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Cook Nuclear Plant
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January 31, 2012

AEP-NRC-2012-4
10 CFR 50.4

Docket Nos.: 50-315
50-316

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

Donald C. Cook Nuclear Plant Units 1 and 2
ALLOY 600 AGING MANAGEMENT PROGRAM
RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

References:

1. Letter from J. P. Gebbie, Indiana Michigan Power Company, to U. S. Nuclear Regulatory Commission Document Control Desk, "Donald C. Cook Nuclear Plant Units 1 and 2, License Renewal Submittal – Alloy 600 Aging Management Program," AEP-NRC-2011-39, dated August 17, 2011, Agencywide Documents Access and Management System (ADAMS) Accession Number ML11238A069

2. Memorandum from P. S. Tam, U. S. Nuclear Regulatory Commission, to H. L. Etheridge, Indiana Michigan Power Company, "D. C. Cook Units 1 and 2, Draft RAI on the Alloy 600 Aging Management Program (TAC Nos. ME6882 and ME6883)," dated October 27, 2011, ADAMS Accession Number ML113000267

3. Memorandum from P. S. Tam, U. S. Nuclear Regulatory Commission, to H. L. Etheridge, Indiana Michigan Power Company, "Revised D. C. Cook Units 1 and 2 RAI on the Alloy 600 Aging Management Program (TAC ME6882 and ME6883)," dated November 3, 2011, ADAMS Accession Number ML11307A485

By Reference 1, Indiana Michigan Power Company (I&M), the licensee for Donald C. Cook Nuclear Plant (CNP) Units 1 and 2, submitted the Alloy 600 Aging Management Program for Nuclear Regulatory Commission (NRC) review and approval. By Reference 2, the NRC communicated a draft Request for Additional Information (RAI) to I&M for information needed to complete the review and approval of the Alloy 600 Aging Management Program.

A104
A047
NRR

On November 3, 2011, a teleconference was conducted between NRC staff and I&M staff to discuss Reference 2. Based on the discussion, Reference 2 was revised as Reference 3. At the conclusion of the teleconference, the due date of February 3, 2012, was established for the response to the RAI.

I&M's response to the RAI is provided as Enclosure 1 to this letter. A revised copy of the Alloy 600 Aging Management Program is provided as Enclosure 2 to assist in your review.

There are no new or revised regulatory commitments made as a part of this submittal.

Should you have any questions, please contact Mr. Michael K. Scarpello, Regulatory Affairs Manager, at (269) 466-2649.

Sincerely,

Joel P. Gebbie

Joel P. Gebbie
Site Vice President

SJM/jen

Enclosures:

1. Response to Request for Additional Information on the Alloy 600 Aging Management Program
2. Alloy 600 Aging Management Program

c: J. T. King, MPSC
S. M. Krawec, AEP Ft. Wayne, w/o enclosures
MDEQ – WHMD/RPS
NRC Resident Inspector
C. D. Pederson, NRC Region III
P. S. Tam, NRC Washington, DC

ALLOY 600 AGING MANAGEMENT PROGRAM
RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

By letter dated August 17, 2011, Indiana Michigan Power Company (I&M), the licensee for Donald C. Cook Nuclear Plant (CNP) Units 1 and 2, submitted the Alloy 600 Aging Management Program (AMP) for Nuclear Regulatory Commission (NRC) review and approval (Accession No. ML11238A069). The NRC staff reviewed the proposed Alloy 600 AMP in accordance with Chapter XI.M11B, Cracking of Nickel-alloy Components and Loss of Material Due to Boric Acid-Induced Corrosion in Reactor Coolant Pressure Boundary Components (PWRs Only)" in NUREG-1801, Revision 2, Generic Aging Lessons Learned (GALL) Report. By e-mail dated November 3, 2011, the NRC communicated a Request for Additional Information (RAI) for information needed to complete their review and approval of the proposed Alloy 600 AMP. I&M's response to the RAI is provided below. As a separate enclosure to this letter (Enclosure 2), I&M has provided a revised copy of the Alloy 600 AMP to assist the NRC in their review.

NRC Item 1

To manage Alloy 600 components, the Program Description section of XI.M11B in NUREG-1801, Revision 2, references American Society of Mechanical Engineers (ASME) Code Case N-729, "Alternative Examination Requirements for PWR Reactor Vessel Upper Heads With Nozzles Having Pressure-Retaining Partial-Penetration Welds, Section XI, Division 1," which is part of the requirements in Title 10, Code of Federal Regulations, Paragraph 50.55a (10 CFR 50.55a). However, Section 1.7 of the proposed Alloy 600 AMP does not reference this code case. Although the licensee has replaced the Units 1 and 2 reactor vessel head with Alloy 690/52/152 material, 10 CFR 50.55a (g)(6)(ii)(D) still applies and thus should be referenced.

