May 19, 2003

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

SUBJECT: PALISADES PLANT - RISK-INFORMED INSERVICE INSPECTION PROGRAM (TAC NO. MB4420)

Dear Mr. Cooper:

By letter dated March 1, 2002, as supplemented on August 8, 2002, and February 28, 2003, Nuclear Management Company, LLC (NMC) submitted a request for relief, “Alternate American Society of Mechnical Engineers (ASME) Code, Section XI, Risk-Informed Inservice Inspection Program” for implementation during the second inspection period of the third 10-year inspection interval for the Palisades Nuclear Plant.

The proposed risk-informed inservice Inspection (RI-ISI) program, that includes ASME Class 1, 2, 3, and non-class piping systems, is an alternative to the current ASME Code, Section XI inservice inspection program. The proposed RI-ISI program was developed in accordance with Westinghouse Owners Group topical report WCAP-14572, Revision 1-NP-A. The U.S. Nuclear Regulatory Commission (NRC) staff has reviewed the relief request. Based on its review of the information provided by NMC, the NRC staff concludes that the licensee’s proposed RI-ISI program for ASME Class 1, 2, 3 and non-class piping systems is an acceptable alternative to the requirements of the ASME Code, Section XI for inservice inspection of Code Class 1 piping, Categories B-F and B-J welds and Class 2 piping, Categories C-F-1 and C-F-2 welds.

In addition, NMC proposed to perform a visual (VT-2) examination each outage in lieu of volumetric or surface examination for those high safety significant ASME Code Class 1 and 2 socket welds of two-inch diameter or less identified in the RI-ISI program. The NRC staff concurs that volumetric examination of socket welds is inconclusive due to geometric limitations imposed by a socket weld. The NRC staff also concurs that it is not necessary to perform the Code-required surface examination of socket welds in the absence of an environment which would cause outside surface-initiated flaws.

NMC's request for relief is authorized pursuant to 10 CFR 50.55a(a)(3)(i) on the basis that the proposed alternative provides an acceptable level of quality and safety. The enclosed safety evaluation authorizes application of the proposed RI-ISI program during the third 10-year ISI interval for the Palisades Nuclear Plant. Please contact the NRC Project Manager, Johnny Eads at (301) 415-1471 if you have any questions.

Sincerely,

/RA/

L. Raghavan, Chief, Section 1
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-255

Enclosure: Safety Evaluation

cc w/encl: See next page

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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RISK-INFORMED INSERVICE INSPECTION PROGRAM

NUCLEAR MANAGEMENT COMPANY, LLC

PALISADES PLANT

DOCKET NO. 50-255

1.0 INTRODUCTION

By letter dated March 1, 2002, as supplemented on August 8, 2002, and February 28, 2003, Nuclear Management Company, LLC (NMC) submitted a request for relief, "Alternate American Society of Mechnical Engineers (ASME) Code, Section XI, Risk-Informed Inservice Inspection (RI-ISI) Program" for implementation during the second inspection period of the third 10-year inspection interval for the Palisades Nuclear Plant (References 1, 2 and 3).

2.0 BACKGROUND

2.1 Applicable Requirements

Title 10 Part 50.55a(g) of the *Code of Federal Regulations* (10 CFR) requires that ISI of the ASME Code Class 1, 2, and 3 components be performed in accordance with Section XI of the ASME Code and applicable addenda, except where specific relief has been granted by the Commission pursuant to 10 CFR 50.55a(g)(6)(i). Pursuant to 10 CFR 50.55a(g)(4), ASME Code Class 1, 2, and 3 components (including supports) shall meet the requirements set forth in the Code, to the extent practical within the limitations of design, geometry, and materials of construction of the components. The regulations require that ISI of components conducted during the first 10-year interval and subsequent intervals comply with the requirements in the latest edition and addenda of the Code incorporated by reference in 10 CFR 50.55a(b) twelve months prior to the start of the 120-month interval, subject to the limitations and modifications listed therein.

10 CFR 50.55a(a)(3) states in part that alternatives to the requirements of paragraph (g) may be used, when authorized by the Nuclear Regulatory Commission (NRC), if the applicant demonstrates that the proposed alternatives would provide an acceptable level of quality and safety or if the specified requirement would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety. Further guidance in defining acceptable methods for implementing an RI-ISI program is also provided in Regulatory Guide (RG) 1.174, RG 1.178, and Standard Review Plan (SRP) Chapter 3.9.8. In accordance with the guidance provided in RGs 1.174 and 1.178, an engineering analysis of the proposed changes is required using a combination of traditional engineering analysis and supporting insights from the probabilistic risk assessment (PRA).

