INDIANA MICHIGAN POWER

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Nuclear Generation Group
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October 7, 2009

AEP-NRC-2009-70
10 CFR 50.55a

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

Subject: Donald C. Cook Nuclear Plant Unit 1
Docket No. 50-315
Revised MRP-139 Deviation Notification

Reference: Letter from R. A. Hruby, Indiana Michigan Power Company, to Nuclear Regulatory Commission Document Control Desk, "Donald C. Cook Nuclear Plant Unit 1, Notice of MRP-139 Deviation," AEP-NRC-2009-58, dated July 24, 2009 (ML092230352).

Dear Sir or Madam:

In the above reference, Indiana Michigan Power Company (I&M) submitted a notice of deviation from Materials Reliability Program (MRP)-139 for Donald C. Cook Nuclear Plant (CNP) Unit 1. Subsequently, based on industry comments, the original technical justification has been revised. The revised technical justification for the deviation has been prepared and concurrence by an independent third party materials expert has been obtained in accordance with Nuclear Energy Institute (NEI) 03-08 requirements.

In accordance with the NEI Guideline for the Management of Materials Issues (NEI 03-08), I&M is providing notification of the revised deviation from the Electric Power Research Institute (EPRI) MRP: Primary System Piping Butt Weld Inspection and Evaluation Guidelines (MRP-139). This notification is being sent consistent with the industry initiatives to provide timely communications to U. S. Nuclear Regulatory Commission staff regarding conformance with industry guidance. This notification is for information only and no response is requested.

This is a schedular deviation from the implementation requirements of MRP-139, "Primary System Piping Butt Weld Inspection and Evaluation Guideline." The specific deviation being requested is the deferral of ultrasonic inspections of CNP Unit 1 Reactor Vessel Hot Leg Nozzle to Safe End welds until the U1C23 refueling outage, now scheduled for March of 2010. The schedular deferral (for a March 2010 refueling outage) represents an approximate three-month deviation for the reactor vessel hot legs and no required deviation for the reactor vessel cold legs.

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NRR

There are no new or revised commitments in this letter. Should you have any questions, please contact Mr. James M. Petro, Jr., Regulatory Affairs Manager, at (269) 466-2489.

Sincerely,

R. A. Hruby, Jr.

Raymond A. Hruby, Jr.
Vice President – Site Support Services

RSP/rdw

Attachment:

Letter from Joseph N. Jensen, Indiana Michigan Power Company, to Electric Power Research Institute (EPRI) Materials Reliability Program (MRP), "Revised Technical Justification for Deviation from EPRI MRP-139 Inspection Requirements for Reactor Vessel Alloy 600/82/182 Welds at DC Cook Nuclear Plant."

c: T. A. Beltz – NRC Washington, DC
J. T. King – MPSC
S. M. Krawec, Ft. Wayne AEP
MDEQ – WHMD/RPS
NRC Resident Inspector
M. A. Satorius – NRC Region III

Attachment to AEP-NRC-2009-70

Letter from Joseph N. Jensen, Indiana Michigan Power Company, to Electric Power Research Institute (EPRI) Materials Reliability Program (MRP), "Revised Technical Justification for Deviation from EPRI MRP-139 Inspection Requirements for Reactor Vessel Alloy 600/82/182 Welds at DC Cook Nuclear Plant."

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Electric Power Research Institute
Materials Reliability Program
Attn: Christine King, Program Manager
3420 Hillview Avenue
Palo Alto, CA 94304

September 25, 2009

Dear Ms. King:

Subject: Revised Technical Justification for Deviation from EPRI MRP-139 Inspection Requirements for Reactor Vessel Alloy 600/82/182 Welds at DC Cook Nuclear Plant,

Indiana Michigan Power Company (I&M) is submitting this revised report pursuant to the requirements of NEI 03-08, "Materials Guidelines Implementation Protocol," for a deviation to the requirements of EPRI MRP-139, "Material Reliability Program: Primary System Piping Butt Weld Inspection and Evaluation Guideline."
This report provides the engineering evaluation establishing the basis to deviate from the MRP-139 ultrasonic (UT) inspection schedule requirement for the D.C.Cook Plant Alloy 82/182 Reactor Vessel Nozzle welds.

Should you have any questions, please call Carl Lane at (269) 466-2894 at your convenience.

Sincerely,

Joseph Jensen
Chief Nuclear Officer and Senior Vice President

PD/adg

1 Enclosure

NDM Correspondence Control #2009-695

CR Number: 00844056
Technical Justification for Deviation from EPRI MRP-139 Mandatory Schedule for Inspection Requirements for Alloy 600/82/182 Welds for Cook Unit 1

Prepared By (Preparer)	9/16/09 Date
Reviewed By (Reviewer)	9/17/09 Date
Approved By (Supervisor)	09/17/2009 Date
Engineering Programs Manager	9/21/09 Date
Vice President – Site Support Services	9/23/2009 Date
Senior Vice President & Chief Nuclear Officer	9/25/09 Date
SIGNATURE PER ATTACHED E-MAIL CRL 9/21/09 Independent Reviewer	Date

Executive Summary

Materials Reliability Program initiative (MRP-139) requires inspection or mitigation of dissimilar metal welds (DMW) that are susceptible to primary water stress corrosion cracking (PWSCC). The requirements, as they apply to Cook Nuclear Plant, are broken into the following applicable categories:

- Pressurizer nozzle safe-end welds
- Reactor vessel nozzle safe-end welds

Cook Nuclear Plant (CNP) has completed mitigation activities for all of the Pressurizer nozzle safe-end welds for both Unit 1 and Unit 2.