(a) The licensee needs to cite this code case and 10 CFR 50.55a(g)(6)(ii)(D) or justify why it is not referenced in Section 1.7.

(b) Section 1.7 identifies ASME Code Cases N-722-1 and N-770-1. The NRC has imposed conditions on both code cases in 10 CFR 50.55a(g)(6)(ii)(E) and 10 CFR 50.55a(g)(6)(ii)(F), respectively. Section 1.7 also needs to include the reference to these two NRC regulations. As a general rule, whenever these three ASME code cases are mentioned in a regulatory document, they should be followed with a reference to the NRC regulations in 10 CFR 50.55a(g)(6)(ii)(D), (E), or (F) as applicable

I&M Response to NRC Item 1

(a) By letter dated August 11, 2004, I&M made the following commitment related to license renewal (Accession No. ML042470410):

> "I&M will continue to participate in industry initiatives, such as the Westinghouse Owners Group and the EPRI MRP. Susceptibility rankings and program inspection requirements regarding Alloy 82/182 pipe butt welds will be consistent with the later version of the EPRI MRP safety assessment or its successors."

Section 1.7 of the proposed Alloy 600 AMP was added to specifically address this commitment. ASME Code Case N-729-1 and 10 CFR 50.55a(g)(6)(ii)(D) address examination requirements for reactor vessel upper heads with nozzles having pressure-retaining partial-penetration welds. Since this code case does not specifically apply to Alloy 82/182 pipe butt welds, it is not included in Section 1.7 of the proposed Alloy 600 AMP.

ASME Code Case N-729-1 is listed as an inspection requirement in Attachment 2 of the proposed Alloy 600 AMP. I&M has revised Attachment 2 of the proposed Alloy 600 AMP to also include a reference to 10 CFR 50.55a(g)(6)(ii)(D).

(b) I&M has revised Section 1.7 of the proposed Alloy 600 AMP. This section now includes references to 10 CFR 50.55a(g)(6)(ii)(E) and 10 CFR 50.55a(g)(6)(ii)(F).

NRC Item 2

Attribute Item 1 of XI.M11B in NUREG-1801, Revision 2, provides the generic scope of an Alloy 600 AMP. Section 3.5.4 of the proposed AMP states that "...[d]etermine appropriate sample locations, inspection techniques, and acceptance standards in accordance with industry guidelines..."

(a) In addition to industry guidelines, Section 3.5.4 of the proposed AMP needs to state that the sample locations, inspection techniques and acceptance standards are also determined in accordance with 10 CFR 50.55a and relevant NRC generic communications or justify why NRC regulations are not included.

(b) Discuss the inspection samples for the components identified in Attachment 2 (e.g., how many welds and/or components will be inspected) during each inspection.

(c) Deleted.

I&M Response to NRC Item 2

(a) I&M has revised Section 3.5.4 of the proposed Alloy 600 AMP to state that "...sample locations, inspection techniques and acceptance standards are determined in accordance with industry guidelines, NRC regulations (10 CFR 50.55a), and relevant NRC generic communications."

(b) Attachment 2 of the proposed Alloy 600 AMP provides a list of inspection requirements for components in the Alloy 600 Material Management Program. For the inspections that are required by 10 CFR 50.55a, the inspection samples (e.g., how many welds and/or components will be inspected) are determined in accordance with the requirements provided in 10 CFR 50.55a (e.g., ASME Boiler & Pressure Vessel Code, Section XI, or ASME Boiler & Pressure Vessel Code Case). The required inspection samples for Steam Generator tube inspections are provided in the approved Technical Specifications (TS) for CNP.

NRC Item 3

Attribute Item 3 in XI.M11B in NUREG-1801, Revision 2, of the GALL report specifies parameters that are to be monitored or inspected.

(a) Deleted.

(b) Attachment 2 specifies "Visual" as the inspection method for various steam generator components such as divider plate, tubesheet cladding, manway diaphragm, nozzle dam rings and shell cladding. Discuss whether the visual inspection is equivalent to ASME Code, Section XI, IWA-2210, visual examination category VT-1, VT-2, or VT-3. If not, explain what the visual inspection is (e.g., what is being examined and what are acceptance criteria).

(c) Deleted.

I&M Response to NRC Item 3

(b) Attachment 2 of the proposed Alloy 600 AMP lists several visual inspections that are performed on steam generator components. These inspections are not equivalent to any ASME Boiler & Pressure Vessel Code examinations. These inspections consist of a visual scan of the listed component for any signs of degradation or cracking. The acceptance criteria for these inspections are met if no signs of degradation or cracking are observed.

