NRC Staff Position on Inspection of Socket Welds

Attached are four documents that provide the staff's position on acceptable methods for inspecting socket welds.

- Internal Memorandum, "Safety Evaluation of Risk-Informed ISI Program, Relief Request RR-ENG-2-16 for the Second Interval for South Texas Project, Units 1 and 2," June 16, 2000.

- Letter from T. J. Jordan, South Texas Project, "Response to Request for Additional Information Regarding Risk-Informed Inservice Inspection Application for Section XI Examination Requirements for Class 1 and 2 Piping Welds (RR-ENG-2-23)," January 10, 2002.

- Letter from the NRC to Nuclear Management Company, LLC, "Palisades Plant — Risk-informed Inservice Inspection Program," dated May 19, 2003.

- Letter from the NRC to Virginia Electric and Power Company, "Surry Power Station, Unit 2 — American Society of Mechanical Engineers Section XI Risk-informed Inservice Inspection Interval Update for the Fourth 10-Year Interval," August 8, 2005.

The NRC staff determined that the licencee's proposed alternative to perform visual VT-2 examinations of HSS socket welds and their associated branch connections, NPS 2 and smaller, in lieu of the required volumetric examinations provides reasonable assurance of structural integrity of the subject piping welds. In addition, complying with the specific requirement would result in a hardship or unusual difficulty without a compensating increase in the level of quality and safety.

August 8, 2005

Mr. David A. Christian
Sr. Vice President and Chief Nuclear Officer
Virginia Electric and Power Company
Innsbrook Technical Center
5000 Dominion Blvd.
Glen Allen, Virginia 23060-6711

SUBJECT: SURRY POWER STATION, UNIT 2 - AMERICAN SOCIETY OF MECHANICAL ENGINEERS SECTION XI RISK-INFORMED INSERVICE INSPECTION INTERVAL UPDATE FOR THE FOURTH 10-YEAR INTERVAL (TAC NO. MC3134)

Dear Mr. Christian:

By letter dated May 13, 2004, as supplemented by letter dated March 23, 2005, Virginia Electric and Power Company (VEPCO) requested approval to use its risk-informed inservice inspection (RI-ISI) program for the fourth 10-year ISI interval and approval of Relief Request R-1. The licensee, in Relief Request R-1, seeks relief from certain requirements of its RI-ISI program, as required by WCAP-14572, Revision 1-NP-A, "Westinghouse Owners Group Application of Risk-Informed Decision Making: Inservice Inspection of Piping," pertaining to high safety significant (HSS) socket weld examinations. The Nuclear Regulatory Commission (NRC) staff has completed its review of these relief requests, and the NRC staff's evaluations and conclusions are contained in the enclosed Safety Evaluation.

For the implementation of the RI-ISI program, the NRC staff concludes that VEPCO's proposed RI-ISI program provides an acceptable level of quality and safety and, therefore, is authorized pursuant to Title 10 of the *Code of Federal Regulations* Section 50.55a(a)(3)(i) for the fourth 10-year ISI interval at Surry, Unit 2.

Regarding Relief Request R-1, the NRC staff concludes that VEPCO's proposal to perform visual VT-2 examinations during each refueling outage in lieu of the volumetric examinations specified WCAP-14572, Revision 1-NP-A, for the subject HSS socket welds and their associate branch connections nominal pipe size 2 and smaller provides reasonable assurance of structural integrity of the subject pipe welds. In addition, the NRC staff has determined that complying with the specified requirement would result in a hardship or unusual difficulty without a compensating increase in the level of quality and safety. Therefore, the NRC staff authorizes the proposed alternative pursuant to 10 CFR 50.55a(a)(3)(ii) for the fourth 10-year ISI interval at Surry, Unit 2.

The NRC staff is closing out TAC No. MC3134 with this letter.

Sincerely,

/RA/

Evangelos Marinos, Chief, Section 1
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-281

Enclosure: As stated

cc w/encl: See next page

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO THE FOURTH 10-YEAR INTERVAL INSERVICE INSPECTION PROGRAM

SURRY POWER STATION, UNIT 2

VIRGINIA ELECTRIC AND POWER COMPANY

DOCKET NO. 50-281

1.0 INTRODUCTION

By letter dated May 13, 2004 (Reference 1), as supplemented by a letter dated March 23, 2005 (Reference 2), Virginia Electric and Power Company (the licensee) submitted a request to extend the risk-informed inservice inspection (RI-ISI) program plan from the third to the fourth 10-year inservice inspection (ISI) interval and a request to implement Relief Request R-1. Both relief requests were for Surry Power Station, Unit 2.

In its letter dated April 27, 2000 (Reference 3), as supplemented by letter dated September 27, 2000 (Reference 4), the licensee submitted the RI-ISI program for the third 10-year ISI interval at Surry, Unit 2. The licensee's RI-ISI program was developed in accordance with the methodology contained in the Westinghouse Owners Group (WOG) Report WCAP-14572, Revision 1-NP-A, "Westinghouse Owners Group Application of Risk-Informed Decision Making: Inservice Inspection of Piping" (Reference 6), which was reviewed and approved by the U.S. Nuclear Regulatory Commission (NRC) staff. Subsequently, by letter dated January 26, 2001 (Reference 5), the NRC staff approved the licensee's RI-ISI program for use in the third 10-year ISI interval at Surry, Unit 2. On June 13, 2002 (Reference 9), the licensee sought relief (Relief Request R-1) from its approved third 10-year RI-ISI program regarding examination of high safety significant (HSS) socket welds. The RI-ISI program Relief Request R-1 was approved by the NRC staff for the third 10-year ISI interval at Surry, Unit 2 on September 23, 2003 (Reference 10).

The licensee proposes to extend the same RI-ISI program, as submitted on April 27, 2000, for the third 10-year ISI interval, to the fourth 10-year ISI interval at Surry, Unit 2. In addition, the licensee requests approval of relief similar to R-1 for the fourth 10-year ISI interval at Surry, Unit 2.

2.0 REGULATORY REQUIREMENTS

Title 10 of the *Code of Federal Regulations* (10 CFR), Section 50.55a(g) specifies that ISI of nuclear power plant components shall be performed in accordance with the requirements of the American Society of Mechanical Engineers (ASME), Boiler and Pressure Vessel Code (Code), Section XI, except where specific written relief has been granted by the Commission pursuant to 10 CFR 50.55a(g)(6)(i). As stated, in part, in 10 CFR 50.55a(a)(3) 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.

The Surry, Unit 2 RI-ISI program is an alternative pursuant to 10 CFR 50.55a(a)(3)(i). In its letter dated May 13, 2004, the licensee requested NRC staff authorization to utilize the same RI-ISI program, previously approved for use in the third interval, for use in the fourth ISI interval at Surry, Unit 2. The licensee also requested relief R-1 from the WOG methodology pursuant to 10 CFR 50.55a(a)(3)(ii) regarding the examination of socket welds and their branch connections in piping nominal pipe size (NPS) 2 inches and less. The scope of the RI-ISI program is limited to the inspection of ASME Code Class 1 piping (Categories B-F and B-J welds). The applicable version of the Code for the fourth 10-year ISI interval at Surry, Unit 2 is the 1998 Edition through the 2000 Addenda.

3.0 TECHNICAL EVALUATION FOR RI-ISI PROGRAM

Summary of Proposed Changes

By letter dated May 13, 2004, as supplemented by a letter dated March 23, 2005, the licensee submitted a relief request pursuant to 10 CFR 50.55a(a)(3)(i). The licensee sought relief from the requirements of ASME Code, Section XI to utilize an RI-ISI program plan at Surry, Unit 2 to perform ISI inspections during the fourth 10-year ISI interval. An RI-ISI program was reviewed and approved by the NRC staff for the third 10-year ISI interval. In its submittal dated May 13, 2004, the licensee requested approval to utilize the same RI-ISI program for the fourth 10-year ISI interval.