The MRP-139 guidelines stipulate by December 31, 2009, Alloy 82/182 butt welds that are greater than 14" NPS and exposed to temperatures equivalent to the hot leg will be volumetrically inspected per this guideline. The guideline provides direction when the mandatory requirements cannot be met.

Status: Unit 1 Reactor Vessel Hot Leg Nozzle to Safe End welds meet these criteria. Unit 2 Reactor Vessel Hot Leg Nozzle to Safe End has stainless steel welds and does not meet these criteria. No other Reactor Coolant System butt welds are in the population because they are stainless steel. This was determined via WCAP-16198-P, July 2004, Revision 1 PWSCC Susceptibility Assessment of the Alloy 600 and Alloy 82/182 Components in D. C. Cook Units 1 and 2.

The Unit 1 Reactor Vessel Hot Leg Nozzle to Safe End welds have been examined ultrasonically from the ID as part of the 10 year vessel examination using qualified personnel and procedures. These examinations were completed to ASME Section XI criteria. 100 percent coverage was obtained using the industry standard technology with no flaws reported.

CNP has implemented the PWROG enhanced leakage guideline. The guidance is contained in CNP Surveillance Procedure 1-OHP-4030-102-016 "Reactor Coolant System Leak Rate Test". The procedure meets the latest Westinghouse guidance for conducting reactor coolant system (RCS) leak rate testing. The procedure requires that a small increase in the RCS leak rate (0.05 GPM) be reported to plant management for action. This low threshold is designed to alert plant management of a potential PWSCC through wall leak.

The Unit 1 reactor vessel hot and cold leg nozzles have had bare metal visual (BMV) examinations during the last refueling outage in Spring 2008 (U1C22). The construction record review revealed no repairs to the area of interest.

Original Schedule: The Unit 1 examinations were to be completed in U1C23 (Fall 2009). The contract documents have been executed. Equipment development and qualification is in progress. The project plan has the examinations conducted first to meet the commitment, followed by the mitigation activities. Several sets of contingency actions are planned to ensure the commitment is satisfied.

Revised Schedule: Unit 1 has been in a forced outage since September 20, 2008. This was caused by a main turbine failure. The reactor has been and is currently operating in Mode 5 with the reactor coolant system depressurized. There are currently no forced outage scenarios that would require CNP to defuel the reactor. Repair scenarios include straightening of the turbine rotors, which would support a unit restart in the fall of 2009. Based on these developments, the previously scheduled U1C23 outage date has been moved from the fall of 2009 to the spring of 2010.

CNP Unit 1 hot and cold leg welds are not configured for a qualified exterior examination. The nozzle configuration, especially the overall thickness, would make an OD examination very difficult. Significant machining and surface preparation would be necessary for an exterior or OD examination. The exam would have to be single sided which does not meet the greater than 90% PDI coverage requirement. The ability to perform machining would also impact minimum reinforcement of the nozzle. Additionally, the ice condenser plant design includes many missile and divider blocks that divide the upper containment from the lower containment volume. To access the reactor cavity, the missile blocks weighing several tons have to be moved. The current forced outage does not provide the plant configuration for removing the core barrel and performing the examination of the DMW from the interior or ID of the reactor nozzles. The missile block, head lift, fuel removal, and core barrel lift are all high risk activities that are associated with a refueling outage. The scheduled U1C23 refueling outage schedule contains those activities to perform the sequence discussed above in conjunction with the performance of the 10-year ISI reactor vessel examination. The current forced outage does not require refueling of the reactor. Therefore, entering a refueling sequence of work during this forced outage would constitute an unnecessary burden and nuclear safety risk.

CNP's U1C23 refueling outage date has been set for March 2, 2010. This represents an approximate three month deviation from the MRP-139 guideline date of December 31, 2009. It is CNP's intention to inspect and mitigate, not only the Unit 1 Reactor Vessel Hot Leg Nozzle to Safe End welds but also the Unit 1 Reactor Vessel Cold Leg Nozzle to Safe End welds. The justification for the deviation is as follows:

- Unit 1 is currently in Mode 5 with no plans to defuel the reactor.
- The current operating condition does not provide the temperature requirements to support PWSCC as an active degradation mechanism. This conclusion is based on several MRP documents that show the crack growth slows significantly as the temperature gets below 600°F. CNP normal operating conditions are Cold Leg temperatures at 520°F and Hot Leg temperatures at

590°F. These relatively low operating temperatures provide additional margin. Cold Shutdown conditions (since September 2008) have the RCS depressurized at a temperature of approximately 123°F.