NRC Item 4

Attribute Item 4 in NUREG-1801, Revision 2, Detection of Aging Effects, states that "...Reactor coolant pressure boundary leakage can be monitored through the use of radiation air monitoring and other general area radiation monitoring, and technical specifications for reactor coolant pressure boundary leakage..." Discuss the capability of the RCS leakage detection systems in Units 1 and 2 with respect to Regulatory Guide 1.45, Revision 1, "Guidance on Monitoring and Responding to Reactor Coolant System Leakage."

I&M Response to NRC Item 4

The construction permits for CNP were issued and the majority of construction was completed prior to issuance of 10 CFR 50, Appendix A, General Design Criteria, in 1971 by the Atomic Energy Commission (AEC). CNP was designed and constructed to comply with the AEC General Design Criteria (GDC) as proposed on July 10, 1967. The application of the AEC proposed General Design Criteria to CNP is contained in the CNP UFSAR as the Plant Specific Design Criteria (PSDC). Appendix A of 10 CFR 50 GDC differs both in numbering and content from the PSDC for CNP.

PSDC 16, Monitoring Reactor Coolant Leakage, describes the means that are provided to detect significant uncontrolled leakage from the reactor coolant pressure boundary. This requirement meets the intent of General Design Criterion 30, which requires that means be provided for detecting and, to the extent practical, identifying the location of the source of reactor coolant leakage.

CNP is not committed to RG 1.45, Reactor Coolant Pressure Boundary Leakage Detection Systems. However, the requirements of RG 1.45 were followed to the extent practical in the design of the leakage monitoring systems. A detailed discussion of RCS leakage monitoring is provided in CNP UFSAR Section 4.2.7, "Leakage."

The Unit 1 containment atmosphere particulate radioactivity monitor has a licensing basis leak detection capability of 0.8 gpm in 1 hour. The Unit 1 containment atmosphere gaseous radioactivity monitor has a licensing basis leak detection capability of 1 gpm in 4 hours. The Unit 2 containment atmosphere particulate and gaseous radioactivity monitors have a licensing basis leak detection capability of 1 gpm in 4 hours.

CNP's leakage detection capability was most recently reviewed and approved by the NRC in the issuance of License Amendments 317 and 300 for Unit 1 and Unit 2 respectively, by Safety Evaluation dated November 1, 2011 (ADAMS Accession No. ML11249A090). These amendments revised TS 3.4.15, RCS Leakage Detection Instrumentation, and reviewed the associated TS Bases pages consistent with Revision 3 of Technical Specification Task Force (TSTF) Standard Technical Specification Traveler, TSTF-513, "Revise PWR Operability Requirements and Actions for RCS Leakage Instrumentation." The changes to the TS Bases pages, as a result of these approved amendments, incorporated CNP's approved license basis leakage detection capabilities.

NRC Item 5 (Deleted)

NRC Item 6

Attribute Item 6 in NUREG-1801, Revision 2, specifies the acceptance criteria for all indications of cracking and loss of material due to boric acid-induced corrosion. Section 4.7.1 of the proposed AMP states that "...[i]nspection results that do not meet the acceptance criteria are documented via PMP-7030-CAP-001, Action Initiation..." The acceptance criteria are not clearly defined in the proposed AMP. The proposed AMP needs to include a section on acceptance criteria for the inspection results. The acceptance criteria section should include the individual criterion for the inspected components as defined in Attachment 2.

I&M Response to NRC Item 6

I&M has revised the proposed Alloy 600 AMP to include a section on acceptance criteria (Section 4.7). Attachment 2 of the proposed Alloy 600 AMP lists the specific acceptance criteria for each inspection.

NRC Item 7

Attribute Item 7 of XI.M11B in NUREG-1801, Revision 2, Corrective Actions, specifies that relevant flaw indications of susceptible components found to be unacceptable for further services are corrected through implementation of appropriate repair or replacement as dictated by 10 CFR 50.55a and industry guidelines (e.g., MRP-139). In addition, detection of leakage or evidence of cracking in susceptible components require scope expansion of current inspection and increased inspection frequencies of some components, as required by 10 CFR 50.55a and industry guidelines (e.g., MRP-139). Section 4.7.5 of the proposed AMP states that "...based on the initial inspection results, the need for additional inspections are determined. This information is used to develop future inspection scope and associated inspection intervals. Subsequent inspections may include inspections of the additional locations..." If the initial inspection discovers degradation in Alloy 600 components, discuss the scope and intervals of the additional inspection with respect to the ASME Code, Section XI requirements.