NRC Staff Evaluation

An acceptable RI-ISI program plan is expected to meet the following five key principles discussed in Regulatory Guide (RG) 1.178 (Reference 7), Standard Review Plan 3.9.8 (Reference 8), and the Electric Power Research Institute (EPRI) TR-112657.

1. The proposed change meets the current regulations unless it is explicitly related to a requested exemption or rule change.

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 core damage frequency (CDF) or risk, the increases should be small and consistent with the intent of the Commission's Safety Goal Policy Statement.

5. The impact of the proposed change should be monitored by using performance measurement strategies.

The first principle is met in this relief request because an alternative ISI program may be authorized pursuant to 10 CFR 50.55a(3)(i) and, therefore, an exemption request is not required. The second and third principles require assurance that the alternative program is consistent with the defense-in-depth philosophy and that sufficient safety margins are maintained, respectively. Assurance that the second and third principles are met is based on the application of the approved methodology and not on the particular inspection locations selected.

The methodology used to develop the fourth 10-year RI-ISI interval program is the WCAP-14572, Revision 1-NP-A methodology. However, in its letter dated March 23, 2005, the licensee stated that the Surry, Unit 2 RI-ISI program deviated from the approved WCAP-14572 methodology.

WCAP-14572, Revision 1-NP-A requires assigning all degradation mechanisms present in a segment to a single weld, resulting in the most limiting failure frequency for the entire segment. The licensee stated that the Surry, Unit 2 RI-ISI program deviated from the WOG methodology by including 12 multiple pipe size segments in which the most limiting failure frequency was not used for the segment. To resolve this deviation, the licensee reevaluated these segments based on the most limiting failure frequency and conducted a sensitivity study to determine the impact on the risk ranking of the segments.

In its reevaluation, the licensee found that the failure probabilities for two segments increased by about two orders of magnitude, causing these segments to be quantitatively ranked HSS from medium safety significant (MSS). The licensee stated that the increase occurred from the failure probability being based on a socket weld rather than a butt weld as evaluated in the original RI-ISI program. Although these segments were quantitatively classified as MSS in the original evaluation for the third 10-year RI-ISI program, the expert panel had classified these segments as HSS due to a high stress concern and water solid pressurizer concern. Since the reevaluation quantitatively classifies these segments as HSS and they remain in the HSS category, the licensee stated that the categorization of these two segments remains unchanged.

The licensee also reported that, in its reevaluation, the risk reduction worth (RRW) for a segment increased very slightly to just inside the MSS range at 1.001 from a low safety significant (LSS) value. The expert panel unanimously voted to retain the segment as LSS as permitted by the approved methodology. Therefore, the reevaluation did not affect the categorization of this segment.

The risk ranking due to the reevaluation to resolve the deviation from the WOG methodology caused three segments, previously categorized as LSS, to become MSS. The expert panel had previously classified these segments as HSS due to a concern with the single check valve separating these segments from the HSS reactor coolant system (RCS) segments. The expert panel continued to classify these segments as HSS, and they remain in the HSS category. Therefore, the reevaluation did not affect the categorization of these three segments.

The licensee compared the number and location of ISI inspections developed using its methodology to the number and location of inspections resulting from reevaluation without the deviation. The reevaluation did not result in any changes in the number or location of ISI inspections. The NRC staff concludes that the number of welds selected from inspection under the licensee's RI-ISI program is consistent with those produced by the application of WCAP-14572, Revision 1-NP-A. Moreover, the licensee's reevaluation of its RI-ISI program resolved the deviation from the approved WOG methodology by using the most limiting failure frequency for a segment. Any RI-ISI program that deviates from an approved methodology will require prior NRC staff review and approval of a request for relief.

The licensee stated in its letter dated May 13, 2004, that the methodology used to develop the fourth 10-year RI-ISI interval program is the approved WCAP-14572, Revision 1-NP-A. Additionally, in its letter dated March 23, 2005, the licensee stated that the RI-ISI program for the fourth 10-year RI-ISI interval program was reevaluated to resolve a deviation in the third 10-year RI-ISI interval program from the WCAP-14572 methodology. The fourth 10-year RI-ISI interval program is consistent with a program that follows the approved WCAP-14572, Revision 1-NP-A, methodology. Therefore, the second principle (the proposed change is consistent with the defense-in-depth philosophy) and the third principle (the proposed change maintains sufficient safety margins) are met.

The fourth principle (any increase in CDF or risk should be small and consistent with the intent of the Commission's Safety Goal Policy Statement) requires an estimate of the change in risk, and the change in risk is dependent on the location of inspections in the proposed ISI program compared to the location of inspections that would be inspected using the requirements of ASME, Section XI. The licensee stated that, consistent with WCAP-14572, Revision 1-NP-A methodology, new information was incorporated into its RI-ISI analysis as part of its "living program." This new information included changes to the probabilistic risk assessment (PRA) model and deterministic information provided to the expert panel. By letter dated May 13, 2004, the licensee stated that re-analysis of the PRA resulted in one segment's RRW to be lowered from HSS to MSS. However, the expert panel retained the segment within the HSS category. The licensee stated that the number and location of the volumetric examinations between the intervals remains the same for Surry, Unit 2.

Relief was granted in January 26, 2001, from selected requirements in the 1989 edition of the ASME Code, Section XI, which was the licensee's Code of record when relief was requested. The licensee stated in its letter dated May 13, 2004, that the Surry, Unit 2 current Code of record is the ASME Section XI 1998 Edition, through the 2000 Addenda. Although the Code of record edition has changed, the accuracy of the change in risk calculations does not warrant recalculating the ASME Section XI risk analysis with the current Code of record. Therefore, the NRC staff finds the comparison of the risk estimate between the current Code of record and the Code of record from which relief was granted in January 26, 2001, appropriate and acceptable. On May 13, 2004, the licensee reported that the change in risk analysis was re-performed in accordance with WCAP-14572, Revision 1-NP-A to compare the original ASME, Section XI program with the revised fourth 10-year interval RI-ISI program. On March 23, 2005, the licensee added that since the PRA model was updated since the implementation of the original RI-ISI program, the change-in-risk analysis was re-performed to ensure risk neutrality of the program. To maintain total plant risk neutrality, one segment was added. The licensee stated that this resulted in one additional visual VT-2 exam being added to the RI-ISI program. In its letter dated March 23, 2005, the licensee stated that the total change in risk and system level change in risk estimates for the proposed fourth 10-year RI-ISI interval program are within acceptance guidelines in WCAP-14572, Revision 1-NP-A. The NRC staff finds that the change in risk estimate is appropriate and the results provide assurance that the fourth key principle is met. By letter dated May 13, 2004, the licensee stated that consistent with the fifth key principle in RG 1.174, the fourth 10-year RI-ISI interval program is a "living program." Maintenance of a living program requires feedback of new relevant information to ensure the appropriate identification of HSS locations. As a result of recent and ongoing issues related to degradation due to pressurized-water stress corrosion cracking (PWSCC) in components that contain Alloy 600/82/182, the NRC staff requested that the licensee provide information regarding any dissimilar metal welds within the RI-ISI program that contain Alloy 82/182 filler metal. The licensee responded (Reference 2) that certain Surry, Unit 2 welds contain Alloy 82/182 filler metal. The RTD bypass line elimination design change implemented on Surry, Unit 2 installed Alloy 600 thermowells. Socket welds used to attach the thermowells to the existing stainless steel scoops inserted into the RCS loop piping. According to the licensee, these socket welds are either 1-inch or 1.5-inch in size and contain Alloy 82/182 filler metal. These welds are periodically examined as part of the augmented boric acid corrosion control program and the corresponding visual inspection program. The welds (12 total) are grouped into three RI-ISI segments based upon the RCS location. The original submittal had not identified these locations as having Alloy 82/182. The licensee states that had Alloy 82/182 material been previously identified, the expert panel would have considered these segments to be HSS, and going forward, these segments will be considered as such. The licensee stated that these welds will be included in an augmented inspection program. The licensee referenced a letter (Reference 11) that provided a supplemental response to NRC Bulletin 2002-01 in which it explained the discovery of these welds and described the licensee's augmented inspection program. The augmented inspection program for these welds includes a bare metal visual inspection during every refueling outage. A bare metal visual examination has proven to be a successful method of inspection for locating early stage leakage due to PWSCC. The NRC staff considers the licensee's bare metal visual inspection of these components a best effort and therefore acceptable given that a volumetric examination is not possible on the applicable weld joint configuration. However, since the licensee's relief request R-1 commits to performing a visual VT-2 exam on all HSS socket welds and their branch connections 2-inch NPS and smaller, it is the NRC staff's expectation that the licensee also conducts a visual VT-2 examination on the aforementioned welds as detailed in its relief request R-1. The NRC staff finds that the licensee's placement of the 12 welds in the RCS piping that contain Alloy 82/182 filler metal into the HSS category is consistent with the "living program," and therefore, the fifth key principle is met.