- Restarting Unit 1 in October 2009 and operating until March 2, 2010 will equate to an operating period approximately 8 months shorter than the operating period originally scheduled (pre-turbine event). Unit 1 would have had 23.9 Effective Full Power Years (EFPY) through October 2009, the originally scheduled refueling outage. The EFPY is now projected to be 22.9 assuming a restart in October 2009 and a refueling outage in March 2010.

Unit 1 has been operating at a reduced temperature and pressure since June 1989 as part of the program to preserve the steam generators. The reduced temperature and pressure slows any potential crack growth. MRP-115, which is referenced by MRP-139, provides the basis for the crack growth rates. The MRP equation 4-5 is exponential as a function of temperature. As the temperature is lowered, the crack growth rate slows exponentially. This is discussed in detail in the Detailed Evaluation and Industry Safety Assessment Sections.

- Postponing the inspection activities until the U1C23 refueling outage would remove any need for an additional fuel handling campaign. This would in be in direct support of ALARA and safety principles.
- The U1C23 refueling outage also includes the ASME code required 10-year ISI Reactor Vessel examination.
- The ASME Boiler & Pressure Vessel Code Section XI Paragraph IWA-2430(e) allows the interval to be extended when a plant has been out of service for more than 6 months.

Reason for Evaluation/ Scope:

This document provides the basis for schedule deviation from the implementation requirements of Material Reliability Program (MRP) document MRP-139, "Primary System Piping Butt Weld Inspection and Evaluation Guideline." (Reference 1) The specific deviation being requested is the deferral of ultrasonic (UT) inspections of Cook Nuclear Plant Unit 1 Reactor Vessel Hot Leg Nozzle to Safe End welds until the U1C23 refueling outage. Unit 2 Reactor Vessel Hot and Cold Leg Nozzle to Safe End welds are not in this population, because they are fabricated with non-susceptible materials (Stainless Steel) (Reference 16 & 21). Also no other Reactor Coolant System butt welds are in the population because they are stainless steel (Reference 21) or have weld overlays applied. The U1C23 refueling outage has been rescheduled due to the failure of the Unit 1 Main Turbine. Repair scenarios indicate a return to service date of Fall 2009. The U1C23 refueling outage is now scheduled for March of 2010. All 8 legs (Hot and Cold Legs) will be ultrasonically inspected. The schedule deferral (for a March 2010 refueling outage) represents an approximate three month deviation for the reactor vessel hot legs and no required deviation for the reactor vessel cold legs. Unit 1 would have had 23.9 Effective Full Power Years (EFPY) through October 2009, the originally scheduled refueling outage. The EFPY is now projected to be 22.9 assuming a March 2010 refueling. Unit 1 has been operating at reduced temperature and pressure since June 1989 as part of the program to preserve the steam generators. Amendment No. 126 to Operating License No. DPR-58 for CNP Unit 1 permits the operation of the plant at reduced primary system temperature and pressure conditions. This reduced operating temperature would slow any potential crack growth (Reference 13). The MRP equation 4-5 is exponential as a function of temperature. As the temperature is lowered the crack growth rate slows exponentially (Reference 13). The BMV have been conducted every outage and no evidence of leakage has been found.