I&M Response to NRC Item 7

Section 4.7.5 of the proposed Alloy 600 AMP was deleted and Attachment 2 was revised to provide a list of inspection requirements for components in the Alloy 600 Material Management Program. As required by 10 CFR 50.55a, several of these inspections are conducted in accordance with ASME Boiler and Pressure Vessel Code (Section XI or applicable Code Case). I&M evaluates the results of these inspections in accordance with the applicable ASME Boiler and Pressure Vessel Code and 10 CFR 50.55a to determine if any additional inspections are required. If additional inspections are required, I&M implements those inspections in accordance with the applicable ASME Boiler and Pressure Vessel Code and 10 CFR 50.55a.

NRC Item 8

Attribute Item 8 of XI.M11B in NUREG-1801, Revision 2, Confirmation Process, and Attribute Item 9, Administrative Controls, references 10 CFR Part 50, Appendix B. Section 4.7.2 of the proposed AMP states that "...WHEN the test acceptance criteria are not met, THEN an Engineering Evaluation is performed in accordance with 10 CFR Part 50, Appendix B, and documented in accordance with PMP-7030-CAP-002, Condition Evaluation, Action, and Closure, in order to verify that the intended functions of the in-scope components can be maintained consistent with the current licensing basis. [Ref. 5.2.1a]..."

(a) Clarify why 10 CFR Part 50, Appendix B, is used as the basis for the engineering evaluation, in lieu of 10 CFR 50.55a which requires the use of the evaluation in the ASME Code, Section XI. Discuss the procedures that will be used to implement the requirements 10 CFR Part 50, Appendix B.

(b) Discuss whether the engineering evaluation would include corrective actions.

I&M Response to NRC Item 8

(a) I&M has revised the "Corrective Measures" section of the proposed Alloy 600 AMP. Section 4.8.2 of the proposed Alloy 600 AMP now states:

> "WHEN the acceptance criteria are not met, THEN an evaluation is performed in accordance with 10 CFR 50.55a, ASME Boiler and Pressure Vessel Code (Section XI or applicable Code Case), and industry guidelines."

Inspection results that do not meet the acceptance criteria are documented and addressed in accordance with the Corrective Action Program, which is compliant with Appendix B, Criterion XVI, Corrective Action. Corrective action procedures are implemented in accordance with the requirements of 10 CFR 50, Appendix B.

(b) As stated in Section 4.8 of the proposed Alloy 600 AMP, when acceptance criteria are not met, an evaluation is performed in accordance with 10 CFR 50.55a, ASME Boiler and Pressure Vessel Code (Section XI or applicable Code Case), and industry guidelines. The evaluation will determine if the component is acceptable for continued service and the corrective actions required (further evaluation, additional examinations, increased inspection frequency, repair/replacement, etc.).

Enclosure 2 to AEP-NRC-2012-4

ALLOY 600 AGING MANAGEMENT PROGRAM
Revision 4

Doc No.:	EHI-5070-ALLOY600	Rev No.: 004
Title:	ALLOY 600 MATERIAL MANAGEMENT PROGRAM	

Alteration Cat.: Minor Revision
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Approvals

Name	Review/Approval Type/Capacity	Date
Weir, Daniel	7 Approval Authority	12/30/2011 11:33
Kleimola, Daniel	3 Technical Review	12/30/2011 08:56
Kalinowski, Robert	1 Cross-Discipline Review	12/28/2011 13:11
Patterson, Kyle	1 Cross-Discipline Review	12/22/2011 12:55
Etheridge, Helen L	1 Cross-Discipline Review	12/22/2011 12:07
Hall, Roy E	1 Cross-Discipline Review	12/22/2011 09:06

Signature Comments

Approved by acting Programs Manager 2011-918
Noticed that there are no rev bars for the table although it changed. Also, rev summary only needs to be 1 page; second page is blank.
none
none.
None

Alloy 600 Material Management Program

Information		
Jeff Calderwood	Site Procedure Group	Engineering Programs
Writer	Document Owner	Cognizant Organization

TABLE OF CONTENTS

1 PURPOSE AND SCOPE

1.1 This document describes the overall programmatic requirements that Cook Nuclear Plant (CNP) will follow for the development, control, and implementation of an Alloy 600 Material Management Program for CNP Units 1 and 2.

1.2 This procedure describes the Alloy 600 Material Management Program that is required for license renewal [Ref. 5.2.1b, 5.2.1c and 5.2.1d]. This procedure also implements License Renewal Commitment No. 8244 [Ref. 5.2.1a]. License Renewal background documentation and NRC correspondence is to be considered when changing this procedure.

1.3 This Program will manage aging effects of both pressure and non-pressure boundary Reactor Coolant System components constructed of Alloy 600/690 and welds constructed of the associated weld metals, Alloy 82/182 and Alloy 52/152.

1.4 This Program was developed utilizing EPRI MRP-126, Material Reliability Program: Generic Guidance for Alloy 600 Management, November 2004, which specifies the objectives and requirements for an Alloy 600 management plan. The Alloy 600 Material Management Program is a living document and will be revised periodically to reflect the latest plant configurations and regulatory requirements.