Based on the above discussion, the NRC staff finds that the five key principles of risk-informed decisionmaking are ensured by the licensee's proposed fourth 10-year RI-ISI interval program plan and, therefore, the proposed program for the fourth 10-year ISI inspection interval is acceptable.

3.1 Conclusion for RI-ISI Program

Based on the information provided by the licensee, the NRC staff has determined that the proposed alternative RI-ISI program provides an acceptable level of quality and safety and, therefore, it is authorized pursuant to 10 CFR 50.55a(a)(3)(i) for the fourth 10-year ISI interval at Surry, Unit 2.

4.0 RELIEF REQUEST R-1

4.1 The Components for Which Relief is Requested:

ASME Class 1 socket weld connections and their branch connections, nominal pipe size 2 inches (NPS 2) and smaller, identified as being HSS.

4.2 Code Requirement:

The licensee uses an alternative Class 1 RI-ISI program per WCAP-14572, Revision 1-NP-A, instead of ASME, Section XI requirements for piping (Categories B-F and B-J). WCAP-14572, Revision 1-NP-A, Table 4.1-1, requires a volumetric examination of HSS components (including socket welds).

4.3 Licensee's Proposed Alternative:

A VT-2 exam will be performed on the subject socket weld connections and their branch connections, NPS 2 and smaller, on a refueling outage frequency while the component is pressurized.

The VT-2 examination and pressure test required by Relief Request R-1 will conform to the requirements of ASME Section XI IWA-2000 & 5000 of the 1998 Edition through 2000 Addenda. Additionally, NRC approved Code Case –498-1 (or later revision as approved by the NRC) may be applied for the end of interval testing.

The following pressure test hold times will apply:

- Insulated components - 4 hours minimum at test pressure
- Non-insulated components - 10 minutes minimum at test pressure

A similar relief request was recently approved for Surry Unit 1 (3rd ISI Interval) in an NRC letter to Virginia Electric and Power Company dated September 23, 2003. This precedent is directly applicable to Surry Unit 2.

4.4 Licensee's Basis for Requesting Relief:

Pursuant to WCAP-14572, Revision 1-NP-A, certain socket weld connections and their branch connections, NPS 2 and smaller, have been identified as HSS and require volumetric examination for their postulated failure mechanism. Currently only three piping segments have been identified for Surry Unit 2. These volumetric examinations are associated with a postulated thermal fatigue damage mechanism, which is selected as the default mechanism for HSS segments when there is no assumed active mechanism or other postulated mechanism occurring. Performing a volumetric examination on a socket weld connection or the branch connection, NPS 2 and smaller, provides little or no benefit, due to limitations imposed by the joint configuration and the smaller pipe size. The socket welds are partial penetration welds and the branch connections may be partial or full penetration welds. These weld designs and pipe sizes under current Category B-J requirements would only require a surface examination.

The only thermal fatigue that could credibly affect the subject piping would be the low cycle fatigue previously considered in the design. Low cycle fatigue has a very low probability of causing cracking. Furthermore, even if cracking were to occur, it would most likely originate on the inside diameter of the pipe. In addition, the Class 1 RI-ISI program did not identify any locations susceptible to external chloride stress corrosion cracking. The Class 1 piping is not located in areas that are subject to an aggressive environment that would promote external chloride stress corrosion cracking (i.e., there are very low levels of chloride (if any) and moisture is not typically present on the pipe). No other externally driven damage mechanism can reasonably be postulated for this piping. Consequently, a surface exam would be of negligible benefit.

The ASME Code Committee has recognized the problem this relief request is addressing and has substituted the VT-2 examination method for all damage mechanisms on socket weld connections selected as HSS. ASME Code Case –577-1 has been issued and provides the requested substitution in Note 12 of Table 1 of the Code Case. Incorporation of the branch connection, NPS 2 and smaller, into the Code Case is now under consideration by the committee for similar size and joint configuration limitation reasons.

The industry is evaluating volumetric examination methods for socket welded connections for certain damage mechanisms. Dominion is following this effort and will address these developments as part of the WCAP-14572 "living program" process, if applicable.

Performing volumetric examinations on socket weld connections or their branch connections, NPS 2 and smaller, would result in unusual difficulty without providing any meaningful results or compensating increase in the level of quality and safety. Therefore, relief is requested per 10 CFR 50.55a(a)(3)(ii). Substituting a VT-2 examination as an alternative on a refueling outage frequency for these locations ensures reasonable assurance of component integrity.

4.5 NRC Staff Evaluation for Relief Request R-1

The licensee requested to perform a VT-2 examination each refueling outage in accordance with the requirements of ASME, Section XI IWA-2000 and 5000, or Code Case –498-1 on HSS socket welds and their associated branch connections, NPS 2 and smaller, in lieu of the Code-required surface exam or the volumetric exam directed by the WOG methodology in WCAP-14572, Revision 1-NP-A. The licensee indicated that Code Case –577 has been revised to allow the substitution of the VT-2 examination method for all damage mechanisms on socket welds identified as HSS. The licensee requested and received approval from the NRC staff for a similar relief request for the third interval at Surry, Unit 2. The licensee stated (Reference 2) that three segments are addressed by current Relief Request R-1, and for these segments no changes to the number or type of inspections are being requested from the previously approved relief request on socket welds for the Surry, Unit 2 RI-ISI program. Table IWB-2500-1 of the Code requires surface examination, not volumetric examination, at the socket welds. Surface examination (i.e., liquid penetration examination) is an effective method for the identification of outside surface-initiated flaws of specific concern, that is, flaws induced by low-cycle fatigue or by external chloride stress corrosion cracking (ECSCC). Information provided by the licensee in support of its previous relief request indicated that these three piping segments are not located in areas that are subject to an environment-promoting ECSCC or that the piping material is not considered susceptible to ECSCC, and an outside surface initiated flaw has a very low probability of occurrence due to the inclusion of low cycle fatigue in the piping design. Therefore, the NRC staff finds the proposed alternative reasonable because volumetric examinations are inconclusive due to the geometric limitations imposed by the socket welds and branch connections, and the licensee's alternative provides reasonable assurance of structural integrity of the subject piping welds

4.6 Conclusion for Relief Request R-1

Based on the above evaluation, the NRC staff concludes that the proposed visual VT-2 examinations performed during each refueling outage in lieu of the volumetric examinations specified in WCAP-14572, Revision 1-NP-A, for the subject HSS socket welds and their associate branch connections, NPS 2 and smaller, provide reasonable assurance of structural integrity for the subject piping welds. In addition, complying with the specified requirement would result in a hardship or unusual difficulty without a compensating increase in the level of quality and safety. Therefore, the NRC staff authorizes the proposed alternative pursuant to 10 CFR 50.55a(a)(3)(ii) for the fourth 10-year ISI interval at Surry, Unit 2.