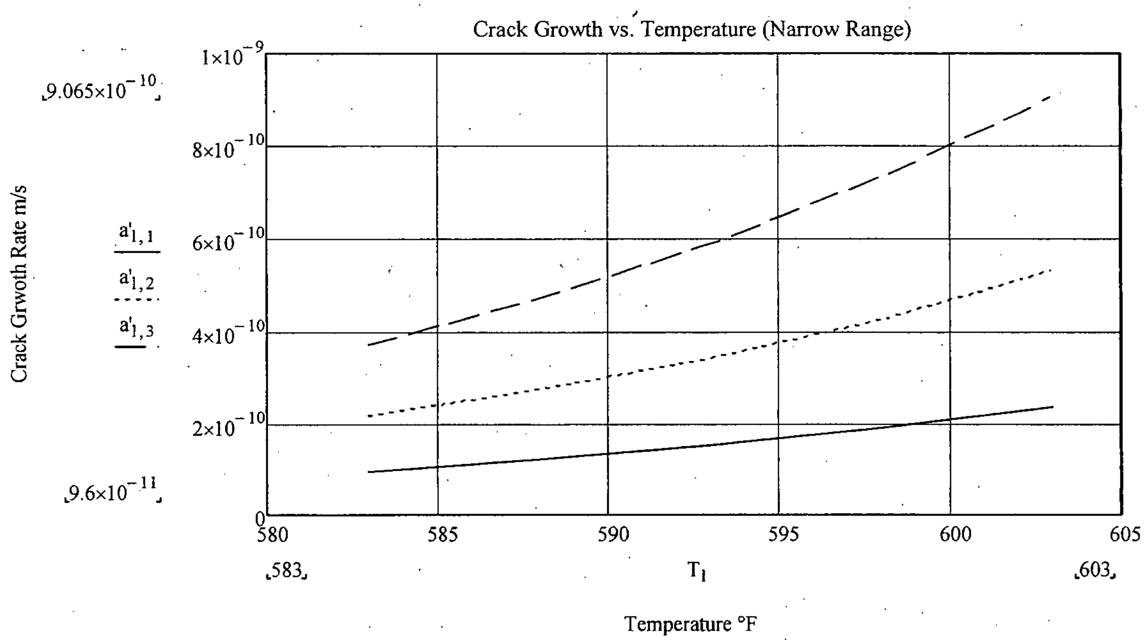
Detailed Evaluation

MRP-139 Section 1.2 states, "By December 31, 2009, Alloy 82/182 butt welds that are greater than 14" NPS and exposed to temperatures equivalent to the hot leg will be volumetrically examined per this guideline. By December 31, 2010, Alloy 82/182 butt welds that are greater than 4 inches NPS and exposed to temperatures equivalent to the cold leg will be volumetrically examined per this guideline." Section 6.8.2 states "Owners who know that their welds cannot be volumetrically inspected are not required to perform a best effort NDE; however, by the time the examination is due, they are required to have an approved Deviation in place including a plan to address either the susceptibility of the weld or the inspectability of the weld. Actions identified in this plan will be performed at the earliest possible RFO."

Due to the Cook Unit 1 turbine failure related forced outage and subsequent rescheduling U1C23 refueling outage, the mandated requirements of MRP-139 will not be met (Reference 3).

The unit is in cold shutdown and since Primary Water Stress Corrosion Cracking is temperature dependent, cracking is not postulated to grow or initiate (Reference 13). There is no physical impact on the plant as a result of not meeting the mandatory requirements.

Unit 1 has been operating at reduced temperature and pressure since June 1989 as part of the program to preserve the steam generators. The conclusion is based on several MRP documents, including MRP-115, that show the crack growth slows significantly as the temperature gets below 600°F. CNP normal operating conditions are Cold Leg temperatures at 520°F and Hot Leg temperatures at 590°F. These relatively low operating temperatures provide additional margin as shown below:



a'_1 (m/sec) uses $K_I = 15$ MPa$\sqrt{m}$
a'_2 (m/sec) uses $K_I = 25$ MPa$\sqrt{m}$
a'_3 (m/sec) uses $K_I = 35$ MPa$\sqrt{m}$

At the higher stress intensities the crack growth rate almost doubles when the temperature increases from 590°F to 600°F.

Reactor Vessel Hot and Cold Leg Nozzle to Safe End Welds
Original Schedule (pre-September 2008): The Unit 1 examinations will be completed in U1C23 (Fall 2009). The procurement of services, outage planning, engineering and mockup activities are in progress. The project plan had the examinations conducted first to meet the commitment, followed by the mitigation activities that included baseline UT examination. Several sets of contingency actions are planned to ensure the commitment is satisfied.

Revised Schedule: Unit 1 has been in a forced outage since September 20, 2008. This was caused by a main turbine failure (Reference 3). The reactor has been and is currently operating in Mode 5 with the reactor coolant system depressurized. There are currently no forced outage scenarios that would require CNP to defuel the reactor. Repair scenarios include repairing the turbine rotors which would support a unit restart in the fall of 2009. Based on these developments, the previously scheduled U1C23 outage date has been moved from the fall of 2009 to spring of 2010. Inspection and Mitigation activities (Reference 5) are continuing to be pursued for completion during the U1C23 refueling outage in March 2010.

The Unit 1 Reactor Vessel Hot Leg Nozzle to Safe End welds have also been evaluated for leak before break (LBB) (Reference 17 &18). PWSCC was not considered during the licensing of the main loop piping for LBB. These welds have been examined ultrasonically from the ID as part of the 10 year vessel examination using qualified personnel and procedures and no flaws were reported. This examination was conducted in October 1995 (Reference 14). The next inspection will be conducted in U1C23 currently scheduled for March 2010 (Reference 4). These nozzles have had bare metal visual examinations each refueling outage.

In summary, CNP's previously recognized aggressive approach to mitigation of this material issue has resulted in an industry leading program. The events surrounding the turbine failure of CNP Unit 1 has caused a delay in the completion of the mitigation activities associated with the Unit 1 Reactor Vessel Hot Leg Nozzle to Safe End welds. CNP Unit 1 has been and continues to be operating in Mode 5 with the reactor coolant system depressurized. This operating condition does not provide the temperature requirements to support PWSCC as an active degradation mechanism (Reference 13). Therefore, CNP is pursuing a deviation from the requirement that calls for all Alloy 82/182 butt welds that are greater than 14" NPS and exposed to temperatures equivalent to the hot leg will be volumetrically inspected per this guideline by December 31, 2009.