RG 1.174 defines the following safety principles that should be met in an acceptable RI-ISI program: (1) the proposed change meets current regulations unless it is explicitly related to a requested exemption, (2) the proposed change is consistent with the defense-in-depth philosophy, (3) the proposed change maintains sufficient safety margins, (4) when proposed changes result in an increase in risk, the increases should be small and consistent with the intent of the Commission's Safety Goal Policy Statement, and (5) the impact of the proposed changes should be monitored using performance measurement strategies.

The proposed program maintains the fundamental requirements of ASME Code, Section XI, such as the examination technique, examination frequency and acceptance criteria. However, the proposed program reduces the number of required examination locations significantly and is able to demonstrate that an acceptable level of quality and safety is maintained. Thus, the proposed alternative approach is based on the conclusion that it provides an acceptable level of quality and safety and, therefore, is in conformance with 10 CFR Section 50.55a(a)(3)(i).

For Class 1, 2, and 3 piping components, the licensee is currently required to perform ISI of ASME Code Category B-F, B-J, C-F-1, and C-F-2 piping welds during successive 120-month (10-year) intervals. Excluding piping exempted from volumetric and surface examination, currently all B-F welds and 25 percent of Category B-J welds and 7.5 percent of all C-F-1 and C-F-2 welds are selected for volumetric and/or surface examination based on existing stress analyses and cumulative usage factors. According to the information provided in Reference 1, Palisades is currently in the second period of the third 10-year inspection interval.

2.2 Summary of Proposed Approach

Current ISI requirements for the Palisades Nuclear Plant are contained in the 1989 Edition of Section XI, Division 1 of the ASME Boiler and Pressure Vessel Code, entitled *Rules for Inservice Inspection of Nuclear Power Plant Components* (hereinafter called Code). The licensee has developed a RI-ISI program in accordance with the Westinghouse Owners Group (WOG) Topical Report WCAP-14572, Revision 1-NP-A (WCAP) (Reference 4), which was previously reviewed and approved by the NRC staff in a letter dated December 15, 1998.

Pursuant to 10 CFR 50.55a(a)(3)(i), the licensee has proposed to implement the RI-ISI program for ASME Class 1, 2, 3 and non-class piping systems as an alternative to the Code examination requirements for ASME Code, Section XI inservice inspection program of Code Class 1 and 2 piping, Categories B-F, B-J, C-F-1 and C-F-2 welds for piping system for the Palisades Nuclear Plant. The licensee requested approval of this alternative for implementation during the second inspection period of the third 10-year ISI interval.

3.0 EVALUATION

The staff reviewed the licensee's submittal with respect to the methodology and criteria contained in the NRC-approved WOG Topical Report WCAP-14572, Revision 1-NP-A. Further guidance in defining acceptable methods for implementing an RI-ISI program is also provided in RG 1.174 (Reference 5), RG 1.178 (Reference 6), and Standard Review Plan (SRP) Chapter 19 (Reference 7), and SRP Chapter 3.9.8 (Reference 8).

During the review of licensee's March 1, 2002, submittal, the NRC staff noted that the Palisades Expert Panel re-categorized 27 of 91 segments that were originally placed into the high-safety-significance (HSS) category based on the quantitative analysis results into the low-safety-significant (LSS) category based on their engineering judgement. As discussed in Section 3.4 of this safety evaluation, the Expert Panel may re-categorize segments provided that sufficient justification is documented as part of the re-categorization. In order to clarify this issue, the NRC conducted an on-site audit of the documentation supporting the RI-ISI relief request on September 12-13, 2002.

3.1 Proposed Changes to the ISI Program

The number and locations of inspection based on the ASME guidelines will be replaced by the number and locations of inspection based on the RI-ISI guidelines. As illustrated in Table 5-1 of Reference 1, current ASME Code, Section XI selects a total of 509 non-destructive exams while the proposed RI-ISI program selects a total of 116 non-destructive exams. Reference 3 stated that an additional 48 exams were added to the RI-ISI program for a total of 164 non-destructive examinations. The surface examinations required under the Code will be discontinued while system pressure tests and VT-2 visual examinations shall continue. These results are consistent with the concept that, by focusing on inspections of high safety significant welds in the presence of an active degradation mechanism, the number of inspections can be reduced while maintaining adequate protection of public health and safety.

The ASME Code, Section XI guidelines requiring a fixed number of additional examinations after finding an unacceptable flaw will be modified. In the RI-ISI program, an evaluation shall be made to determine whether other elements are subject to the same root cause and degradation mechanism. Additional examinations shall be performed on these elements up to a number equivalent to the number of elements initially required to be inspected on the segment or segments. If unacceptable flaws or relevant conditions are again found similar to the initial flaw or relevant condition, the remaining elements identified as susceptible shall be examined. No additional examination, however, needs to be performed if there is no other element identified to be susceptible to the same root cause or degradation mechanism. Sample expansions and the disposition of flaw shall be conducted within the outage when the flaw or the relevant condition was identified.

