October 14, 2004

Mr. Thomas Coutu
Site Vice President
Kewaunee Nuclear Power Plant
Nuclear Management Company, LLC
N490 State Highway 42
Kewaunee, WI 54216

SUBJECT: KEWAUNEE NUCLEAR POWER PLANT - REQUEST FOR ADDITIONAL INFORMATION REGARDING RISK-INFORMED INSERVICE INSPECTION PROGRAM PLAN (TAC NO. MC2537)

By letter dated December 16, 2003, Nuclear Management Company, LLC, submitted its risk-informed inservice inspection program plan for the fourth 10-year inservice inspection interval. Based on the Nuclear Regulatory Commission staff's review of your program plan, please provide additional information as discussed in the enclosure to this letter.

The enclosed request was discussed with Mr. G. Riste of your staff on October 12, 2004. A mutually agreeable target date of December 17, 2004, for your response was established. If circumstances result in the need to revise the target date, please contact me at (301) 415-2296 at the earliest opportunity.

Sincerely,

/RA/

Carl F. Lyon, Project Manager, Section 1
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-305

Enclosure: Request for Additional Information

cc w/encl: See next page

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REQUEST FOR ADDITIONAL INFORMATION

FOURTH 10-YEAR INTERVAL

RISK-INFORMED INSERVICE INSPECTION PROGRAM PLAN

KEWAUNEE NUCLEAR POWER PLANT

NUCLEAR MANAGEMENT COMPANY, LLC

DOCKET NO. 50-305

1.0 INTRODUCTION

By letter dated December 16, 2003, Nuclear Management Company, LLC (the licensee), proposed an alternative to the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section XI, Rules for Inservice Inspection of Nuclear Power Plant Components. In accordance with Title 10 of the *Code of Federal Regulations* (10 CFR), Section, 50.55a(a)(3)(i), the licensee proposed to use a Risk-Informed Inservice Inspection (RI-ISI) Program as an alternative to certain requirements listed in Section XI for the inspection of ASME Code Examination Categories B-F, B-J, C-F-1 and C-F-2 piping at Kewaunee Nuclear Power Plant (KNPP). The alternative is based on the risk-informed process described in Electric Power Research Institute (EPRI) Topical Report (TR) 112657 Rev. B-A "Revised Risk Informed Inservice Inspection Evaluation Procedure".

Inservice inspection of ASME Code Class 1, 2, and 3 components is to be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code, and applicable addenda, as required by 10 CFR 50.55a(g), except where specific relief has been granted by the Nuclear Regulatory Commission (the Commission or NRC) pursuant to 10 CFR 50.55a(g)(6)(i). The regulation at 10 CFR 50.55a(a)(3) states that alternatives to the requirements of paragraph (g) may be used, when authorized by the NRC, if the licensee demonstrates that (i) the proposed alternatives would provide an acceptable level of quality and safety or (ii) compliance with the specified requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

Pursuant to 10 CFR 50.55a(g)(4), ASME Code Class 1, 2, and 3 components (including supports) shall meet the requirements, except the design and access provisions and the preservice examination requirements, set forth in the ASME Code, Section XI, Rules for Inservice Inspection (ISI) of Nuclear Power Plant Components, to the extent practical within the limitations of design, geometry, and materials of construction of the components. The regulations require that inservice examination of components and system pressure tests conducted during the first 10-year interval and subsequent intervals comply with the requirements in the latest edition and addenda of Section XI of the ASME Code, which was incorporated by reference in 10 CFR 50.55a(b) 12 months prior to the start of the 120-month interval, subject to the limitations and modifications listed therein. The Code of Record for the KNPP fourth 10-year interval inservice inspection program is the 1998 Edition of Section XI of the ASME Boiler and Pressure Vessel Code, with the 2000 Addenda.

2.0 SCOPE

The licensee's RI-ISI Program alternative was submitted in summary form developed under the guidelines described in EPRI TR 112657 Rev. B-A "Revised Risk Informed Inservice Inspection Evaluation Procedure." The RI-ISI Program encompasses all Code Class 1 and 2 piping at KNPP. The NRC staff has reviewed the licensee's submittal and, based on this review, determined the following information is required to complete the evaluation.