1.5 The main objectives of the Alloy 600 Material Management Program include:

- Maintain plant safety
- Minimize the impact of Primary Water Stress Corrosion Cracking (PWSCC) on plant availability
- Develop and execute long-term strategies for Alloy 600/690 and related weld metal material management

1.6 The Alloy 600 Material Management Program shall encompass planning strategic efforts to prevent crack initiation and crack growth, finding cracks before leakage occurs, and preparing for repair and replacement activities that may become necessary because of extensive degradation.

1.7 CNP participates in industry initiatives, such as the Pressurized Water Reactor Owners Group (PWROG) and the Electric Power Research Institute (EPRI) Materials Reliability Program (MRP). Program inspection requirements regarding Alloy 82/182 pipe butt welds are consistent with ASME Boiler and Pressure Vessel Code, Case N-770-1 and Case N-722-1 subject to the conditions specified in 10 CFR 50.55a(g)(6)(ii)(F) and 10 CFR 50.55a(g)(6)(ii)(E), respectively [Ref. 5.2.1a].

1.8 The Alloy 600 Material Management Program will detect cracking from PWSCC using the examination and inspection requirements specified in ASME Section XI.

1.9 Alloy 600/690 components and related welds are also included in the following License Renewal programs:

- Control Rod Drive Mechanism and Other Vessel Head Penetration Inspection Program
- Steam Generator Integrity Program
- Reactor Vessels Internals Program

2 DEFINITIONS AND ABBREVIATIONS

Term	Meaning
Alloy 600	A nickel-based alloy used in Reactor Coolant System locations which is susceptible to cracking due to PWSCC. (designated as UNS N06600)
Alloy 82	A weld metal associated with Alloy 600. (designated as UNS N06082)
Alloy 182	A weld metal associated with Alloy 600. (designated as UNS W86182)
Alloy 690	A nickel-based alloy often used to replace Alloy 600 due to its resistance to PWSCC. (designated as UNS N06690)
Alloy 52	PWSCC resistant weld metal associated with Alloy 690. (designated as UNS N06052)
Alloy 152	PWSCC resistant weld metal associated with Alloy 690. (designated as UNS W86152)
ASME	American Society of Mechanical Engineers
BMI	Bottom Mounted Instrument
BMV	Bare Metal Visual
Commitment No. 8244	Alloy 600 Material Management Program that is required for license renewal.
CRDM	Control Rod Drive Mechanism
DM weld	Dissimilar Metal weld
EPRI	Electric Power Research Institute
GTAW	Gas Tungsten Arc Welding
ISI	Inservice Inspection
MRP	Materials Reliability Program conducted by EPRI to address the myriad of material aging issues faced by the nuclear industry.
MSIP	Mechanical Stress Improvement Process, an approved technique to mitigate certain PWSCC susceptible locations
NRC	United States Nuclear Regulatory Commission
PWR	Pressurized Water Reactor
PWROG	Pressurized Water Reactor Owners Group

Term	Meaning
PWSCC	Primary Water Stress Corrosion Cracking, phenomenon where susceptible material with residual stress exposed to primary water under high temperature can propagate a crack-like flaw.
RCPB	Reactor Coolant Pressure Boundary, all pressure-retaining piping and components within the boundary of the RCS.
RCS	Reactor Coolant System, the system containing borated water for the purpose of cooling the reactor core and controlling nuclear reactor criticality.
RFO	Refueling Outage
RPV	Reactor Pressure Vessel, the pressure vessel containing the nuclear fuel.
RVCH	Reactor Vessel Closure Head
RVHV	Reactor Vessel Head Vent
RVLIS	Reactor Vessel Level Indication System
SG	Steam Generator
UT	Ultrasonic examination
VHP	Vessel Head Penetration

3 RESPONSIBILITIES

3.1 CNP Plant Manager:

3.1.1 Maintains overall site responsibility for the effectiveness and implementation of the Alloy 600 Material Management Program.

3.2 Manager, Engineering Programs:

3.2.1 Verifies all elements of the Alloy 600 Material Management Program are being implemented in accordance with this instruction and associated procedures.

3.2.2 Approves Alloy 600 Material Management Program improvements and clarifications.

3.3 Supervisor, Engineering Programs

3.3.1 Maintains overall responsibility for the elements and effectiveness of the Alloy 600 Material Management Program.

3.3.2 Verifies the Alloy 600 Material Management Program has been established and is implemented.

3.3.3 Verifies a program owner has been assigned the responsibility to implement and maintain the elements of the Alloy 600 Material Management Program.

3.3.4 Verifies program ownership, qualification and training are implemented.

3.3.5 Verifies Alloy 600 Material Management Program Owner performs effective and timely evaluations and that corrective actions are implemented to address and resolve PWSCC issues.