The implementation of an RI-ISI program for piping would ideally be initiated at the start of a plant’s 10-year ISI interval consistent with the requirements of the ASME Code, Section XI, Edition and Addenda committed to by the Owner in accordance with 10 CFR 50.55a. However, the implementation may begin at any inspection period during a 10-year inspection interval as long as the examinations are scheduled and distributed to be consistent with the Code requirements, e.g., the minimum percentage of examinations completed at the end of each of the three inspection periods under ASME Code, Section XI, Program B should be 16 percent, 50 percent, and 100 percent, respectively, and the maximum examinations credited at the end of the respective periods should be 34 percent, 67 percent, and 100 percent. In the current third 10-year inspection interval, Palisades has completed approximately one third of the Class 1 and 2 piping weld examinations required under the existing ASME Code, Section XI ISI Program. This, coupled with Palisades commitment to examine at least 66 percent of the examinations associated with the RI-ISI Program by the end of the third interval, constitutes an alternative examination program which will provide an acceptable level of quality and safety. However, the ASME Code, Section XI requires that a minimum of 50 percent of the required welds be examined by the end of the second period of the interval. Since the licensee’s RI-ISI program plan was under review by NRC, complying to ASME Code, Section XI period requirements would have required NMC to perform additional examinations not currently scheduled for the March 2003 refueling outage. Performing examinations in accordance with the current ASME Code, Section XI inspection plan would have resulted in unnecessary personnel radiation exposure. The NRC staff determined that performance of Class 1 and 2 piping weld examinations during the current refueling outage (March 2003) to meet the Code minimum percentage of examinations for the second inspection period, would result in a hardship or unusual difficulty without a compensating increase in the level of quality and safety. By letter dated March 20, 2003, the NRC authorized the licensee's proposed alternative pursuant to 10 CFR 50.55a(a)(3)(ii), for a deferral of 2 years from February 13, 2003, or through the remainder of the third ISI interval, whichever is sooner, to comply with the Class 1 and 2 piping weld examination requirements of the 1989 Edition, ASME Code, Section XI, for the third 10-year ISI interval. This authorization, however, did not apply to any augmented examination requirements. The licensee has stated that scheduled examinations under the RI-ISI program would examine 66 percent of the required remaining locations by the end of the third period (conclusion of third inspection interval).

3.2 Engineering Analysis

In accordance with the guidance provided in RGs 1.174 and 1.178 (Refs. 5 and 6), the licensee provided the results of an engineering analysis of the proposed changes, using a combination of traditional engineering analysis and probabilistic risk assessment (PRA). The licensee stated that the results of the engineering analysis demonstrate that the proposed changes are consistent with the principle of defense-in-depth. This is accomplished by evaluating a location's susceptibility to each potential degradation mechanism that may be a precursor to leak or rupture and then performing an independent assessment of the consequence of a failure at that location. No changes to the evaluation of design-basis accidents in the final safety analysis report are being made by the RI-ISI process. Therefore, sufficient safety margins will be maintained.

In the licensee's proposed RI-ISI program, piping failure potential estimates were determined using a software program contained in Supplement 1 to Reference 4, entitled "Westinghouse Structural Reliability and Risk Assessment (SRRA) Model for Piping Risk-Informed Inservice Inspection," which utilizes probabilistic fracture mechanics technology, industry piping failure history, plant-specific piping failure history, and other relevant information. Using the failure potential and supporting insights on piping failure consequences from the licensee's PRA, safety significance ranking of piping segments was established to determine inspection locations. The program maintains the fundamental requirements of the Code, such as the examination technique, frequency, and acceptance criteria. The RI-ISI program is intended to reduce the number of required examination locations significantly while maintaining an acceptable level of quality and safety.

The licensee stated that the applicable aspects of the ASME Code not affected by the proposed alternative RI-ISI program and the ongoing augmented inspection programs would be retained. This is consistent with the approved WCAP and, therefore, it is acceptable.

3.3 Probabilistic Risk Assessment

The licensee used the PRA version PSAR1A, dated April 2000, to support this RI-ISI submittal. The base core damage frequency (CDF) from PSAR1A is 5.5E-5/yr and the base large early release frequency (LERF) is 3.9E-8/yr.

The NRC safety evaluation report (SER) on the licensee's individual plant examination (IPE) identified four limitations in the human reliability analyses. Three of the limitations are related to the IPE's application of the Technique for Human Error Rate Prediction methodology (Ref. 9). In Reference 2, the licensee stated that, except for pre-initiator errors, the PRA was updated and now used the accident sequence evaluation program (ASEP) methodology (Ref. 10) human reliability analysis. The ASEP methodology includes plant-specific factors influencing human performance and evaluates both diagnosis and post diagnosis tasks. Although maintaining the original pre-initiator error analysis may limit the use of the PRA for risk-informed applications related to these events, pre-initiator errors are generally negligible contributors to system and functional failures probabilities and changes in these error probabilities are not expected to influence the results used in the RI-ISI evaluations. The fourth limitation regarding excessively low screening values has also been addressed by increasing the screening values to be consistent with the ASEP methodology.

