December 31, 2003

EA-03-180

Mr. Daniel J. Malone
Site Vice President
Palisades Nuclear Plant
Nuclear Management Company, LLC
27780 Blue Star Memorial Highway
Covert, MI 49043-9530

SUBJECT: FINAL SIGNIFICANCE DETERMINATION FOR A WHITE FINDING (NRC INSPECTION REPORT 50-255/03-05)

Dear Mr. Malone:

The purpose of this letter is to provide you with the final results of our significance determination of the preliminary White finding discussed in the subject inspection report and by letter dated October 3, 2003. The inspection finding was assessed using the Significance Determination Process and was preliminarily characterized as White (i.e. a finding with low to moderate safety significance, which may require additional NRC inspection). This preliminary White finding was identified when on March 25, 2003, with your Palisades Nuclear Plant shutdown for a planned refueling outage, a loss of offsite power and loss of shutdown cooling event occurred when a signpost being installed in the plant parking lot was driven into a conduit and damaged a cable which contained a combination of energized indication circuitry and de-energized protective relaying circuitry. The metal signpost cut and shorted together several of the conductors within the cable, generating a fault signal to the breakers supplying offsite power to the plant, causing the event.

At your request, a Regulatory Conference was held on November 25, 2003, to further discuss your views on this issue. At the Regulatory Conference, your staff presented an overview of the event, related corrective actions, and the methodology and results of their independent safety assessment of the preliminary White finding. During the meeting, your staff also provided the results of simulator activities and thermal-hydraulic analyses accomplished following the identification of this issue as a preliminary White finding. Based upon the additional information presented by your staff, you concluded that the finding would be more properly characterized as Green, that is, a finding of very low safety significance. The NRC and your staff held extensive discussions regarding specific technical issues related to your assessment. Topics discussed included the assumption that operators would be under only low to moderate stress levels during a station blackout event, as well as the use of a thermal-hydraulic computer code which had not been formally reviewed by the NRC or benchmarked against appropriate test data.

With regard to operator stress levels, you presented information regarding the time available to operators to perform necessary recovery actions to prevent core damage. Your judgement of operator stress at a low to moderate level was based, in large part, on the time required to perform these recovery actions to prevent core damage in comparison to the time available to take these actions prior to the onset of core damage. In addition, to account for the low to moderate stress you judged to exist, you chose to reduce the human error probabilities estimated in a moderate to high stress environment by a factor of two, since the Accident Sequence Evaluation Program (ASEP) methodology you employed did not provide human error probabilities for a low to moderate stress environment. We have reviewed your methodology in this matter and have concluded that the introduction of this factor is not supported by the ASEP methodology and should not be applied. In addition, because operators are generally not specifically aware of the time available and the time required to take necessary recovery actions to prevent core damage following an event, such as a station blackout event, we do not believe that the rationale you used to determine the stress levels of the operators is sufficiently justified. It is our view that the very nature of a station blackout event would likely increase the stress levels of operators above the low to moderate levels that you assumed in your independent safety assessment, since this significant event is seldom encountered. Based on these concerns, we have concluded that your presentation did not provide us with the necessary justification to assume a low to moderate stress level of operators in the calculations you performed to develop your independent safety assessment.

With regard to the thermal-hydraulic model used to determine the time available to operators to address station blackout scenarios to preclude core damage, you utilized a model which had not been formally reviewed by the NRC. During our discussions with you regarding the model which you used, you did not demonstrate that your model had been sufficiently benchmarked against a model previously validated by the NRC or against actual test data to provide a reasonable level of assurance that the results from your model could reasonably predict actual critical plant parameters during station blackout scenarios. Therefore, to determine if the two models were similar, we ran a validated NRC model using the same input assumptions that you used in your model and compared the results. We also performed hand calculations to determine the time to core damage in a station blackout scenario and compared those results with the results from the validated model. Following these reviews, we determined that the results of your model were not similar enough to the results of our model and hand calculations to provide us with the measure of confidence necessary to accept your results. We also recognize that during your presentation you did not provide any information which would suggest that the NRC's thermal-hydraulic analysis model was deficient or that the results obtained from the model were incorrect. Therefore, we chose to use the results from the validated NRC model as a basis for our significance determination.

Therefore, after considering the information developed during the inspection; the additional information you provided in your letter dated October 28, 2003; the information you provided at the conference; and based on the NRC's independent analysis; the NRC has concluded that the inspection finding is appropriately characterized as White, that is, an issue with low to moderate safety significance which may require additional NRC inspection.

You have 30 calendar days from the date of this letter to appeal the staff's determination of significance for the identified White finding. Such appeals will be considered to have merit only if they meet the criteria given in NRC Inspection Manual 0609, Attachment 2.

Because no violation of regulatory requirements was identified, no specific response to this letter is required.

Because plant performance for this issue has been determined to be in the regulatory response band, we will use the NRC Action Matrix to determine the most appropriate NRC response for this event. We will notify you by separate correspondence of that determination.

In accordance with 10 CFR 2.790 of the NRC's "Rule of Practice," a copy of this letter and your response, if any, will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's document system (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

Steven A. Reynolds, Acting Director
Division of Reactor Projects

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