June 13, 2002

Mr. Douglas E. Cooper
Site Vice President
Palisades Plant
27780 Blue Star Memorial Highway
Covert, MI 49043-9530

SUBJECT: PALISADES PLANT - REQUEST FOR ADDITIONAL INFORMATION REGARDING RISK-INFORMED INSERVICE INSPECTION PROGRAM (TAC NO. MB4420)

Dear Mr. Cooper:

The NRC staff is reviewing your letter of March 1, 2002, requesting approval to implement a risk-informed inservice inspection (RI-ISI) program as an alternate to the American Society of Mechanical Engineers Code, Section XI, ISI requirements for piping at Palisades. We find that additional information, as identified in the enclosure, is needed to complete our review.

On the basis of discussions with R. Gerling of your organization, a mutually agreeable date for your response is within 60 days of the date of this letter.

If you have questions regarding this letter or if unexpected circumstances prevent you from meeting the response date, please contact me at (301) 415-3049 or by e-mail at dsh@nrc.gov.

Sincerely,

/RA/

Darl S. Hood, Senior Project Manager, Section 1
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-255

Enclosure: Request for Additional Information

cc w/encl: See next page

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REQUEST FOR ADDITIONAL INFORMATION

RISK-INFORMED INSERVICE INSPECTION (RI-ISI) PROGRAM

PALISADES PLANT

1. In the enclosure to the Nuclear Management Company's (the licensee's) letter dated March 1, 2002, Table 3.7-1 and the associated notes indicate that, on the basis of their judgement, the expert panel moved 27 piping segments having risk-reduction-worth (RRW) values greater than 1.005 from the category of "high safety significant" (HSS) to "low safety significant" (LSS). The Nuclear Regulatory Commission (NRC) staff recognizes that Topical Report WCAP-14572, Revision 1-NP-A, allows the expert panel to use deterministic information to place segments with RRW values greater than 1.005 into the LSS category, but page 143 of the Topical Report states that HSS "segments should not be classified lower by the expert panel without sufficient justification that is documented as part of the [RI-ISI] program. The expert panel should be focused primarily on adding piping to the higher classification." The expert panel apparently used three factors to move HSS segments to LSS: (1) non-proceduralized operator actions, (2) proceduralized operator actions, and (3) other considerations. Each of these is addressed below.

1.1 Operator Actions

The notes in Table 3.7-1 indicate that some of the segments were placed in the LSS category based upon the expert panel's judgment that the "with operator action" RRWs are credible and may be used as a basis for the segment classification, while the "without operator action" RRWs are inappropriate and discarded. The "with" and "without" operator action rankings are intended to reduce the impact of the highly uncertain human error evaluation on the categorization to facilitate preparation and review of short, template RI-ISI relief requests. Reducing the safety significant category based upon the expert panel's judgement about the likelihood of successful operator actions weakens this important element in the approved process.

1.1.1 Non-Proceduralized Operator Actions

The notes in Table 3.7-1 indicate that some of the operator actions credited by the expert panel to move HSS segments into the LSS category are not proceduralized actions. Crediting non-proceduralized operator actions is generally not acceptable in PRA analyses unless they are simple, skill-of-the-craft actions such as manually starting a standby pump following a failure of auto-start. When the expert panel moves a HSS segment into the LSS category based upon an operator action, the judgement on the incredibility of failure of the operator to perform the required action becomes the dominate contributor to the final disposition of the segment in RI-ISI, and thereby of the inspection requirements in the segment. The actions required to mitigate pipe ruptures must be taken in response to highly unusual and stressful events such as

loss-of-coolant accidents (LOCAs) outside containment. The selection of the “with human action” RRW as the only credible result assumes that the operator will always succeed. This assumption is inconsistent with acceptable probabilistic risk assessment (PRA) methodologies because it assumes the non-proceduralized action will always be successfully performed, and inconsistent with the approved RI-ISI methodology where the RRW values with and without operator action are used to reduce the sensitivity of the results on the highly uncertain human error evaluation. The uncertainties in the evaluation of these actions are further increased due to the non-proceduralized action.

Because discarding the "without operator action" RRW negates a major element of the approved methodology, and recognizing the greater than normal uncertainty associated with evaluating the likelihood of success of non-proceduralized actions, the NRC staff believes that sufficient justification does not exist for moving HSS segments to the LSS category based on the judgement of the expert panel about the incredibility of the “without operator action” RRW for non-proceduralized operator actions. Please identify all piping segments placed in the LSS category based upon crediting non-proceduralized actions, place the segments in the category specified in the Topical Report (e.g., HSS if any of the four RRWs is greater than 1.005), and modify your inspection location selection accordingly.

1.1.2 Proceduralized Operator Actions

The notes in Table 3.7-1 indicate that some of the operator actions credited are proceduralized. Proceduralized actions must be more than credible before the “without operator action” RRWs may be discarded. The failure of proceduralized operator actions are not discarded (i.e., success assumed with a probability of 1.0) in PRA analyses. When the expert panel moves a HSS segment into the LSS category based upon an operator action, the judgement regarding the likelihood of the action becomes the dominate contributor to the final disposition of the segment in the RI-ISI. Justification and documentation of each of these actions must be of sufficient quality to support this judgement as the final arbitrator of safety significance. The justification for each action should include:

- Identification of the procedure containing the required action.
- The indications available to the operators to identify the specific action.
- The location of the action.
- The time available to perform the action.
- The time required to perform the action.
- Identification and characterization of the performance shaping factors that might influence the ability of the operators to accomplish the task.
- An integrated discussion of the above information justifying that the failure of the operator to perform the action is of such a low likelihood that the “without operator action” RRW may be discarded.