The licensee stated that the PSAR1A has been extensively reviewed, including a detailed self-assessment in April 2000, and a Combustion Engineering Owners Group (CEOG) peer review in May 2000. The CEOG peer review identified two areas as inadequate; incomplete documentation of various aspects of the PRA and failure to perform a through dependency analysis for the human actions modeled in the PRA. Regarding the documentation, the peer review team primarily noted that the independent technical review and approvals were not completed for the supporting engineering analyses. Incomplete documentation is, in and of itself, an administrative weakness and the licensee reported in Reference 1 that the documentation of the required analyses has been completed.

The licensee performed a dependency analysis for the human actions modeled in the PRA using a multi-step iterative process. The process (Ref. 2) initially assigns conservative probabilities (i.e., higher than expected failure probabilities) to human actions and combinations of human actions. Beginning with the largest contributors to CDF, these conservative probabilities are replaced with probabilities derived from the systematic analysis of the specific combinations of actions credited in each accident sequence. The dependency analysis was continued until the change in CDF caused by changing additional human error probabilities became negligible. The licensee reported that this process yielded a relatively small increase in the CDF indicating that the dependencies between human actions within important accident sequences are low and therefore the original PRA model used to support the RI-ISI relief request (PSAR1A) adequately accounted for significant dependencies between the human actions. The NRC staff concurs that a relatively small change in CDF is sufficient indication that the RI-ISI program would be minimally, if at all, affected by a re-analysis using the updated PRA.

The NRC staff did not review the PRA analysis to assess the accuracy of the quantitative estimates. Quantitative results of the PRA are used, in combination with a quantitative characterization of the pipe segment failure likelihood, to support the assignment of segments into broad safety significance categories reflecting the relative importance of pipe segment failures on CDF and LERF and to provide an illustrative estimate of the change in risk. Inaccuracies in the models or assumptions large enough to invalidate the analyses developed to support RI-ISI should have been identified in the licensee's or the NRC's reviews. Minor errors or inappropriate assumptions will only affect the consequence categorization of a few segments and will not invalidate the general results or conclusions. The NRC staff finds that the quality of the Palisades PRAR1A PRA is sufficient to support this submittal.

The licensee stated in Reference 1 that the change in risk calculations was performed according to the guidance provided on page 213 of WCAP. Table 3.10-1 of Reference 1 presents a comparison of CDF and LERF for current ASME Code, Section XI ISI program and RI-ISI program. In response to NRC concerns, the licensee subsequently modified the proposed RI-ISI program by redefining multiple pipe size segments into multiple segments and by reducing the number of HSS segments that the Expert Panel reduced to LSS. Reference 3 includes a modified Table 3.10-1 that reflects the changes in the RI-ISI program. Inspection of the modified table indicates that the system level and the aggregate change in risk caused by replacing the current ASME Code, Section XI program with the proposed RI-ISI program satisfies the change in risk criteria specified in the WCAP-14572, Rev. 1-NP-A. Based on the use of the approved methodology as modified by the licensee and discussed in this SE, and on the reported results, the NRC staff finds that any change in risk associated with the implementation of the RI-ISI program is small and consistent with the intent of the Commission's Policy Statement and with RG 1.178.

3.4 Integrated Decisionmaking

The proposed RI-ISI program presents an integrated approach that considers in concert the traditional engineering analysis, the risk evaluation, and the implementation and the performance monitoring of piping. The selection of pipe segments to be inspected is described in References 1, 2, and 3 using the results of the risk category rankings and other operational considerations. Table 3.7-1 (Ref. 1) identified the number of segments in the different systems that met or exceeded the quantitative criteria to be assigned HSS, and the number of segments whose categorization was changed by the Expert Panel. The table identifies 91 segments that were initially classified HSS according to the quantitative guidelines. Inspection of the table indicates that the Expert Panel reclassified 27 initially HSS segments as LSS and 66 initially LSS segments as HSS.

The WCAP methodology provides the following guidance on the reclassification of the safety-significance of segments by the Expert Panel.

> "The expert panel (such as the expert panel used for the Maintenance Rule) evaluates the risk-informed results and makes the final decision be identifying the high-safety-significant pipe segments for ISI. The piping segments that have been determined by quantitative methods to be high safety significant should not be classified lower by the expert panel without sufficient justification that is documented as part of the program. The expert panel should be focused primarily on adding piping to the higher classification."