Justification Basis:

Guidance:

Nuclear Energy Institute (NEI) document NEI 03-08 (Reference 2), "Guideline for the Management of Materials Issues" allows deviations with the appropriate justification and documentation. Addendum E, Section 7, "Deviations" of this document states, " When a utility determines that "Mandatory" or "Needed" work product elements will not be fully implemented or will not be implemented in a manner consistent with their intent, or when a work product will not be implemented in the timeframe specified by the responsible (Industry Materials Issue Program) IP, a technical justification for deviation shall be developed and retained with the owner's program documentation or owner-controlled tracking systems. In addition, deviations from "Mandatory" and "Needed" work product elements will be entered into corrective action programs (CAP). The technical justification shall provide the basis for determining that the proposed deviation meets the same objective and intent, or level of conservatism exhibited by the original work product, and shall clearly state how long the deviation will be in effect. Deviations from "Mandatory" and "Needed" elements shall receive final concurrence by knowledgeable materials expert independent of the utility "justifying the deviation".

As mentioned in MRP-139 Section 6.8.2, owners are required to have a plan in place to address either the susceptibility or the inspectability of the Alloy 82/182 welds. The outage has been moved from the fall of 2009 to spring of 2010 due to an unrelated turbine forced outage (Reference 3).

Industry Safety Assessment:

CNP's original plans included operating Unit 1 for the entire fuel cycle (18 months). The current plan will result in Unit 1 operating for some period less than a full operating cycle (approximately 11 months) (Reference 19). In addition, while Unit 1 is in a forced outage, due to the turbine failure, the unit is in cold shutdown with the RCS depressurized. Thus, removing the temperature requirements that PWSCC depends on for cracking to either grow or initiate (Reference 13).

Guidance:
From MRP-139 Section 5.1.7 A) Perform a bare metal visual examination prior to the required volumetric exam completion date and B) repeat at the frequency defined in Table 6-2 of MRP-139. C) Local leak monitoring should be considered where access to visual exams is limited.

Cook Plant Actions:

A) Bare metal visual examinations were performed in the refueling outage prior to the required completion date. This was in the spring of 2008 (U1C22).
B) Visual exams (VT-2) and bare metal visual examinations are performed each refueling outage until ultrasonic examinations are completed.
C) Leak monitoring is conducted in accordance with the Cook Surveillance Procedure 1-OHP-4030-102-016 "Reactor Coolant System Leak Rate Test" (Reference 8) meets the latest Westinghouse guidance (References 11 & 12) for conducting reactor coolant system (RCS) leak rate testing.

Guidance:
Perform a degradation assessment in accordance with the flaw evaluation methodology of Section 7 of MRP-139 for any portion of the required volume that remains unexamined. This assessment should demonstrate reasonable assurance to the licensee that either:

- An assumed flaw will not grow to a critical flaw prior to establishing full examination compliance or
- Plant leakage detection capabilities can reliably detect leakage from the subject location and support timely initiation of the necessary plant actions.

Cook Plant Actions:
Assessment of Critical Flaw Size
The critical flaw size is assumed to be the flaw size at the end of the required examination period. The assumed flaw size is the maximum permitted flaw that would occur over the examination interval. The flaw growth to the critical size is dependent on several factors including stress intensity, temperature, and material composition. The stress intensity is determined by the applied loads from operating pressure and temperature, deadweight, thermal loadings, and other applied loads. The crack growth rates are temperature dependent. The weld material in the Cook Plant Hot Legs is Alloy 82.

The flaw evaluation is a sum of the crack growth lengths over the time of interest. To simplify the discussion, the last cycle will be defined as from the previous refueling outage to the refueling outage before the end of the required examination period. This will be compared to the proposed change to the examination date. The crack growth rates are simplified using a constant growth and stress intensity for a specific temperature.

Using MRP-115 crack growth predictions, several analysis assumptions are used. The critical flaw size will be defined as 2.5 inch (0.0635 m) deep flaw. This is the design thickness of the RCS pipe. The operating stress intensity is defined as $K_i = 35\ MPa\sqrt{m}$. This is a typical high value given in the literature for stress intensity values for computing crack growth rates. Included in this value are the operating loads. The stress intensity during the shutdown cold conditions is defined as K_i = 15 MPa$\sqrt{m}$ which is a typical low value given in the literature. This stress intensity includes loads from deadweight and decay heat (RHR) pump operation.

At 590°F the crack growth rate would be approximately 5×10^{-9} m/s. Using 18 month operating cycle the flaw growth in the original cycle would have been approximately 0.87 inches (0.022 m). The change in flaw size (growth) would have been from 1.63 inches to 2.5 inches.

The crack growth is determined using the proposed examination date. The initial flaw size as determined above is 1.63 inches. The first and last part of the cycle the unit operated under the conditions assumed above. The time at full power is 7 months (April to September) until the turbine failure and 5 months until the March 2010 refueling outage (November to end of February). This is 1 EFPY or 23.9 EFPY – 22.9 EFPY. The crack growth in this period would be .58 inches (0.015m) using the same stress intensity and crack growth used above. The crack growth rate for the shutdown period of 9.87×10^{-13} m/s was determined using the minimal stress intensity and the lowest crack growth rate for which temperature data is available. The stress intensity value was chosen as a conservative representation which would include any reasonable residual stress, deadweight, and operational loads. These operational loads are RHR flows, testing, and pump swaps etc. The period will be 13 months (September 2008 to October 2009). The crack growth over this period is 0.0013 inches. Summing the crack growth and adding them to the calculated initial flaw gives 2.21 inches (1.63 inches initial flaw + 0.58 inches full power crack growth + 0.0013 inches shutdown crack growth). Therefore the flaw size is smaller due to the extended outage.