Please identify all piping segments placed in the LSS category based upon crediting proceduralized actions and provide a copy of the justification and documentation developed by the expert panel for each segment. If the original documentation developed by the expert panel does not include the information discussed above, please provide a copy of the original documentation that was developed, and also provide the requested information.

1.2 Other Considerations

The notes in Table 3.7-1 indicate that a number of piping segments were moved from the HSS category to the LSS category by the expert panel based upon considerations other than the potential for operator actions. The justification should include:

- The specific weakness in the quantitative evaluation that causes the segment to be inappropriately placed in HSS.

- A discussion of the more appropriate assumption that corrects the weakness.

- A discussion of the magnitude of the impact of the more appropriate assumption that supports moving the HSS segment into LSS.

For example, Note 7 states that there are “no active failure mechanism” in several pressurizer (PRZ) segments and that they “would be subjected to the lowest temperature of all PCS [primary coolant system]/PRZ segments”. If the environment in these segments is so benign, why were they HSS according to the RRW values? What is the inappropriate assumption that lead to the HSS categorization? What is the more appropriate assumption and how much does it influence the consequence and/or the frequency of the segments’ failure? Please identify each of these segments and provide a copy of the justification developed and documented by the expert panel to move the segment from the HSS to the LSS category. If the original documentation developed by the expert panel does not include the information discussed above, please provide the original documentation that was developed, and also provide the requested information.

2. In the NRC Staff Evaluation Report (SER) on the Palisades Individual Plant Examination (IPE), dated February 7, 1996, the NRC staff concluded that there were limitations in the human reliability assessment (HRA) approach used by the licensee that could limit the Palisades IPE in future regulatory uses. These limitations include:

- Treatment of pre-initiator and post-initiator errors using the Technique for Human Error Rate Prediction which limits the degree of insights about plant-specific factors influencing human performance.

- Use of screening values that are significantly lower than values typically used for post-initiator actions and not including dependencies in the initial quantification.

- Treatment of diagnosis for post-initiator actions, which is not consistent with most nuclear power plant HRAs.

- In one of the two post-initiator actions that was quantified using the Accident Sequence Evaluation Program model, the calculated human error probability (HEP) was the lowest value of all post-initiator HEPs.

Please explain how these limitations in the HRA have been addressed during the evaluations performed in support of this relief request.

3. In the enclosure to your letter dated March 1, 2002, Section 1.2, "PSA Quality," states that the Combustion Engineering Owners Group peer review performed in May 2000 found a weakness in performing a thorough dependancy analysis for the operator actions modeled in the PSA. You state that the "weakness related to the appropriateness of the magnitude of human error probabilities used in the model given the possibility that dependent operator actions may not have been adequately considered." In general, omitting the dependencies between multiple operator actions yields lower human error rates for a sequence of actions. Your reported investigation that observed a slight increase in risk after including the dependencies in a number of multiple actions is consistent with this general observation. However, on page 2, you continued that you chose not to "remove further some of the conservatism by assessing more human error combinations...[because]...additional evaluation is not considered necessary due to the already small increase in [core damage frequency] CDF."

   Including dependencies between human actions does not "remove conservatism" -- it removes non-conservatism. The small increase in CDF observed from the completed evaluation does not alone support the conclusion that further evaluation would not yield larger increases.

   Please explain how the incomplete review of dependencies could affect the evaluations performed in support of this relief request.

4. On page 3 of the enclosure to your letter dated March 1, 2002, in the section on "Deviations," you state that four piping segments in the safety injection/refueling water tank and containment sump system are in a category of thin-walled piping and have attributes of low pressure, low temperature, low design stress, and low fatigue stress, that are not appropriate for use with the Westinghouse statistical (Perdue) model. These attributes do not appear to be outside the bounds of applicability of the Perdue model listed on page 178 of WCAP-14572. You further state that a 7.5-percent sample was selected from each of the segments as an alternative to running the Perdue model. Please explain why the Perdue model is not applicable to these four piping segments.

5. In Section 3.1 on page 4 of the enclosure to your letter dated March 1, 2002, you state that the reactor cavity flood system was excluded from system scope consideration in the RI-ISI program. What is the basis stated in the documentation maintained at the site for excluding this system?

6. Table 5-1 of the enclosure to your letter dated March 1, 2002, shows that, for the primary coolant system, there are 5 volumetric examinations in the 29 HSS piping segments. Please explain why only 5 volumetric examinations are performed.

7. Will the RI-ISI program be updated every 10 years and submitted to the NRC consistent with the current ASME Code, Section XI, requirements?

8. Under what conditions would the RI-ISI program be resubmitted to the NRC before the end of any 10-year interval?

9. Since 66 percent of the scheduled examinations under the RI-ISI program are being examined by the end of the third inspection interval, how will the welds be selected in terms of risk category?