to approved methodologies would be that containment leakage is based on bounding actual surveillance test results and predicted post-LOCA containment pressure profiles rather than on the TS LCO 3.6.1 L_a leakage limit for 24 hours and half the TS value for the remaining 29 days. The specific exception would be to RG 1.183, Appendix A, Section 3.7.

For control room dose analyses of the current PNP configuration, the same exception for containment leakage is expected, as described for offsite dose analyses. Other exceptions and deviations are expected to be as described in the following paragraphs.

Atmospheric dispersion factors would be based on wind tunnel testing data. While this is not an exception to RG 1.194, "Atmospheric Relative Concentrations For Control Room Radiological Habitability Assessments At Nuclear Power Plants," Section 7, atmospheric dispersion factors based on wind tunnel data have not been approved for use in design basis calculations at PNP.

Radionuclide source term would be based on bounding 10 CFR 50, Appendix K fuel failures rather than 100% fuel melt. While this is not an exception to the 10 CFR 50.67 direction that "fission product release assumed... should be based upon a major accident, ...Such accidents have generally been assumed to result in substantial meltdown of the core with subsequent release of appreciable quantities of fission products," it would be an exception to the release fractions specified in Table 2 of RG 1.183.

ESF leakage would be based on bounding actual leakage surveillance test results rather than on the TS 5.5.2, "Primary Coolant System Outside Containment Program," limits. The specific exception would be to RG 1.183, Appendix A, Section 5.2.

Additionally, numerous best-estimate values of parameters with some conservatisms would be employed, such as reactor power and core average burnup. The specific exception would be to RG 1.183, Sections 5.1.1 and 5.1.3.

In summary, the following exceptions to RG 1.183 would be needed:

1. To meet offsite dose limits without TSP, NMC expects to require the following exception:

    - Containment leakage based on bounding actual leakage surveillance test results and predicted post-LOCA pressure profiles rather than on the TS LCO 3.6.1 L_a limit

2. To meet control room limits without TSP, NMC expects to require the following exceptions:

   - Containment leakage based on bounding actual leakage surveillance test results and predicted post-LOCA pressure profiles rather than on the TS LCO 3.6.1 L_a limit

   - Atmospheric dispersion factors based on wind tunnel testing data rather than ARCON96 code (Ref. NUREG/CR-6331, "Atmospheric Relative Concentrations in Building Wakes")

   - Radionuclide source term based on bounding 10 CFR 50, Appendix K fuel failures rather than 100% fuel melt

   - ESF leakage based on bounding actual leakage surveillance test results rather than on the TS 5.5.2 limits

   - Use of bounding best-estimate values for some parameters

In summary, NMC has identified several issues that will need to be fully addressed to support removal of TSP in an expeditious manner to completely eliminate the potential chemical precipitants discussed in IN 2005-26, pending long term resolution of Generic Safety Issue (GSI) 191. This discussion concentrated on radiological dose consequences, but all relevant issues will need to be addressed. NMC intends to pursue a one-cycle License Amendment Request (LAR) for TSP removal expeditiously.

Conclusion

NMC has reviewed the compensatory measures taken in response to BL 2003-01, in light of IN 2005-26, and concluded that these measures are similarly effective for chemical effects type debris. NMC has reviewed the IN test results and identified several significant differences between PNP conditions and the ANL test conditions. These differences suggest that calcium phosphate precipitants would be significantly less than observed during the ANL tests. In addition, the PNP design contains a reactor cavity flooding system that serves as an additional barrier to protect public health and safety by mitigating the effects of containment sump blockage during events requiring containment spray. Finally, in addition to the ongoing efforts to further reduce the amount of CalSil available for transport to the containment sump, NMC has expended considerable effort to evaluate expeditious removal of TSP from containment, as summarized in the following paragraph.

Based on the approach outlined above, offsite doses are expected to remain within limits with TSP removed, using established design basis methods with the single exception of reduced containment leakage. Control room doses are expected to remain within limits with TSP removed, based on design basis methods with the exceptions described above. NMC believes the remaining technical issues concerning corrosion, hydrogen generation and EEQ can be adequately addressed.

PNP TS LCO 3.5.5 requires the presence of between 8,300-11,000 lbs of TSP in containment. NMC plans to submit a LAR to remove TSP from containment at PNP for one operating cycle, at which time the GSI-191 modifications will be implemented in accordance with the GL 2004-02 timeline. NMC requests a meeting with the NRC staff, prior to the end of January 2006, to discuss the TSP removal LAR strategy. NMC desires to obtain approval of the LAR in time to support removal of TSP from the PNP containment during the 2006 refueling outage, scheduled to commence on April 2, 2006.

REACTOR CAVITY FLOODING SYSTEM