Plant Leakage
Cook Surveillance Procedure 1-OHP-4030-102-016 "Reactor Coolant System Leak Rate Test" (Reference 8) meets the latest Westinghouse guidance (References 11 & 12) for conducting reactor coolant system (RCS) leak rate testing. The procedure requires that a small increase in the RCS leak rate (0.05 GPM) be reported to plant management for action. This low threshold is designed to alert plant management of a potential PWSCC through wall leak.

Operating Experience
During the Salem Unit 1 Fall 2008 refueling outage (1R19), a significant flaw was identified in the 14 hot leg reactor pressure vessel (RPV) nozzle-to-safe end Alloy 600, Inconel 182 dissimilar metal (DM) weld (Reference 9). The circumferentially oriented, ID connected flaw was discovered during the pre-Mechanical Stress Improvement Process (MSIP) Ultrasonic (UT) examination. The flaw was analyzed and determined to be acceptable for applying MSIP as a form of stress improvement. The flaw was also analyzed in accordance with

ASME Section XI, IWB-3640 and would have allowed for 36 months additional plant operation without repair or mitigation. Based on this flaw and its analysis, the change in schedule for the Cook Plant of approximately 3 months is less than the 36 months of predicted growth at Salem Plant. Therefore, the Cook Plant is bounded.

H.B.Robinson performed an ID inspection of their three hot leg and three cold leg RV nozzles and shared the official results on an EPRI phone call (October 21, 2008). Axial flaws were detected in all three hot legs and two cold legs. None of the flaws were determined to be ID connected (PWSCC initiated). Eddy current was required to confirm the UT results. The largest flaw was on the "B" cold leg (0.6 to 1.3 "deep and 1 to 1.5" in length). Using their flaw handbook, they calculated they can operate for at least three years without repair. Nine (9) flaws were found on the hot legs and varied in depth from 0.36 to 0.51" and from 0.25 to 0.60" in length. A number of circumferential flaws were detected in the cladding.

They did not experience any issues with obtaining 100% coverage of the area to be inspected as their weld IDs were machined after application of the cladding during fabrication.

The extent of the "B" cold leg flaw would necessitate re-inspection for the next three outage cycles. They plan on pulling up their mitigation plans in lieu of constant re-inspections. They will pursue an ID mitigation since they are limited to three inches of OD clearance.

Details of the H.B.Robinson plant reactor vessel nozzle inspections are:

1. HB Robinson is a 3 Loop Westinghouse Unit, Reactor Vessel fabricated by Combustion Engineering. The 10 year ISI was scheduled for 2011, but to meet the inspection requirements of MRP-139, they decided to perform the RV Nozzle Dissimilar Welds now. These are RV Nozzle to Safe-End welds.

2 Discovered flaws in all 3 hot legs (HL) and in one cold leg (CL). Thickness of pipe, HL= 2.6 inches and CL = 2.5 inches.

3 Hot Legs, Loop A - 1 flaw; Loop B had 4 flaws; and Loop C had 4 flaws. Sizes vary Length= 0.25 in to 0.51 inch, and depth= 0.36 to 0.6 inches. The Hot leg welds were not buttered.

4 Cold leg Loop B- One flaw, Length 0.6 inch to 1.5 inches, depending on the angle of view. Cold Leg Welds are buttered. Cold leg weld has gone thru several repairs during initial fabrication as well as several post weld heat treatments. Utility considers all flaws

are fabrication flaws. Last inspection of these welds was in 2001. Since then some of the flaws that were observed in 2001 have appeared to have grown in size.

5 Inspection was by both UT and Eddy Current from the inside wall. UT inspection was unable to tell if these were inside surface connected. Therefore Eddy Current was performed. Inside surface is smooth and no coverage issues were encountered.

6 Several axial and circumferential indications have been identified in the cladding.

All flaws are embedded flaws. However, when measurement uncertainty (0.39 inches) is added to both ends of the flaw, they have to be treated as surface flaws (inside surface). Based on these flaws, the schedule change for the Cook plant is bounded.