5.0 CONCLUSIONS

Based on the above evaluations, the NRC staff has determined that the licensee's RI-ISI program based on the WOG methodology provides an acceptable level of quality and safety and, therefore, it is authorized pursuant to 10 CFR 50.55a(a)(3)(i) for the fourth 10-year ISI inspection interval at Surry, Unit 2. The NRC staff has also determined that the licensee's proposed alternative as stated in relief request R-1 to perform visual VT-2 examinations of HSS socket welds and their associate branch connections, NPS 2 and smaller, in lieu of the required volumetric examinations provides reasonable assurance of structural integrity of the subject piping welds. In addition, complying with the specified requirement would result in a hardship or unusual difficulty without a compensating increase in the level of quality and safety. Therefore, the NRC staff authorizes the proposed alternative pursuant to 10 CFR 50.55a(a)(3)(ii) for the fourth 10-year ISI interval at Surry, Unit 2. All other requirements of the ASME Code, Section XI for which relief has not been specifically requested and approved remain applicable, including third-party review by the Authorized Nuclear Inservice Inspector.

6.0 REFERENCES

1. Letter from Leslie N. Hartz, Virginia Electric and Power Company, dated May 13, 2004, to U.S. Nuclear Regulatory Commission, *Virginia Electric and Power Company, Surry Power Station Unit 2, ASME Section XI Fourth Inservice Inspection (ISI) Interval Update, Risk Informed Inservice Inspection (RI-ISI) Program.*

2. Letter from William R. Matthews, Virginia Electric and Power Company, dated March 23, 2005, to U.S. Nuclear Regulatory Commission, *Virginia Electric and Power Company, Surry Power Station Unit 2, ASME Section XI Fourth Inservice Inspection (ISI) Interval Update, Risk Informed Inservice Inspection (RI-ISI) Program, Response to NRC Request for Additional Information.*

3. Letter from Leslie N. Hartz, Virginia Electric and Power Company, dated April 27, 2000, to U.S. Nuclear Regulatory Commission, *Virginia Electric and Power Company, Surry Power Station Unit 2, Risk-Informed Inservice Inspection Program.*

4. Letter from Leslie N. Hartz, Virginia Electric and Power Company, dated

September 27, 2000, to U.S. Nuclear Regulatory Commission, *Virginia Electric and Power Company, Surry Power Station Unit 2, Risk-Informed Inservice Inspection Program, Request for Additional Information.*

5. Letter from Richard L. Emch, Jr., U.S. Nuclear Regulatory Commission, dated January 26, 2001, to David A. Christian (Virginia Electric and Power Company), *Request to Use an Alternative Risk-Informed Inservice Inspection (RI-ISI) Program at Surry Unit 2 (TAC No. MA8835).*

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

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

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

9. Letter from Leslie N. Hartz, Virginia Electric and Power Company, dated June 13, 2002, to U.S. Nuclear Regulatory Commission, *Virginia Electric and Power Company, North Anna Power Station Units 1 and 2, Surry Power Station Units 1 and 2, Risk-Informed ISI Relief Request R-1.*

10. Letter from John A. Nakoski, U.S. Nuclear Regulatory Commission, dated September 23, 2003, to David A. Christian (Virginia Electric and Power Company), Surry Power Station, Units 1 and 2, and North Anna Power Station, Units 1 and 2 - Risk-Informed Inservice Inspection Relief Request R-1 (TAC Nos. MB5437, MB5438, MB5439, and MB5440)

11. Letter from Leslie N. Hartz, Virginia Electric and Power Company, dated January 5, 2002, to U.S. Nuclear Regulatory Commission, *Virginia Electric and Power Company, North Anna and Surry Power Stations Units 1 and 2, NRC Bulletin 2002-01 Reactor Pressure Vessel Head Degradation and Reactor Coolant Boundary Integrity Supplemental Response.*

Principal Contributors: R. Davis
S. Malik

Date: August 8, 2005

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

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 <u>Engineering Analysis</u>

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.

Principal Contributor: P. Patnaik
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Date: May 19, 2003

Nuclear Operating Company

South Texas Project Electric Generating Station P.O. Box 289 Wadsworth, Texas 77483

January 10, 2002
NOC-AE-02001234
File No.: G25
10CFR50.55a

U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, DC 20555-0001

South Texas Project
Units 1 and 2
Docket Nos. STN 50-498, STN 50-446
Response to Request for Additional Information Regarding Risk-Informed
Inservice Inspection Application for Section XI Examination
Requirements for Class 1 and 2 Piping Welds (RR-ENG-2-23)

References: 1) "Request for Additional Information Re: Risk-Informed Inservice Inspection Application for South Texas Project, Units 1 and 2 (TAC Nos. MB1277 and MB1278)," Jack Donohew to William T. Cottle, dated December 10, 2001.

2) "Relief Request for Application of an Alternative to the ASME Boiler and Pressure Vessel Code Section XI Examination Requirements for Class 1 Socket-Welded Piping and Class 2 Piping Welds (RR-ENG-2-23)," T. J. Jordan to NRC Document Control Desk, dated February 27, 2001 (NOC-AE-01001034).

Pursuant to your request of December 10, 2001 (reference 1), the South Texas Project submits the attached responses to the Nuclear Regulatory Commission's questions regarding our request for relief from the ASME Section XI code requirements for inservice inspection of Class 1 socket-welded piping (Category B-J) and Class 2 piping welds (Categories C-F-1 and C-F-2) (reference 2). The relief request proposes a risk-informed inservice inspection program providing an acceptable level of quality and safety as an alternative in accordance with 10CFR50.55a(a)(3)(i).

If there are any questions, please contact either Mr. M. S. Lashley at (361) 972-7523 or me at (361) 972-7902.

T. J. Jordan
Manager,
Nuclear Engineering

PLW

Attachment: Response to Request for Additional Information Regarding Application of Risk-Informed Inservice Inspections at the South Texas Project, Units 1 and 2

A047

cc:

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U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D.C. 20555-0001

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION REGARDING APPLICATION OF RISK-INFORMED INSERVICE INSPECTIONS AT THE SOUTH TEXAS PROJECT, UNITS 1 AND 2

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

   STP Response: The ISI program will be updated and submitted to the NRC consistent with regulatory requirements in effect at the time such update is required (currently every 10 years). This may again take the form of a relief request to implement an updated RI-ISI program depending on future regulatory requirements.

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

   STP Response: The RI-ISI program will be resubmitted to the NRC prior to the end of any 10-year interval if there is some deviation from the RI-ISI methodology described in the initial submittal or if industry experience determines that there is a need for significant revision to the program as described in the original submittal for that interval. The South Texas Project will ensure that the RI-ISI program is monitored and periodically reviewed for risk ranking in accordance with the commitments made in Section 4 of the initial submittal. Revisions made as a result of these reviews will be considered for submittal as outlined above.