3.3.6 Facilitates interface relationships among various programs associated with the Alloy 600 Material Management Program.

3.4 Manager, System Engineering:

3.4.1 Verifies the overall impact of PWSCC on systems, structures and components is addressed and included in Quarterly System Health Reports.

3.4.2 Verifies System Managers (Engineers) monitor and track applicable system performance parameters via system performance monitoring plans.

3.5 Alloy 600 Material Management Program Owner

3.5.1 Maintains overall responsibility for implementing and maintaining the elements of the Alloy 600 Material Management Program.

3.5.2 Develops and implements the program to satisfy the requirements of relevant NRC bulletins and other applicable industry guidance.

3.5.3 Establishes policies and methodologies for the control of PWSCC concerns in Alloy 600/690 and related weld metals not specifically expressed through regulatory requirements.

3.5.4 Verifies that appropriate sample locations, inspection techniques, and acceptance standards are determined in accordance with industry guidelines, NRC regulations (10 CFR 50.55a), and relevant NRC generic communications.

3.5.5 Verifies that effective and timely evaluations and corrective actions are implemented to address and resolve any failure found in Alloy 600/690 or related weld metals.

3.5.6 Issues periodic health reports assessing Alloy 600 Material Management Program performance.

3.5.7 Reviews and assesses industry operating experience and the results of industry meetings and workshops for potential improvements to the Alloy 600 Material Management Program.

3.6 Inservice Inspection Program Owner

3.6.1 Implements the augmented examinations of Alloy 600 Material Management Program inspection scope items.

3.6.2 Verifies augmented examinations are scheduled via the inservice inspection schedule and that examination results are documented and appropriately evaluated.

3.6.3 Attends ASME Code meetings, providing input/feedback for relevant code activities.

3.7 Design Engineering

3.7.1 Attends Pressurized Water Reactor Owners Group (PWROG) and EPRI MRP meetings, providing input/feedback for relevant material aging activities.

3.7.2 Provides input to Alloy 600 Material Management Program Owner concerning susceptibility rankings, program inspection requirements, and other program issues based on industry participation.

3.8 Performance Verification

3.8.1 Provides qualified personnel to perform certified examinations as directed by Engineering personnel to support the inspection schedule.

3.9 Radiation Protection/Chemistry/Environmental

3.9.1 Provides Radiation Protection support and off-site shipment of radioactive material.

3.9.2 Provides Chemistry analysis support.

3.9.3 Provides Environmental support and off-site shipment of hazardous waste.

3.10 Maintenance

3.10.1 Provides Maintenance support.

3.11 Work Control

3.11.1 Provides Work Control support.

4 DETAILS

4.1 General Program Information and History

4.1.1 Attachment 5, General Program Information, provides background information related to industry experience with PWSCC, CNP operating experience, and NRC generic communications.

4.2 Locations

4.2.1 Information on generic locations is available from several sources.

a. The scope of components that are known to contain Alloy 600/82/182 can be found in EPRI MRP-126, Material Reliability Program: Generic Guidance for Alloy 600 Management, November 2004.

4.2.2 CNP specific locations are documented within WCAP-16198-P, Rev. 1, PWSCC Susceptibility Assessment of the Alloy 600 and Alloy 82/182 Components in D.C. Cook Units 1 and 2, July 2004.

4.2.3 A comprehensive list of the Alloy 600/82/182 and Alloy 690/52/152 locations in the Reactor Coolant System for CNP Units 1 and 2 is provided in Attachment 1, Locations Containing Alloy 600/690 and Related Weld Metals.

4.3 PWSCC Susceptibility Rankings

4.3.1 Attachment 3, Susceptibility Rankings, contains the susceptibility rankings from WCAP-16198-P, Rev. 1, PWSCC Susceptibility Assessment of the Alloy 600 and Alloy 82/182 Components in D.C. Cook Units 1 and 2, July 2004

- Items in bold have been replaced since the assessment was completed.
- Items in italics have been mitigated since the assessment was completed.
- The susceptibility rankings for replaced or mitigated components are no longer applicable.

4.3.2 A susceptibility index was calculated for each location utilizing data related to microstructure, applied and residual stresses, and operating temperature.

4.3.3 Alloy 690 components and associated welds are not considered susceptible to PWSCC in the report.

4.4 Mitigation

4.4.1 PWSCC requires the confluence of a susceptible material, a chemical environment conducive to cracking, and sufficiently high tensile stresses on the material in contact with the coolant.

4.4.2 Mitigation is intended to extend the life of components by altering one or more of the conditions necessary for PWSCC to occur.