At the Mihama (Reference 10) plant (Japan) shot-peening work to reduce residual stress in the surface of the steam generator (S/G) reactor coolant inlet and outlet nozzle welds (total four welds) where 600 series Ni base alloy had been used was to be performed as a preventive maintenance task. To check the surface conditions of the nozzle weld portions prior to the work, eddy current test (ECT) was performed for the S/G nozzle weld portion surfaces and significant indications were identified for 13 locations on the inlet nozzle weld portion of S/G-A. Also, visual checking of these locations having significant signals was performed and a flaw was identified at one location. For the outlet nozzle of S/G-A and the inlet and outlet nozzles of S/G-B, no significant signal was identified. Further, for the locations found with significant ECT indication, a penetrant test (PT) was performed and as a result significant penetration indicating patterns (maximum length: approx. 17 mm) were identified. In addition an ultrasonic test (UT) was performed and thereby the maximum flaw depth was evaluated at approx. 13 mm (for the location of the maximum length), resulting in the residual thickness of the affected location (approx. 68 mm) evaluated to be below the thickness described (75 mm) in the application for construction permit based on the Electricity Utilities Industry Law (Reference 10). The significance of the Japanese experience is that it shows typical cracking in the axial direction. The axial cracks run along the pipe and are not safety significant. The cracks arrest at the nozzle and safe end wall. The overall pipe integrity is maintained while minor leakage is possible, but well within the plant equipments capability to safely shutdown. These flaws were fairly shallow and significantly smaller than the assumed flaw used in the critical flaw size assessment.

V.C. Summer Plant had a through wall leak that was discovered by visual examination (OE395-001007-1). The leak was in an area that had been repaired several times during original fabrication. Station personnel also noted water weeping from a 3/16-inch hole in the weld near the top of the hot leg pipe. Video camera inspection on the inside of the reactor vessel nozzle-to-hot leg weld revealed a line that initially was believed to be a circumferential crack. However, ultrasonic testing of the inner weld did not reveal any indications of circumferential cracking. The crack seen earlier with the video camera was apparently a shadow created by the surface contours of the pipe and weld. On November 8, ultrasonic and eddy current testing, and video examination inside the hot leg pipe identified a 2.7-inch long hairline axial crack on the inner surface of the weld. These tests also indicated that the crack was through-wall, and aligned with the previously identified 3/16-inch hole in the weld near the top of the pipe. For the Cook Plant the flaw size is consistent to the flaw assumed in the discussion. The bare metal visual examinations each outage and the enhanced leak detection procedures will detect this flaw.

Leak Detection Capability:

Cook Surveillance Procedure 1-OHP-4030-102-016 "Reactor Coolant System Leak Rate Test" (Reference 8) meets the latest Westinghouse guidance (References 11 & 12) for conducting reactor coolant system (RCS) leak rate testing. The procedure requires that a small increase in the RCS leak rate (0.05 GPM) be reported to plant management for action. This low threshold is designed to alert plant management of a potential PWSCC through wall leak.

Inspection Limitations:

Profiling of the weld and adjoining base metal contours of the Hot Leg Nozzles shows 100% inspectable surface from the interior (Reference 14). These welds were made in a factory (Reference 7). The welding controls in a factory or shop setting generally produce higher quality welds than field welds. These welds have no recorded repairs in the area of interest. This is based on a review of receiving documentation. The materials of construction were documented in Westinghouse Letter Report (Reference 7). No abnormalities in construction of the reactor vessel were reported in the area of interest.

The ice condenser plant design includes many missile and divider blocks that divide the upper containment from the lower containment volume (Reference 15). To access the reactor cavity, the missile blocks weighing several tons have to be moved. The current forced outage does not provide the plant configuration for removing the core barrel and performing the examination of the DMW from the interior or ID of the reactor nozzles. The missile block, head lift, fuel removal, and core barrel lift are all high risk activities. The hot and cold leg welds are not configured for an exterior examination. Significant machining and surface preparation would be necessary for an exterior or OD examination (Reference 7). The nozzle configuration especially the overall thickness would make an OD examination difficult. The exam would have to be single sided which does not meet the greater than 90% coverage requirement. The ability to perform machining would also impact minimum reinforcement of the nozzle. The increase to plant risk for these extra activities is not justified for the compliance with the MRP-139 examination requirement to inspect the hot legs, when considering the actual EFPY.

Based on the above, the low operating temperatures, industry operating experience, enhanced leak detection, bare metal visual examinations, construction record review, and the net reduction in the EFPY support the extension of the required examination date by approximately three months.

Duration of Deviation:

CNP is pursuing a deviation from the requirement that calls for all Alloy 82/182 butt welds that are greater than 14" NPS and exposed to temperatures equivalent to the hot leg to be volumetrically inspected per the MRP-139 guideline by December 31, 2009. Cook Unit 1 Turbine repairs are scheduled to be completed by October 2009. Inspection and mitigation activities are continuing to be pursued for completion during the U1C23 refueling outage in March 2010.

Conclusions/Findings:

Due to Cook Unit 1 operating in cold shutdown with the RCS system depressurized, the temperature dependency of PWSCC to initiate or grow is eliminated. Additionally, the overall length of operational time is reduced in comparison to a normal operational cycle. The compensatory measures meet the requirements of MRP-139. These include bare metal visual examinations and enhanced leak detection. The degradation assessment shows the flaw will not reach critical size during the extended period. Therefore, the low operating temperatures, industry operating experience, enhanced leak detection, bare metal visual examinations, construction record review, and the net reduction in the EFPY support the deferral of the MRP-139 ultrasonic (UT) inspections of Cook Nuclear Plant Unit 1 Reactor Vessel Hot and Cold Leg Nozzle to Safe End welds until the U1C23 refueling outage is technically justified. The deviation is to allow the hot legs welds in Unit 1 to be examined in March 2010 vice December 2009.