The Expert Panel classified 21 segments as LSS that would normally be HSS based on the "without operator action" RRWs (risk reduction worth) exceeding the selection guidelines. RG 1.178 (Ref. 6) states that detailed guidelines should be developed and used by the Expert

Panel to support the final determination of the safety-significant classification. The SRP Chapter 3.9.8 (Ref. 8) states that the documentation to be provided to the Expert Panel should ensure that all applicable insights, key principles, and supporting elements were addressed and communicated to the final decision panel. Appendix B in SRP Chapter 19 (Ref. 7) further states that the license's decision-making process should be technically defensible and should be sufficiently detailed to allow an independent party to reproduce the major results.

The NRC audited the documentation supporting the RI-ISI submittal on September 12-13, 2002. There was little documentation supporting and justifying the Expert Panel's re-classification of those segments' justifications. One justification was, for example, "[t]he Ops [operations] representative on the expert panel expressed high confidence in the ability of the operators to correctly identify and take the listed recovery actions." The NRC requested that the licensee provide additional justification for each of the 21 segments including information on the method of identifying and mitigating the event, the time available to take the actions, and the time required to take the actions. In Reference 2 the licensee stated that rather than invest additional time required to prepare the documentation, 17 of the segments would be classified HSS. In Reference 2, the licensee also provided justification for classifying the final four segments as LSS. The four segments are part of the primary coolant letdown isolation system and are downstream of the letdown isolation valve but upstream of the letdown orifices. The small break loss-of-coolant accident initiated by a failure in these segments would be mitigated if the isolation valve is closed using controls in the control room. Isolation of the letdown system is a procedurally driven action very early in the event and the operations personnel are trained frequently for the actions. The NRC staff finds that the justification provided in Reference 2 allows an independent reviewer to reach the same conclusion that there is a very high probability that the operators would initiate proper action within the time that the action will be effective and is, therefore, acceptable.

The licensee also re-classified six segments as LSS based on considerations other than operator actions (e.g., conservative PRA assumptions, non-realistic SRRA inputs, etc.). The NRC requested that the licensee provide additional justification for the re-classification of these six segments. In Reference 2, the licensee stated that rather than invest additional time required to prepare the documentation, five of the segments would be classified HSS. The licensee identified the consequences of the final segment's rupture and explained that the original consequences (loss of two pumps) were inappropriately modeled and that the appropriate consequence (loss of one pump) would have placed the segment in the LSS category. The NRC staff finds that correction of an inappropriate assumption such as described in Reference 2 is a task assigned to the Expert Panel and that the documentation provided sufficient justification for the Expert Panel's decision.

The WCAP describes targeted examination volumes (typically associated with welds) and methods of examination based on the type(s) of degradation expected. The NRC staff has reviewed these guidelines and has determined that, if implemented as described, the RI-ISI examinations should result in improved discovery of service-related discontinuities over that currently provided by the Code.

The objective of ISI required by the Code is to identify conditions (i.e., flaw indications) that are precursors to leaks and ruptures in the pressure boundary that may impact plant safety. Therefore, the RI-ISI program must meet this objective to be found acceptable for use.

Further, since the RI-ISI program is based on inspection for cause, element selection should target specific degradation mechanisms.

Section 4 of the WCAP provides guidelines for the areas and/or volumes to be inspected as well as the examination method, acceptance standard, and evaluation standard for each degradation mechanism. Based on a review of the cited portion of the WCAP, the NRC staff concludes that the examination methods are appropriate since they are selected based on specific degradation mechanisms, pipe sizes, and materials of concern. The licensee reported no deviations in this area from the WCAP methodology and, therefore, its evaluation is acceptable.

3.5 Implementation and Monitoring

Implementation and performance monitoring strategies require careful consideration by the licensee and are addressed in Element 3 of RG 1.178 and SRP 3.9.8. The objective of Element 3 is to assess performance of the affected piping systems under the proposed RI-ISI program by implementing monitoring strategies that confirm the assumptions and analyses used in the development of the RI-ISI program. To approve an alternative pursuant to 10 CFR 50.55a(a)(3)(i), implementation of the RI-ISI program, including inspection scope, examination methods, and methods of evaluation of examination results, must provide an acceptable level of quality and safety.

In Reference 1, the licensee stated that upon approval of the RI-ISI program, procedures that comply with the WCAP guidelines will be prepared to implement and monitor the RI-ISI program. The licensee confirmed that the applicable portions of the Code not affected by the change, such as inspection methods, acceptance guidelines, pressure testing, corrective measures, documentation requirements, and quality control requirements would be retained.