3. **Page 8 of the submittal presents the criteria for engineering evaluation and additional examinations if unacceptable flaws or relevant conditions are found during examinations. The submittal states that the evaluation will include whether other elements in the segment or segments are subject to the same root cause conditions. The submittal further states that additional examinations will be performed on these elements up to a number equivalent to the number of elements required to be inspected on the segment or segments initially. Please address the following:**

   (a) Please clarify the term "initially". Specifically, does it refer to inspections planned for the current outage or the current interval?

   (b) Please clarify how will the elements be selected for additional examinations. Specifically, please verify that the elements will be selected based on the root cause or damage mechanism and include high risk significant as well as medium risk significant elements (if needed) to reach the required number of additional elements.

   STP Response:

   (a) In this application, the term "initially" refers to those examinations originally scheduled for the current refueling outage.

   (b) Elements will be selected for additional examinations based on the root cause or damage mechanism and will include high risk-significant as well as medium risk-significant elements (if needed) to reach the required number of additional elements. Currently, there are no high risk-significant elements identified in the scope of the submittal.

4. **Page 5 of the submittal states that a deviation to EPRI [Electric Power Research Institute] RI-ISI methodology has been implemented in the failure potential assessment for thermal stratification, cycling and striping (TASCS). Please state if the revised methodology for assessing TASCS potential is in conformance with the updated criteria described in EPRI letter to NRC dated March 28, 2001. Also, please confirm that as stated in the subject letter, once the final Materials Reliability Program guidance has been developed, the RI-ISI program will be updated for the evaluation of susceptibility to TASCS, as appropriate.**

   STP Response: The methodology for assessing TASCS potential used in the South Texas Project RI-ISI submittal is identical to the methodology described in the Electric Power Research Institute (EPRI) letter to the NRC, dated March 28, 2001. The South Texas Project will update the RI-ISI program based on the final EPRI Material Reliability Program guidance as warranted.

5. **The submittal states that the scope includes Category B-J socket welds. Please state what examination method will be utilized for the inspection of socket welds.**

   STP Response: The ASME Code through Code Case N-578-1 endorses substitution of visual (VT-2) exams for volumetric exams on socket welds. A surface exam on a socket weld on a ten-year frequency is not likely to identify any inside diameter-originating damage mechanism and is considered an unnecessary radiation exposure burden. A frequent visual examination (VT-2) focused on the area of concern is the best alternative as proposed by the Code. The industry (NEI/EPRI) met with the NRC on August 29, 2000, to discuss risk-informed issues. The VT-2 exam for socket welds was discussed and the proposed substitution was again endorsed by the industry. It was noted during the meeting that the EPRI-MRP thermal fatigue task group was due to issue a formal report in 2001. The report will be reviewed by the South Texas Project for any impact on the RI-ISI program and will be considered as new information with regard to socket welds.

6. **Section 3.6.1 states that, for medium consequence category segments, boundary estimates of 1E-4 and 1E-5 were used for the conditional core damage and large early release frequency respectively. What was used for the high consequence category segments?**

   STP Response:

   The High Consequence Category limits are:

   CCDP > 1E-04

   CLERP > 1E-05

   The Medium Consequence Category limits are:

   1E-06 < CCDP ≤ 1E-04

   1E-07 < CLERP ≤ 1E-05

   The Low Consequence Category limits are:

   CCDP ≤ 1E-06

   CLERP ≤ 1E-07

7. **Section 1.2 of your submittal states that the Level 2 probabilistic safety assessment (PSA) and individual plant examination (IPE) submittal dated August 28, 1992, supplemented by the current probabilistic risk assessment (PRA) model, STP_1997, were used to support the RI-ISI submittal. The October 14, 1999 letter transmitting procedures and diagrams for the proposed Risk Informed Exemption included a copy of the Probabilistic Risk Assessment Program, 0PGP04-ZA-0604, Rev. 3. The procedure includes the following two steps.**

   6.3.8 The overall PRA model results are updated every refueling cycle of Unit 1, not to exceed two years, or when the Risk & Reliability Analysis Administrator determines (using guidance supplied by 0PGP01-ZA-0305) an update is required.

   6.3.20 Each update cycle, the Updated PRA (including the Updated PRA Computer Model) is documented as "complete" via a signed letter from the Risk & Reliability Analysis Administrator to RMS. Computer codes are maintained in accordance with 0PGP05-ZA-0014, "Software Quality Assurance Program."

   It appears that the 1997 model (i.e., STP_1997) referenced in the February 27, 2001, submittal would have been more than two years old when the RI-ISI submittal was being prepared; however, the update procedure indicates that the models should normally be no more than two years old. Please explain the apparent discrepancy of using a "current PRA model, STP_1997" in the February 27, 2001 submittal.

   STP Response: At the time of the evaluations for this RI-ISI submittal (June to September 2000), the approved PRA model was STP_1997. The model name refers to the data freeze date: the STP_1997 freeze date was December 31, 1997. This model was approved for use in March 1999.

   PRA model STP_1999 was being developed during the RI-ISI evaluations. The model freeze date for STP_1999 was December 31, 1999. Because of the effort involved in supporting this submittal and other risk-informed applications, the PRA update process for this model was extended. Model STP_1999 was approved for use in October 2001.

   Procedure 0PGP04-ZA-0604 was revised in May 2001, changing the frequency at which the model is updated. The procedure currently states:

   5.3.5 The at-power PRA applicable to modes 1 and 2 (Level 1 and Level 2 PRA) Reference Model SHALL be periodically updated in accordance with the following requirements:

   - The Reference Model Update incorporates plant design changes and procedure changes that affect PRA model components, initiating event frequency updates, and changes in SSC unavailability that affected the PRA model. These changes will be incorporated into the model on a period not to exceed 36 months. (Ref. 6.5)

   - The comprehensive data update incorporates changes to plant specific failure rate distributions and human reliability, and any other database distribution updates (examples would include equipment failure rates, recovery actions, and operator actions). This second category will be updated on a period not to exceed 60 months. (Ref. 6.5)

- A necessary change to the PRA Reference Model that would result in an increase to CDF of greater than or equal to 10% (Ref. 6.8, and 6.9.1).
- Every refuel cycle, the previous cycle's significant operator experience human performance trends SHOULD be reviewed. This review shall check for adverse trends, and new information that could affect the way operator actions are currently modeled in the PRA Reference Model (Ref. 6.10.1).

LIST OF COMMITMENTS

Commitment	Due Date/Event
The methodology for assessing TASCS potential used in the South Texas Project RI-ISI submittal is identical to the methodology described in the Electric Power Research Institute (EPRI) letter to the NRC, dated March 28, 2001. The South Texas Project will update the RI-ISI program based on the final EPRI Material Reliability Program guidance as warranted.	Upon issuance and review of the final EPRI material reliability program.
The ASME Code through Code Case N-578-1 endorses substitution of visual (VT-2) exams for volumetric exams on socket welds. The EPRI-MRP thermal fatigue task group was due to issue a formal report in 2001 endorsing the substitution of visual (VT-2) exams for volumetric exams on socket welds. The report will be reviewed by the South Texas Project for any impact on the RI-ISI program and will be considered as new information with regard to socket welds.	Upon issuance and review of the thermal fatigue task group formal report.