3.0 ADDITIONAL INFORMATION REQUIRED

3.1 KNPP states that the quantitative risk impact assessment was performed using the "Simplified Risk Quantification Method" described in Section 3.7 of EPRI TR-112657 and that the PBF [pressure boundary failure] is determined by the presence of different degradation mechanisms. KNPP states that best estimate and upper bound failure frequencies assumed in the delta risk assessments were consistent with assumptions made in original EPRI RI-ISI pilot studies at Vermont Yankee (boiling-water reactor) and Arkansas Nuclear One, Unit 2 (Combustion Engineering pressurized-water reactor). Provide justification that you meet the guidance in Fig 3-6 of the EPRI TR-112657 and don't need to perform the more realistic estimates for Delta CDF [change in core damage frequency] and Delta LERF [change in large early release frequency].

3.2 In the risk impact analysis, KNPP assumes that the RI-ISI probability of detection (POD) for service-induced fatigue cracks resulting from thermal stratification, cycling and striping (TASCS) and thermal transients degradation mechanisms is 0.9. A recent POD study by EPRI, DOE [Department of Energy], and NRC, documented in EPRI Guideline MRP-82, developed fatigue crack POD curves for carbon and stainless steel piping based on crack detection data assembled as part of the industry's performance demonstration initiative. These results show that, for all thicknesses of carbon/low alloy steel piping and austenitic stainless steel piping with wall thicknesses < 0.4-inches, the best estimate POD = 0.9 is a reasonable assumption. However, for thicker stainless steel pipes, this assumption is too optimistic with respect to detection of fatigue cracks with through-wall depths identified in Section XI flaw acceptance standards. For stainless steel pipes with section thicknesses greater than or equal to 0.4-inches, the best estimate and 95 percent confidence lower bound PODs for 10 percent through-wall fatigue cracks were approximately 0.8 and 0.7, respectively. Discuss the impact that these recent studies have on the delta risk input results. Please describe the risk impact considering upper bound failure frequencies and 95 percent confidence lower bound POD, as applicable.

3.3 It is noted that Table 5-2 is intended to summarize and compare new RI-ISI with existing ASME examinations, and this table lists relevant degradation mechanisms for elements (examination locations) by plant system, and includes other relevant information. In order to determine if appropriate examination methods are being correctly applied to target specific degradation, further clarification is necessary. Please "break-out" the planned methods for examination, i.e., show how many volumetric or surface examinations will be applied as a result of the RI-ISI process, instead of listing these only as "NDE" or confirm that NDE [nondestructive examination] implies volumetric examination (typically ultrasonic testing). What NDE method will be used for Category 4 items selected for inspection?

3.4 In Section 3 of the KNPP submittal, it is stated that a deviation to the EPRI RI-ISI methodology has been implemented in the failure potential assessment for the damage mechanism of TASCS. For clarification, provide confirmation to the following two items pertaining to the assessment of TASCS:

a) Confirm that the methodology for assessing TASCS in the proposed RI-ISI program is identical to the materials reliability program (MRP) methodology in EPRI TR-000701, "Interim Thermal Fatigue Management Guideline (MRP-24)," January 2001.

b) Currently MRP-24 is an interim document that is undergoing review by EPRI and the NRC. Please confirm that once the final MRP document is published, KNPP will incorporate the applicable final guidance (as approved, restricted, or amended by the NRC) into the KNPP RI-ISI program.

3.5 Several issues are unclear regarding existing “augmented” examination programs at KNPP and how these may be impacted due to the RI-ISI evaluation:

a) With regard to inspections required as a result of IE Bulletin (IEB) 79-17, Pipe Cracks in Stagnant Borated Water Systems at PWR Plants, the licensee argues that no augmented inspections will be performed on stagnant borated lines because, as stated by the licensee, “a review performed by KNPP and the historical performance of the weldments to-date signify that these systems will not experience stress corrosion cracking.”

Similarly, the licensee stated that no augmented inspections will be performed on pipe segments associated with IE Bulletin 88-08, Thermal Stresses in Piping Connected to Reactor Coolant Systems, because examinations that KNPP committed to perform have not revealed any indications.