The licensee stated in Section 4 of Reference 1 that the RI-ISI program is a living program and its implementation will require feedback of new relevant information to ensure the appropriate identification of HSS piping locations. Reference 1 also stated that as a minimum, risk ranking of piping segments will be reviewed and evaluated every ISI period and that significant changes may require more frequent adjustments as directed by any NRC Bulletin or Generic Letter or by industry and plant-specific feedback.

The proposed periodic reporting requirements meet existing ASME Code requirements and applicable regulations and, therefore, are considered acceptable. The NRC staff finds that the proposed process for RI-ISI program updates meets the guidelines of RG 1.174 that risk-informed applications should include performance monitoring and feedback provisions; therefore, the process for program updates is acceptable.

3.6 Deviations from the WCAP Methodology

The licensee identified three deviations from the approved WCAP methodology. One deviation is to perform visual VT-2 examinations as an alternative to volumetric exams for a defined population of welds. The second deviation involves crediting leak detection for some pipe segments that are not reactor coolant system (RCS) piping segments. The third deviation involves determining the number of inspections for some piping segments based on the ASME percentage criteria instead of the statistical sampling methodology. In addition, a fourth deviation involving the method of calculating the failure frequency of segments containing piping with multiple sizes (i.e., different diameters) was identified by the NRC during an on-site audit September 12-13, 2002. These deviations are discussed below.

In the first deviation, NMC proposed to perform visual VT-2 examinations during each refueling outage as an alternative to the volumetric examinations in the WCAP, for ASME Code Class 1 and 2 socket welds in piping of 2-inch diameter and under that are identified as HSS in the RI-ISI program. This alternative is acceptable because the volumetric examination is inconclusive due to the geometric limitations imposed by a socket weld. However, the NRC staff notes that Table IWB-2500-1 and Table IWC-2500-1 of the Code require surface examination, not volumetric examination, of the socket welds, and surface examination (i.e., liquid penetration examination) is an effective method for discovery of potential surface flaws on the outside surface, and specifically, flaws induced by low-cycle, high-bending stress thermal fatigue or by external chloride stress corrosion cracking. The NRC staff believes that such flaws are prevented in Code Class 1 and 2 socket weld piping through piping design, selection and control of piping materials, control of welding processes and cleanliness requirements within the plant. Therefore, these conditions do not exist in the Code Class 1 and 2 piping at Palisades. As for a potential outside surface flaw caused by vibration-induced fatigue, such a flaw is likely to take a long period for initiation. After the initiation phase, the flaw will likely propagate rapidly and cause the pipe to leak which can be detected early with no serious consequence. Therefore, the proposed alternative to conduct VT-2 visual examination each outage in lieu of the Code-required examinations for Categories B-J, C-F-1, and C-F-2 socket welds is acceptable.

The second deviation reported by the licensee involves credit taken for leak detection when calculating pipe failure probabilities. The WCAP allows credit for detecting (and isolating, repairing, or otherwise terminating a potential accident sequence) a leak in the RCS piping before it develops into a pipe break for piping inside of containment. This credit reflects the highly developed leak detection systems used to monitor leakage from the reactor coolant piping. In Reference 1, the licensee states that detection of a leak-before-break is plausible for any non-RCS segment located inside the containment that interfaces with the RCS. Leaks in these segments can be detected with the radiation and sump level monitors as reliably as an RCS leak. Because the segments are subject to essentially the same leak detection capabilities as that of an RCS leak, the extension of credit for leak detection in these segments is reasonable and acceptable.

The third deviation from the approved WCAP methodology involves the determination of the number of welds for inspection in four segments of thin-walled stainless steel piping. In Reference 2, the licensee identified four segments (SSS-001, SSS-002, SSS-002C, and SSS-007) in safety injection/refueling water tank and containment sump suction piping where the selection of the number of locations for inspection deviated from the use of the Westinghouse statistical (Perdue) model as described in the WCAP. These segments are composed of thin-walled austenitic stainless steel. One weld had no construction code radiograph although volumetric and surface examinations were performed on the remaining segments in accordance with the ASME Code, Section XI. The licensee stated that application of the Perdue model to these segments yields an estimated probability of an unacceptable flaw (a 10 percent through-wall crack) of almost unity at the plants present age. This would require 100 percent of the welds to be inspected to meet the 95 percent confidence limit in regard to the target leak rate. Instead, the licensee proposes to select a 7.5 percent sample from each of these segments.

The thin-walled austenitic stainless steel piping segments operate at temperatures below 150 degrees Fahrenheit and pressures less than 100 psi with no known active degradation mechanism. The licensee stated that the criterion for use of the statistical method with existence of an unacceptable flaw at the current age of the plant defeats the statistical method of determining the number of examination locations for these thin-walled piping. The NRC staff has taken into consideration the material type, the fracture toughness, pressure/temperature rating, results of a previous inservice examinations, and existing degradation mechanism to evaluate the licensee's examination locations. The NRC staff accepts the licensee's rationale in regard to a 7.5 percent sample inspection in accordance with the guidelines of the ASME Code, Section XI, of the number of examination locations for these segments.