June 16, 2000

MEMORANDUM TO: Robert A. Gramm, Chief
Project Section IV-1
Division of Licensing Project Management

FROM: Edmund J. Sullivan, Chief ***/ra by/***
NDE & Metallurgy Section
Materials & Chemical Engineering Branch
Division of Engineering

Mark P. Rubin, Chief ***/ra by/***
Safety Program Section
Probabilistic Safety Assessment Branch
Division of Systems Safety & Analysis

SUBJECT: SAFETY EVALUATION OF RISK-INFORMED ISI PROGRAM, RELIEF REQUEST RR-ENG-2-16 FOR THE SECOND INTERVAL FOR SOUTH TEXAS PROJECT, UNITS 1 AND 2 (TAC NOS. MA7789 & 7790)

In a letter dated December 30, 1999, Houston Lighting and Power Company, submitted Relief Request RR-ENG-2-16, which proposed a risk-informed inservice inspection (RI-ISI) program as an alternative to the current ISI program at South Texas Project (STP), Units 1 and 2, for a subset of Class 1 piping welds, the Categories B-F and non-socket B-J welds. The program was developed in accordance with the methodology contained in the Electric Power Research Institute (EPRI) report EPRI-TR 112657, which has been approved by the staff. This relief request was made pursuant to 10 CFR 50.55a(a)(3)(i) for the second ten-year ISI interval.

The staff review of the proposed RI-ISI program concludes that the program is an acceptable alternative to the current ISI program based on ASME Code, Section XI requirements for Class 1, Categories B-F and non-socket B-J welds, and therefore, the licensee's request for relief, RR-ENG-2-16, is authorized pursuant to 10 CFR 50.55a(a)(3)(i) on the basis that the request provides an acceptable level of quality and safety. The attached safety evaluation authorizes implementation of the proposed RI-ISI program for the second ten-year ISI interval, which begins on September 25, 2000, and October 19, 2000, for STP Units 1 and 2 respectively. This completes our task on TAC Nos. MA7789 and MA7790.

Attachment: As stated

Docket Nos.: 50-498 and 50-499

CONTACTS: Shou-nien Hou
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415-2585

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION
RISK-INFORMED INSERVICE INSPECTION PROGRAM
PROPOSED BY THE STP NUCLEAR OPERATING COMPANY
FOR SOUTH TEXAS PROJECT ELECTRIC GENERATING STATION, UNITS 1 AND 2
RELIEF REQUEST RR-ENG-2-16
DOCKET NOS. 50-498 AND 50-499

1.0 INTRODUCTION

Inservice inspection of the American Society of Mechanical Engineers (ASME) Code Class 1, 2, and 3 components shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code (hereafter called ASME Code) and applicable addenda as required by 10 CFR 50.55a(g), except where specific written relief has been granted by the Commission pursuant to 10 CFR 50.55a(g)(6)(i). 10 CFR 50.55a(a)(3) states in part that alternatives to the requirements of paragraph (g) may be used, when authorized by the NRC, if the proposed alternatives would provide an acceptable 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 set forth in the ASME Code, Section XI, "Rules for Inservice Inspection 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 conducted during the first ten-year interval and subsequent intervals comply with the requirements in the latest edition and addenda of Section XI of the ASME 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. For South Texas Project (STP) Electric Generating Station, Units 1 and 2, the applicable edition of Section XI of the ASME Code for the second ten-year inservice inspection (ISI) interval, which begins on September 25, 2000 and October 19, 2000 for Units 1 and 2, respectively, is the 1989 edition.

By letter dated December 30, 1999, the licensee, STP Nuclear Operating Company, submitted Relief Request RR-ENG-2-16 (Reference 1). The submittal proposed a risk-informed inservice inspection (RI-ISI) program as an alternative to the current ISI program for a subset of Class 1 piping welds - Categories B-F welds (pressure retaining dissimilar metal welds in vessel nozzle) and non-socket B-J welds (pressure retaining welds in piping). The licensee states that its program was developed in accordance with the methodology contained in the Electric Power Research Institute (EPRI) report EPRI TR-112657 (Reference 2), which has been reviewed and approved by the Nuclear Regulatory Commission (NRC) staff. This relief request was made pursuant to 10 CFR 50.55a(a)(3)(i) for the second ten-year ISI interval, which begins in the fall of 2000.

2.0 SUMMARY OF PROPOSED APPROACH

The ISI program for the upcoming second 10-year interval will be performed to the 1989 Edition of the ASME Code, Section XI. The ASME Code, Section XI, requires in part, that for each successive ten-year ISI interval, 100% of Category B-F welds and 25% of Category B-J welds for ASME Code Class 1 piping greater than 1-inch in nominal diameter are selected for volumetric and/or surface examination based on existing stress analyses and cumulative usage factors.

The licensee has proposed to use a RI-ISI program for a subset of ASME Class 1 piping (Categories B-F and non-socket B-J welds only), as an alternative to the ASME Code, Section XI requirements. ISI program requirements of other non-related portions of the ASME Code, Section XI will be unaffected. The submittal follows the RI-ISI program template application. Template applications are short overview submittals intended to expedite preparation and review of RI-ISI program submittals that comply with a pre-approved methodology. The licensee proposed to implement the staff approved RI-ISI methodology delineated in EPRI TR-112657.

In addition, the licensee indicated that the augmented ISI program implemented in response to NRC Bulletin 88-08, "Thermal Stresses in Piping Connected to Reactor Coolant Systems," has been subsumed into the RI-ISI program, because the potential for thermal fatigue is explicitly considered in the application of the EPRI RI-ISI process. Other remaining augmented ISI programs are unaffected.

3.0 EVALUATION

The NRC staff, pursuant to 10 CFR 50.55a(a)(3)(i), reviewed and evaluated the licensee's proposed RI-ISI program, including those portions related to applicable methodology and processes contained in EPRI TR-112657, based on guidance and acceptance criteria provided in Regulatory Guides (RG) 1.174 (Reference 3) and 1.178 (Reference 4), and in Standard Review Plan (SRP) Chapter 3.9.8 (Reference 5).

3.1 Proposed Changes to the ISI Program

The scope of the licensee's RI-ISI program covers only Category B-J non-socket piping welds and Category B-F dissimilar metal nozzle welds. The program was proposed as an alternative to the existing ISI program for STP Unit 1 and 2, which is based on examination requirements of the ASME Code, Section XI. A general description of the proposed changes to the ISI program was provided in Sections 3 and 5 of the licensee's submittal.

The staff review verified that the RI-ISI program follows the guidelines contained in EPRI TR-112657, which states that industry and plant-specific piping failure information, if any, is to be utilized to identify piping degradation mechanisms and failure modes, and consequence evaluations are to be performed using probabilistic risk assessments to establish piping segment safety ranking for determining new inspection locations. Thus the staff concludes that the licensee's application of the EPRI TR-112657 approach is an acceptable alternative to the current STP piping ISI requirements with regard to the number, locations, and methods of inspections, and provides an acceptable level of quality and safety pursuant to 10 CFR 50.55a(a)(3).

In Table 5-1 of the submittal, a comparison of inspection location selection between the current ISI program and the proposed RI-ISI program is provided. The staff finds that the information submitted adequately defines the proposed changes to the ISI program.

3.2 Engineering Analysis

In accordance with the guidance provided in RGs 1.174 and 1.178, an engineering analysis of the proposed changes using a combination of traditional engineering analysis and supporting insights from probabilistic risk assessment (PRA) was performed. The licensee discussed how the engineering analyses conducted for the STP, Units 1 and 2, RI-ISI program ensure that the proposed changes are consistent with the principles of defense-in-depth, and that adequate safety margins will be maintained. The licensee evaluated a location's susceptibility to a particular degradation mechanism that may be a precursor to leak or rupture, and then performed an independent assessment of the consequence of a failure at that location.