It is not clear whether inspections performed as a result of IEB 79-17 and IEB 88-08 constitute augmented examinations. Further, the technical justifications provided by the licensee are not adequate for the staff to conclude that the subject piping segments should not be included in the RI-ISI evaluation. If the examinations performed as a result of the subject IEBs are used as a basis to exclude pipe segments from evaluation under the RI-ISI program, then this may constitute a deviation from the process described in the EPRI TR.

Please discuss the basis and requirements for IEB inspections at KNPP, and the relationship between the IEB inspections and the RI-ISI evaluations performed at KNPP. In the discussion include the extent and frequency of examinations performed as a result of commitments made to NRC, and provide a clear justification for why the subject piping segments are not included in the RI-ISI evaluation.

3.6 In Section 3.1, KNPP states that additional plant information including the existing plant ISI program were used to define the Class 1 and 2 piping system (RI-ISI evaluation) boundaries. The submittal does not describe what additional plant information has been used nor how this information was used to define the system RI-ISI evaluation scope.

Since Class 1 and 2 piping are well defined, how was the existing ISI program used to define the evaluation scope? Did the RI-ISI evaluation scope include Class 2 piping segments that are normally exempt from examination in the existing ISI program according to IWC-1220? If not, please explain why these portions of IWC-1220 exempt piping have not been included in the RI-ISI evaluation.

3.7. From the NRC staff's review of the safety evaluation report of the KNPP individual plant examination (IPE), it appears that there were no identified weaknesses with the IPE methodology. Please confirm that this is correct, or otherwise indicate 1) what weaknesses were identified and 2) what was done to correct the identified weaknesses (in subsequent revisions to the KNPP IPE), or why the uncorrected weaknesses are not relevant to the RI-ISI application.

3.8 In Section 1.2 of your submittal, you note that an industry peer review was completed in June 2002, which concluded that "the Kewaunee probabilistic risk assessment (PRA) could be effectively used to support applications involving risk significance determinations supported by deterministic analyses once the Facts and Observations (F&Os) noted in the report are addressed." You then provide details for five Category A F&Os. Below are questions pertaining to two of these.

a) "Long-term condensate storage tank inventory is not appropriately modeled for the loss of service water scenario. The resolution of this issue showed that it did not have a major effect on results."

Please explain why the impact on the consequence ranking of pipe segments would not be affected by the resolution of this F&O.

b) "The bases for the time windows for human actions are not well defined. Work is in progress on resolving this F&O. Preliminary results show that the human error probabilities (HEPs) in the model tend not to be greatly affected by the new time windows."

Please indicate whether or not there have been any changes to the above preliminary results. If so, please identify any significant increases in HEP values, and whether or not these increased HEP values may have an impact on the consequence ranking of pipe segments.

3.9 Your submittal states that, from the June 2002 Westinghouse Owners Group Peer Review, there were 49 Category B F&Os.

a) Please provide these F&Os, and identify those which have already been addressed in the 0101 PRA model.

b) For those F&Os not already addressed in this model, you indicate that none of them would result in a significant increase in failure rates or consequences. Please provide your rationale that led to this conclusion for each of these F&Os.

3.10 Section 2.2 of your submittal lists one augmented inspection program (Generic Letter 89-08) that was considered during the development of this RI-ISI application. You also

provided with your submittal a more detailed description of augmented inspection programs at KNPP in Appendix F of Structural Integrity Calculation/File No. NMC-01-343, “Risk Impact Analysis for the Kewaunee Nuclear Power Plant”, Revision 0, July 29, 2003. With the exception of your information about the relationship between Generic Letter 89-08 and the RI-ISI program, it appears that the other augmented inspection programs are completely independent of the RI-ISI program.

a) Please confirm that none of the programs listed in the above Appendix F is subsumed into the RI-ISI program, or otherwise identify which one are subsumed.

b) Please confirm that none of the piping-related nondestructive examination (NDEs) to be performed for the Appendix F augmented inspection programs is being credited toward the count of NDEs (locations) for RI-ISI. Otherwise, please describe which programs you intend to do this with, how many welds from each program you intend to credit toward the count of NDEs for RI-ISI, and why you feel that the prescribed augmented inspection NDE will adequately suffice for the examination method(s) required under RI-ISI.

Kewaunee Nuclear Power Plant

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