References:

1. Materials Reliability Program: Primary System Piping Butt Weld Inspection and Evaluation Guidelines (MRP-139), EPRI, Palo Alto, CA: 2005. 1010087.

2. Issue of Revision 3 to Materials Guideline Implementation Protocol, NEI 03-08 Addendum E.
3. CR 00838732, Perform RCE on Unit 1 turbine failure
4. RFP 10014 Reactor Vessel Ten Year Examination
5. RFP 10015 RPV Hot & Cold Leg Nozzle Mitigation
6. MRP 2007-0040, Spring 07 Lessons Learned for Ultrasonic Examinations of Dissimilar Metal Weld
7. Westinghouse Letter Report LTR-PCAM-07-74, D.C.Cook Units 1&2 RV Nozzle Welds, Task 2 & 3 Deliverables
8. Operating Department Procedure, Reactor Coolant System Leak Rate Test, 1-OHP-4030-102-016, Rev.19
9. OE27894 - Flaw Detected in Reactor Pressure Vessel Nozzle-to-Safe End Weld Prior to Mechanical Stress Improvement Process Application (Salem)
10. EAR TYO 08-005; Flaws Found in Steam Generator A Reactor Coolant Inlet Nozzle Weld (25.September.2007, Mihama 2, KANSAI)
11. WCAP-16423-NP, Pressurized Water Reactor Owners Group, Standard Process and Methods for Calculating RCS
12. WCAP-16465-NP, Pressurized Water Reactor Owners Group, Standard RCS Leakage Action Levels and Response Guidelines for Pressurized Water Reactors Leak Rate for Pressurized Water Reactors
13. Materials Reliability Program Crack Growth Rates for Evaluating Primary Water Stress Corrosion Cracking (PWSCC) of Alloy 82, 182, and 132 Welds (MRP-115), EPRI, Palo Alto, CA: 2004, 1006696.
14. 1995 Inservice Examination of Selected Components at the Donald C. Cook Nuclear Plant, Unit 1, Final Report, SwRI Project 7184, October 1995
15. Indiana And Michigan Power D. C. Cook Nuclear Plant Updated Final Safety Analysis Report, Revision 22, Chapter 5, Containment System
16. Indiana And Michigan Power D. C. Cook Nuclear Plant Updated Final Safety Analysis Report, Revision 22, Chapter 4, Table 4.2-1, Materials of Construction of The Reactor Coolant System Components
17. Safety Evaluation Report, Amendment 126, June 9, 1989
18. Design Basis Document For The Reactor Coolant System, DB- 12-RCS, Rev.3
19. Design Information Transmittal DIT-S-00705-10, Unit 1 & 2 Burn-up Data, March 26, 2008.
20. ISI Program Basis Document, Third Ten Year Inservice Inspection Interval, D.C.Cook, Units 1& 2
21. WCAP-16198-P, July 2004, Revision 1 PWSCC Susceptibility Assessment of the Alloy 600 and Alloy 82/182 Components in D. C. Cook Units 1 and 2

Extent of Condition:
This applies to the Unit 1 Reactor vessel nozzles only. Unit 2 reactor vessel nozzles are not fabricated with susceptible materials and are not in the scope of MRP-139. The pressurizer nozzles on both units have been mitigated and comply with MRP-139.

<James.Cirilli@exeloncorp .com>

09/10/2009 01:09 PM

To <rgpickard@aep.com>

cc <crlane1@aep.com>, <richard.hall@exeloncorp.com>

bcc

Subject RE: Fw: Cook Plant deviation

History: This message has been replied to.

I am ok with the revised deviation. Since I will be out of pocket starting tomorrow through Oct 9th, I am ok with AEP signing for me via this e-mail approval provided no further changes to the deviation are made. I suggest you work through my boss Rich Hall should any changes to the deviation be made.

Additionally, as I mentioned in my approval of the previous revisions, my signature represents my opinion that the technical justification is adequate, but it does not reflect my or Exelon's endorsement of the deviation request.

Please send me and Rich Hall a copy of the final approved deviation.

Thanks,
Jim Cirilli

-----Original Message-----
From: rgpickard@aep.com [mailto:rgpickard@aep.com]
Sent: Thursday, September 10, 2009 10:53 AM
To: Cirilli, James J.:(GenCo-Nuc)
Cc: crlane1@aep.com
Subject: Re: Fw: Cook Plant deviation

Mr. Cirilli,

Thank you for your timely review of our revised MRP-139 deviation. I revised the document based on one of your two comments. The second comment appeared to be more of a question: "Isn't the entire weld the 'area of interest'?" You are correct, of course. The entire dissimilar metal weld is the area of interest. I believe Paul wrote it up this way because there were no repairs on the weld or in the vicinity of the weld. If you would like for me to change the wording, I can do so.

The revised document is attached below. Thanks.