STP, Units 1 and 2, RI-ISI program is limited to a subset of ASME Class 1 piping welds - Category B-F and non-socket B-J welds only. The licensee stated in its submittal that other non-related portions of the ASME Section XI Code will be unaffected by this program. The licensee's submittal also states that STP chose to include Category B-F dissimilar weld locations (as a deviation to the EPRI methodology) in their application in addition to the non-socket Category B-J weld locations. The staff's review and approval of EPRI TR-112657 includes the full-scope application of RI-ISI methodology not only to Category B-J welds, but also to Category B-F welds. Application of the methodology to B-F welds have been implemented in the RI-ISI programs of Surry Unit 1 and ANO Unit 2. Therefore, we find that contrary to the licensee's statement in their submittal, the inclusion of B-F welds in the STP RI-ISI program is not a deviation from the EPRI-TR methodology.

The RI-ISI program results in a reduction in required examination locations from 151 in Unit 1 and 132 in Unit 2 to 59 in each unit. The submittal also states that for the Class 1 piping at STP, the augmented inspection program implemented during the first inspection interval in response to NRC Bulletin 88-08 regarding thermal fatigue is subsumed in the STP, Units 1 and 2, RI-ISI programs since the potential for thermal fatigue is explicitly considered in the application of the EPRI RI-ISI process. Additionally, the licensee stated that the remaining augmented inspection programs are unaffected by the proposed RI-ISI program. The staff concludes that this approach to augmented inspection programs is consistent with the approved EPRI TR-112657, Chapter 6 guidelines, and therefore, is acceptable.

Piping systems within the scope of the RI-ISI program were divided into piping segments. Pipe segments are defined as lengths of pipe whose failure would lead to the same consequence and which are exposed to the same degradation mechanism. That is, some lengths of pipe whose failure would lead to the same consequences may be split into two or more segments when two or more regions are exposed to different degradation mechanism. The staff finds this appropriate, and necessary, because the methodology combines separate consequence categories with degradation mechanism categories and therefore the two characteristics should not be mixed within a segment. The licensee's submittal also states that failure potential estimates were generated utilizing industry failure history, plant-specific failure history and other relevant information using the guidance provided in EPRI TR-112657. The staff concludes that the licensee has met the SRP 3.9.8 guidelines to confirm that a systematic process was used to identify pipe segments susceptibility to common degradation mechanisms, and to categorize these degradation mechanisms into the appropriate degradation categories with respect to their potential to result in a postulated leak or rupture.

Additionally, the licensee stated that the consequences of pressure boundary failure were evaluated and ranked based on their impact on core damage and containment performance (isolation, bypass and large, early release), and that the impact due to both direct and indirect effects was considered using guidance provided in the EPRI TR-112657 topical report approved by the staff. The licensee reported no deviations from the consequence evaluation methodology approved by the staff in the EPRI topical report. Based on the licensee's statements, the staff finds the consequence evaluation performed for this application to be acceptable.

3.3 Probabilistic Risk Assessment

The licensee used their Individual Plant Examination (IPE) Level 2 STP PRA, supplemented by their current PRA model, STP-1997, to evaluate the consequences of pipe rupture for the RI-ISI assessment. The December 1999 RI-ISI submittal reported a base core damage frequency (CDF) of 1.17E-5 /year and a base large early release frequency (LERF) of 5.50E-7/year.

The STP IPE was submitted in August 1992 and supplemented in November 1994. The IPE identified a point estimate of the total CDF as 4.4E-5/year from both internal events (97%, 4.3E-5/year) and external events (3%, 1.3E-6/year). Additionally, the IPE reported a LERF of about 1E-6/year. The staff evaluation report, dated August 28, 1995, concluded that the STP IPE satisfied the intent of Generic Letter (GL) 88-20, "Individual Plant Examination for Severe Accident Vulnerabilities." The staff evaluation also concluded that the licensee's intent to continue to use and maintain its Probabilistic Safety Assessment (PSA) will enhance plant safety and provide additional assurance that any potentially unrecognized vulnerabilities would be identified and evaluated during the lifetime of the plant.

The staff evaluation report identified four items which the licensee planned to incorporate in its living PSA. These items included: (1) implementation of the RISKMAN 3.0 system conversions for calculating internally generated initiating events, (2) revision of the system analyses and event tree rules to reflect the practice of operating two emergency cooling water (ECW) trains and one standby train instead of one train "on," one train "off," and one train in "standby," (3) incorporation of new system analyses and split fractions data for new top events, and (4) consideration of accident management strategies of intentional primary system depressurization and post core damage recovery. The December 1999 RI-ISI submittal identified that the RISKMAN 3.0 system conversions for calculating internally generated initiating events were implemented in the 1994 model update and are maintained and upgraded in accordance with the licensee's PRA control program. The submittal also stated that the system analyses and event tree rules to reflect the practice of ECW trains were revised in 1994 and expanded to include any possible configuration of operating and standby trains in 1996. Additionally, the licensee stated that new system analyses and split fractions data for new top events were included in 1994 and are maintained in accordance with the licensee's PRA control program. The licensee addressed the issue of accident management strategies by adopting the Severe Accident Management Guidelines (SAMGs) published in June 1997 and including SAMGS in the current Level 2 analyses for the current PRA model, STP-1997.

The staff did not review the IPE analysis to assess the accuracy of the quantitative estimates. The staff recognizes that the quantitative results of the IPE are used as order of magnitude estimates for several risk and reliability parameters used to support the assignment of segments into three broad consequence categories. Inaccuracies in the models or assumptions large enough to invalidate the broad categorizations developed to support RI-ISI should have been identified in the licensee or the staff reviews. Minor errors or inappropriate assumptions will only affect the consequence categorization of a few segments and not invalidate the general results or conclusions. The staff finds the quality of the PRA sufficient to support the submittal.

The degradation category and the consequence category were combined according to the approved methodology described in the EPRI topical to categorize the risk significance of each segment. The risk significance of each segment is used to determine the number of weld inspections required in each segment

The South Texas submittal proposes to reduce the examination of ASME Code, Section XI, category B-F and B-J welds from 151 welds to 59 welds (from 26% to 10%) for Unit 1 and from 132 welds to 59 welds (from 24% to 10.6%) for Unit 2, and to change the location of some of the ASME Code, Section XI weld inspections. The licensee conducted a bounding analysis to estimate the change in risk expected from replacing the current ISI program with the RI-ISI program. The calculations estimated the change in risk due to removing locations and adding locations to the inspection program. For high consequence category segments, the licensee used the conditional core damage probability (CCDP) based on the highest evaluated CCDP. For medium and low consequence category segments, bounding estimates of CCDP were used. The licensee estimated the change in risk using bounding failure rates from the EPRI methodology. The licensee performed their bounding analysis with and without taking credit for an increased probability of detection (POD). The results of their analysis for Units 1 and 2 shown below, indicate that even without credit for improved POD, the estimates are less than the EPRI methodology guideline values.

	ΔCDF using EPRI TR-112657 piping failure frequency	
	No Improved POD	**Improved POD**
Unit 1	8.76E-08	-1.06E-07
Unit 2	7.62E-08	-1.11E-07

The licensee did not calculate the change in LERF in their risk impact assessment. However, in their consequence evaluation, the licensee evaluated the consequence of pressure boundary failures based on core damage and containment performance, including isolation, bypass and large, early release. The licensee found that there were no segments that would create a concern for containment isolation or bypass. Additionally, there were no sequences arising from pipe rupture that had a conditional containment failure probability given core damage greater than 0.1. Since the estimated change in CDF for each evaluation is below 1E-7 (as shown in the table above), and the conditional LERF given core damage is below 0.1, the estimated change in LERF will not exceed 1E-8 and does not exceed the safety goal guidelines recommended in RG 1.174.