4.4.3 Mitigation techniques that are currently available or under evaluation include the following:

- Zinc Addition
- Mechanical Stress Improvement Process (MSIP)
- Waterjet Peening
- Laser Peening
- Outer Diameter Weld Overlay
- Clad with Alloy 690/52/152 material

4.4.4 Attachment 4, Mitigation and Repairs, lists mitigation, repair, and replacement activities that have been undertaken at CNP to address Alloy 600 material issues.

4.4.5 Due to the low probability of failure throughout the industry, CNP does not currently have any plans for mitigation of the BMI penetrations.

4.5 Repair Methods

4.5.1 Selection of the optimum repair method is normally based upon available technology, ASME Code requirements, radiological conditions, and economic factors.

4.5.2 At CNP, the BMI penetrations are the only pressure retaining locations containing Alloy 600 material that have not been mitigated. The most common repair methods for BMI penetrations are listed below.

a. Full Nozzle Repair

- The failed nozzle is replaced in its original configuration.

b. Half Nozzle Repair

- The outer portion of the nozzle is machined out from below, leaving the defect in the inner portion of the nozzle and/or j-groove weld in place.
- A half nozzle is inserted below and welded to the RPV lower head base material.
- This method was successfully implemented at South Texas Project Unit 1 after two cracks were identified in April 2003.

c. Mini-Inside Diameter Temper Bead Repair

- The mid-wall or ID temper bead repair involves removing the nozzle and machining the nozzle remnant away to a depth of approximately half the component wall thickness.
- The bore is liquid penetrant inspected.
- The replacement nozzle is then installed into the bore and welded into place for the inside diameter of the bore using Alloy 52 weld metal.
- A machine GTAW process employing the ambient temperature temper bead welding technique is used.
- The inside diameter of the weld deposit is machined and/or ground to establish the nozzle bore.
- The weld deposit is examined by liquid penetrant and ultrasonic examination.
- This method can be used in nozzle bores as small as one inch in diameter, making it an effective approach for BMI nozzle repairs.

4.6 Inspections

4.6.1 Attachment 2, Inspections, provides a list of inspections that CNP performs for components in the Alloy 600 Material Management Program.

4.6.2 Sources of inspection requirements may include:

- NRC Regulations (10 CFR 50.55a)
- NRC Orders or other relevant NRC communications
- ASME Boiler and Pressure Vessel Code, Section XI
- CNP's Inservice Inspection Program
- Plant Procedures and Programs
- License Renewal Programs
- Joint Industry Issues Programs (EPRI Material Reliability Program)

4.7 Acceptance Criteria

4.7.1 Acceptance criteria are determined in accordance with NRC regulations (10 CFR 50.55a) and applicable industry guidelines.

4.7.2 Attachment 2, Inspections, lists the acceptance criteria for each inspection.

4.8 Corrective Measures

4.8.1 Inspection results that do not meet the acceptance criteria are documented via PMP-7030-CAP-001, Action Initiation.

4.8.2 **WHEN** the acceptance criteria are not met, **THEN** an evaluation is performed in accordance with 10 CFR 50.55a, ASME Boiler and Pressure Vessel Code (Section XI or applicable Code Case), and industry guidelines.

4.8.3 The evaluation will determine if the component is acceptable for continued service and the corrective actions required (further evaluation, additional examinations, increased inspection frequency, repair/replacement, etc.).

4.8.4 Components found to be unacceptable for continued service are corrected through implementation of appropriate repair or replacement as dictated by 10 CFR 50.55a and industry guidelines.

4.9 Records

4.9.1 Inspection results are transmitted to Nuclear Document Management in accordance with PMP-2030-REC-001, Records Management. [Ref. 5.1.5]

5 REFERENCES

5.1 Use References:

5.1.1 ASME Boiler and Pressure Vessel Code, Section XI – Rules for Inservice Inspection of Nuclear Power Plant Components

5.1.2 ASME Boiler and Pressure Vessel Code, Case N-722-1

5.1.3 ASME Boiler and Pressure Vessel Code, Case N-729-1

5.1.4 ASME Boiler and Pressure Vessel Code, Case N-770-1

5.1.5 PMP-2030-REC-001, Records Management

5.1.6 PMI-2291, Work Control Process

5.1.7 PMP-5070-ISI-002, Inservice Inspection Program Implementation

5.1.8 PMP-7030-CAP-001, Action Initiation

5.1.9 PMP-7030-CAP-002, Condition Evaluation, Action, and Closure

5.2 Writing References:

5.2.1 Source References

a. Commitment No. 8244, Alloy 600 Aging Management Program

b. AEP:NRC:3034, Attachment 2, Appendix A, Section A.2.1.1, dated October 31, 2003 (ML033070177 and ML033070182)

c. AEP:NRC:3034, Attachment 2, Appendix B, Section B.1.1, dated October 31, 2003 (ML033070177 and ML033070182)