The fourth deviation involved the method of calculating the failure frequency of segments containing piping with multiple sizes (i.e., different diameters). The licensee reported no deviations from the approved methodology for estimating the segment failure frequency. However, during a September 12-13, 2002, on-site audit of the documentation of the licensee's RI-ISI program development, the NRC staff noted that some piping segments had more than one SRRA failure frequency estimate. The failure frequency calculated by the SRRA computer code is calculated for an individual weld, i.e., a specific weld geometry, material properties, and environmental conditions. The WCAP methodology develops and uses a single-failure frequency estimate to characterize each pipe segment's likelihood of failure regardless of the number of welds within the segment. The NRC staff's approval of the use of a single segment failure frequency independent of the number of welds was based on Westinghouse's proposal that the failure frequencies are obtained from the SRRA code are calculated by inputting the conditions (typically the most limited or bounding) for the entire piping segment. Essentially, the piping failure probability is a representation or characterization of the material properties and environment in the piping segment. Consequently, multiple failure frequency estimates for a single pipe segment indicated a deviation from the methodology.

In Reference 3, the licensee stated that some segments included piping of multiple pipe sizes. The licensee created sub-segments based on pipe size to facilitate estimating the failure frequency using the SRRA code. Failure frequency estimates for segments made up of multiple pipe sizes were determined by performing multiple SRRA cases, one SRRA case for each sub-segment. The most limiting inputs, based on the expected degradation mechanism(s) for the individual sub-segment, were developed for each SRRA case in accordance with the guidance in the WCAP. The highest sub-segment failure probability was used to represent the segment failure probability for risk ranking and change in risk purposes. The licensee's proposed method only combines limiting inputs for each sub-segment and not for the entire segment and is, therefore, a deviation from the approved methodology.

The WCAP methodology allows, but does not require, multiple sized piping within a segment. Although emphasis is placed on defining piping segments as lengths of piping that have the same consequences caused by pipe failure, pipe size is one of the four criteria that can be used to define segments. The NRC staff has determined that there are two alternative methods to incorporate multiple pipe size segments into the analysis that comport with the approved methodology. One method would involve combining the most limiting inputs in the entire segment into a single weld and use the estimated failure frequency of that weld to represent the segment. The other method would be to divide the segment into new segments, each with similar or the same size.

In Reference 3, the licensee proposed to modify the original methodology for estimating the segment failure frequencies. The licensee proposed that all multiple pipe size HSS and LSS segments be divided into multiple new segments, each with the same pipe size. The failure frequency of the multiple pipe size segment (representing the highest failure frequency of all the sub-segments) is assigned to each of the new segments. Multiple pipe size segments that are HSS segments will become multiple HSS segments and inspection locations determined in accordance with WCAP guidelines for inspecting HSS segments. Multiple pipe size LSS segments will become multiple segments for the change in risk estimates, with each segment contributing the same increase in risk as the highest sub-segment. Multiple pipe size HSS segments were not treated as multiple segments in the change in risk calculations because assigning the highest failure frequency to all the original sub-segments could overestimate any risk decrease from initiating inspection within these segments. The licensee reported that this modification resulted in supplemental inspection in new HSS segments, but no supplemental inspections were required to satisfy the change in risk guidelines.

The modification proposed by the licensee in Reference 3 is more conservative than estimating the failure frequency for each of the new segments. The licensee’s original methodology assigned the highest sub-segment failure frequency to the segment. Therefore, the failure frequency of each of the new segments will be the same as or less than the failure frequency of the highest sub-segment. In the categorization, some new HSS segments (previously sub-segments) may become LSS segments because the failure frequency of the new segments may be low enough to bring the risk reduction worth below the guideline value for HSS. No new LSS segments would become HSS segments because none would have a higher risk than the original multiple size segment.

The impact on the change in risk calculations is more complex. Assigning the multiple size failure frequency to each new LSS segment will overestimate the risk increase compared to estimating and using the failure frequency for each new segment. Similarly, assigning the multiple size failure frequency to each new HSS segment would overestimate the risk decrease compared to estimating and using the failure frequency for each new segment. Overestimating the risk increase is conservative while overestimating the risk decrease is non-conservative. The licensee’s use of multiple new LSS segments in the change in risk calculations overestimates the risk increase. The licensee’s use of the original multiple pipe size HSS segments in the change in risk calculations does not overestimate the risk decrease.

Based on the preceding evaluation, the NRC staff finds that the estimation and use of pipe failure frequencies as described in Reference 3 is acceptable because it yields more HSS segments and a higher estimated increase in risk compared to application of a methodology that fully comports with the WCAP.