The staff finds the licensee's bounding analysis of the change in risk acceptable because it accounts for the change in the number of elements inspected, recognizes the difference in degradation mechanism related to failure likelihood, and considers the effects of enhanced inspection. The staff finds that the risk impact assessment follows the approach of EPRI TR-112657 and that re-distributing the welds to be inspected with consideration of the safety-significance of the segments provides assurance that segments whose failure have a significant impact on plant risk receive an acceptable level of inspection. Therefore, the staff concludes that the implementation of the RI-ISI program as described in the licensee's application will have a small impact on risk consistent with the guidelines of RG 1.174, and, thus, will not cause the NRC safety goals to be exceeded.

3.4 Integrated Decision Making

As described in the STP submittal, an integrated approach is utilized in defining the proposed RI-ISI program by considering in concert the traditional engineering analysis, risk evaluation, and the implementation and performance monitoring of piping under the program. This is consistent with the guidelines of RG 1.178.

The selection of pipe segments to be inspected is described in Section 3.5 of the submittal using the results of the risk category rankings and other operational considerations. Table 3.5-1 of the submittal provides the number of locations and inspections by risk category for the various STP Unit 1 and Unit 2 systems. Table 5-1 of the submittal provides a summary table comparing the number of inspections required under the existing ASME Section XI ISI program with the alternative RI-ISI program. Tables 3.7-1 identifies on a per system basis each applicable risk category based on EPRI TR-112657 upper bound failure rates, and Table 3.7-2 identifies on a per system basis each applicable risk category based on EPRI TR-111880, "Piping System Failure Rates and Rupture Frequencies for Use in Risk Informed Inservice Inspection Applications," best estimate failure rates. The licensee used the methodology described in EPRI TR-112657 to guide the selection of examination elements within high and medium risk ranked piping segments. The licensee included Category B-F dissimilar weld locations in the scope of their application, as discussed in Section 3.2 of this SE. The methodology described in the EPRI topical report requires that existing augmented programs, other than thermal fatigue which the RI-ISI program supercedes, be maintained. The EPRI report describes targeted examination volumes (typically associated with welds) and methods of examination based on the type(s) of degradation expected. The staff has reviewed these guidelines and has determined that, if implemented as described, the RI-ISI examinations should result in improved detection of service-related degradations over that currently required by ASME Code, Section XI.

The staff finds that the location selection process is acceptable since it is consistent with the process approved for the EPRI TR-112657, takes into account defense-in-depth, and includes coverage of systems subjected to degradation mechanisms in addition to those covered by augmented inspection programs.

The objective of ISI required by ASME Section XI 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 should meet this objective if found to be acceptable for use. Further, since the risk-informed program is based on inspection for cause, element selection should target specific degradation mechanisms.

Chapter 4 of EPRI TR-112657 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 review of the cited portion of the EPRI report, the staff concludes that the examination methods are appropriate since they are selected based on specific degradation mechanisms, pipe sizes and materials of concern.

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 adequate level of quality and safety.

In the December 30, 1999, submittal, the licensee stated that upon approval of the RI-ISI program, procedures that comply with the EPRI TR-112657 guidelines will be prepared to implement and monitor the RI-ISI program. The licensee confirmed that the applicable portions of the ASME Code, 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 the submittal 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 high safety significant piping locations. The submittal also states that as a minimum, risk ranking of piping segments will be reviewed and adjusted on an ISI period basis and that significant changes may require more frequent adjustment as directed by NRC bulletin or generic letter requirements, 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 staff finds that the proposed process for RI-ISI program updates meets the guidelines of RG 1.174 which provide that risk-informed applications should include performance monitoring and feedback provisions; therefore, the process for program updates is acceptable.

4.0 CONCLUSION

In accordance with 10 CFR 50.55a(a)(3)(i), proposed alternatives to regulatory requirements may be used when authorized by the NRC when the applicant demonstrates that the alternative provides an acceptable level of quality and safety. In this case, the licensee's proposed alternative is to use the risk-informed process described in the NRC-approved report EPRI-TR 112657. As discussed in Section 3.0 above, the staff concludes that the licensee's proposed RI-ISI program, as described in the submittal, will provide an acceptable level of quality and safety pursuant to 10 CFR 50.55a with regard to the number of inspections, locations of inspections, and methods of inspection.

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

The STP 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.

System pressure tests and visual examination of piping structural elements will continue to be performed on all Class 1, 2, and 3 systems in accordance with the ASME Code Section XI program. The RI-ISI program applies the same performance measurement strategies as existing ASME Code requirements and, in addition, increases the inspection volumes at weld locations that are exposed to thermal fatigue.

The STP methodology provides for conducting an engineering analysis of the proposed changes using a combination of engineering analysis with supporting insights from a PRA. Defense-in-depth quality is 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 with active degradation mechanisms as well as selected locations that monitor the performance of system piping.

As discussed above, the staff's review of the licensee's proposed "limited scope" 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 Class 1, Categories B-F and non-socket B-J welds. Therefore, the licensee's request for relief, RR-ENG-2-16 is authorized pursuant to 10 CFR 50.55a(a)(3)(i) on the basis that the request provides an acceptable level of quality and safety. This safety evaluation authorizes implementation of the proposed RI-ISI program for the second ten-year ISI interval, which begins at STP Units 1 and 2 on September 25, 2000, and October 19, 2000, respectively.

5.0 REFERENCES

1. Letter, dated December 30, 1999, T. J. Jordan (South Texas Project, Units 1 and 2, Manager, Systems Engineering), to U.S. Nuclear Regulatory Commission, containing *Risk-Informed Inservice Inspection Program Plan - South Texas Project Units 1 and 2.*

2. EPRI TR-112657, Revision B-A, *Revised Risk-Informed Inservice Inspection Evaluation Procedure*, January 2000.

3. NRC RG 1.174, *An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis*, July 1998.

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

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

Inspection of Socket Welds in Class 1 Small-Bore Piping

Issue

- Aging management of socket welds in Class 1 small-bore piping (less than NPS 4 inches)
- Should socket welds be included in the "One-time Inspection of Small Bore Piping" AMP (XI.M35)

Background

- GALL report for One Time Inspection of Small-Bore Piping does not mention socket welds
- ASME Section XI IWB-2500, Category B-J requires a surface examination for socket welds
- ASME Section XI IWB-2500 exempts piping and components less than 1 inch NPS from examination requirements

Background (cont)

- RI-ISI identifies high-safety-significant (hss) locations for inspection
- Socket welds have been identified as hss
- Approved Topical Reports for RI-ISI require volumetric for all hss locations
- August 29, 2000, industry (NEI/EPRI) met with the NRC staff, industry proposed substitution of VT-2 for volumetric examination of socket welds
- ASME Code Case N-578-1 states "socket welds require only a VT-2 examination during each refueling outage"

Background (cont)

- There is no qualified volumetric examination techniques for socket welds
- Most cracks in socket welds are ID initiated and surface examinations will not detect cracks until they are through wall
- VT-2 examinations have been permitted in a number of plants during each refueling outage in lieu of volumetric or surface examinations

Oyster Creek Experience with Socket Welds

- Operating Experience – 1998 to present, no failures of small bore piping due to OD cracking
- 1985 to 2000 – in response to industry concerns about vibration related and thermal fatigue failures of small bore socket welds, one failure of a socket weld identified but the root cause was defective weld
- Oyster Creek does not use RI-ISI so hss socket welds have not been identified

Conclusions

- No additional examinations will be required for socket welds in excess of the current ASME code requirements