4.0 CONCLUSION

10 CFR 50.55a(a)(3)(i) permits alternatives to regulatory requirements when authorized by the NRC if the applicant demonstrates that the alternative provides an acceptable level of quality and safety. In this case, the licensee's proposed alternative is based on the risk-informed selection process described in the NRC-approved Topical Report WCAP-14572, Rev. 1-NP-A.

The licensee identified three deviations from the approved WCAP methodology. One deviation is to perform visual VT-2 examinations as an alternative to volumetric exams for a defined population of welds. The second deviation involves crediting leak detection for some pipe segments that are not RCS piping segments. The third deviation involves determining the number of inspections for some piping segments based on the ASME percentage criteria instead of the statistical sampling methodology. The NRC identified a fourth deviation involving the method of calculating the failure frequency of segments containing piping with multiple sizes (i.e., different diameters) during an on-site audit on September 12-13, 2002. As discussed in Section 3.6, the NRC staff approved the three deviations identified by the licensee without modification. The NRC staff approved the fourth deviation after the licensee modified the methodology as described in Reference 3.

The Palisades risk-informed methodology provides for conducting an analysis of the proposed changes using a combination of engineering analysis with supporting insights from a PRA. Defense-in-depth and quality are not degraded in that the methodology provides reasonable confidence that any reduction in existing inspections will not lead to degraded piping performance when compared to existing performance levels. Inspections are focused on locations that are classified as HSS with active degradation mechanisms.

The Palisades methodology also considers implementation and performance monitoring strategies. Inspection strategies ensure that failure mechanisms of concern have been addressed and there is adequate assurance of detecting damage before structural integrity is affected. The risk significance of piping segments is taken into account in defining the inspection scope for the RI-ISI program.

The NRC staff finds that the results of different elements of the engineering analysis are considered in an integrated decision-making process. The impact of the proposed changes in the ISI program is founded on the adequacy of the engineering analysis and acceptable estimation of changes in plant risk in accordance with RG 1.174 and RG 1.178 guidelines.

System pressure tests and visual examination of piping structural elements shall continue to be performed on all Code Class 1and 2 systems in accordance with the ASME Code Section XI program. The RI-ISI program applies the same performance measurement strategies as the existing ASME Code requirements.

The NRC's review of the licensee's proposed RI-ISI program concludes that the program is an acceptable alternative to the current ISI program, which is based on ASME Code, Section XI requirements for Code Class 1, Categories B-F and B-J welds and for Code Class 2, Categories C-F-1 and C-F-2 welds. Therefore, the licensee's proposed RI-ISI program is authorized for the third 10-year ISI interval pursuant to 10 CFR 50.55a(a)(3)(i) on the basis that the request provides an acceptable level of quality and safety. All other ASME Code, Section XI requirements for which relief was not specifically requested and approved remain applicable, including third party review by the Authorized Nuclear Inservice Inspector.

5.0 REFERENCES

1. Letter, dated March 1, 2002, Paul A. Harden (Director, Engineering, Nuclear Management Company, LLC) to U. S. Nuclear Regulatory Commission, containing Palisades Nuclear Station - Request for Approval of Relief Request, “Alternate ASME Code, Section XI, Risk-informed Inservice Inspection Program”.

2. Letter, dated August 8, 2002, Paul A. Harden (Director, Engineering, Nuclear Management Company, LLC) to U. S. Nuclear Regulatory Commission, containing Palisades Nuclear Station - Response to Request for Additional Information.

3. Letter, dated February 28, 2003, Douglas E. Cooper (Site Vice-President, Palisades Nuclear Plant) to U. S. Nuclear Regulatory Commission, containing Supplement to Palisades Nuclear Plant Risk-Informed Inservice Inspection Piping Program.

4. WCAP-14572, Revision 1-NP-A, Westinghouse Owners Group Application of Risk-Informed Methods to Piping Inservice Inspection Topical Report, February 1999.

5. NRC Regulatory Guide 1.174, An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis, November 2002.

6. NRC Regulatory Guide 1.178, An Approach for Plant-Specific Risk-Informed Decision Making: Inservice Inspection of Piping, July 1998.

7. NRC NUREG-0800, Chapter 19, Use of Probabilistic Risk Assessment in Plant-Specific, Risk-Informed Decisionmaking: General Guidance, November 2002.

8. NRC NUREG-0800, Chapter 3.9.8, Standard Review Plan for Trial Use for the Review of Risk-Informed Inservice Inspection of Piping, May 1998.

9. NRC NUREG/CR-1278, “Handbook for Human Reliability Analysis with Emphasis on Nuclear Power Plant Applications, August 1983 (THERP).

10. NRC NUREG/CR-4772, “Accident Sequence Evaluation Program (ASEP) Human Reliability Analysis Procedure,” February, 1987.

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