d. Safety Evaluation Report (SER) Related to the License Renewal of the Donald C. Cook Nuclear Plant, Units 1 and 2, May 2005 (ML-51510092)

e. EPRI MRP-126, Material Reliability Program: Generic Guidance for Alloy 600 Management, November 2004

f. EPRI MRP-169, Material Reliability Program: Technical Basis for Preemptive Weld Overlays for Alloy 82/182 Butt Welds in PWRs, October 2005

g. WCAP-16198-P, Rev. 1, PWSCC Susceptibility Assessment of the Alloy 600 and Alloy 82/182 Components in D.C. Cook Units 1 and 2, July 2004

h. AEP:NRC:4034-10, Attachment 1, Section B.1.1.2-1, dated August 11, 2004 (ML042470410)

i. Donald C. Cook Nuclear Plant, Unit 1 Technical Specifications

j. Donald C. Cook Nuclear Plant, Unit 2 Technical Specifications

5.2.2 General References

a. LRP-EAMP-01, Rev. 3, Evaluation of Aging Management Programs, Section 3.1

b. LRP-MAMR-01, Aging Management Review of the Reactor Coolant System

Component	Unit	Location	Material	Description	Reference
Reactor Vessel	1	Hot and Cold Leg Nozzles	Alloy 82/182	Unit 1 reactor vessel hot and cold leg nozzles are welded to the safe-ends utilizing Alloy 82/182 buttering and welds. All welds have been mitigated using MSIP.	WCAP-16198-P EC-0000048752
Reactor Vessel	1, 2	BMI Nozzles	Alloy 600 Alloy 82/182	The Alloy 600 nozzles are connected to the reactor vessel using Alloy 82/182 partial penetration welds. The welds connecting the BMI nozzles to the stainless steel guide tubes are also Alloy 82/182.	WCAP-16198-P
Reactor Vessel	1, 2	Leak-off Monitor Tubes and welds	Alloy 600 Alloy 82/182	The reactor vessel leak-off monitor tubes are Alloy 600 with Alloy 82/182 welds.	WCAP-16198-P
Reactor Vessel	1, 2	CRDM Nozzles and welds	Alloy 690 Alloy 52/152	The CRDM penetration nozzles are fabricated from Alloy 690. They are attached to the RVCH with Alloy 52/152 welds.	1-MOD-55520 2-MOD-55516
Reactor Vessel	1, 2	RVLIS and RVHV Nozzles and associated welds	Alloy 690 Alloy 52/152	The RVLIS and RVHV nozzles are fabricated from Alloy 690. They are attached to the RVCH with Alloy 52/152 welds.	1-MOD-55520 2-MOD-55516
Reactor Vessel	1, 2	Core Support Pads	Alloy 600 Alloy 82/182	The core support pads are fabricated from Alloy 600 and they are attached to the reactor vessel using Alloy 82/182 welds.	WCAP-16198-P
Reactor Vessel	1, 2	Reactor Vessel Internals	Alloy 600	The clevis inserts are fabricated from Alloy 600 material.	WCAP-16198-P
Pressurizer	1, 2	Spray, Surge, Safety and Relief Nozzles	Alloy 82/182 Alloy 52	Nozzle safe-end welds contain Alloy 82/182. All welds were mitigated with full structural weld overlay using Alloy 52.	WCAP-16198-P WCAP-16428-P EC-MOD-ECC-0000046930 EC-0000050750

Component	Unit	Location	Material	Description	Reference
Steam Generator	1	Divider Plate	Alloy 690 Alloy 52/152	The primary head divider plate with corner pieces is composed of Alloy 690. It is attached to the primary head and tubesheet with Alloy 52/152 welds.	7803A035 7803E037
Steam Generator	2	Divider Plate	Alloy 600 Alloy 82/182	The primary head divider plate and stub runner are composed of Alloy 600. The stub runner to tubesheet weld contains Alloy 82/182. The divider plate is attached to the stub runner and the lower bowl using Alloy 82/182 welds.	WNEP-8737 WCAP-16198-P
Steam Generator	1, 2	Tubesheet Cladding	Alloy 82/182	The tubesheets in Unit 1 are clad with Alloy 82. The tubesheets and primary head radius in Unit 2 are clad with Alloy 82/182.	7803A035 7803E037 WCAP-16198-P
Steam Generator	1	Primary Manway Diaphragm	Alloy 690	The primary manway diaphragm is composed of Alloy 690.	7803A035 7803E037
Steam Generator	1	Primary Nozzle	Alloy 690 Alloy 52	The primary nozzle dam ring is composed of Alloy 690. The primary nozzle to safe-end weld contains Alloy 52.	7803A035 7803E037
Steam Generator	2	Primary Nozzle	Alloy 600	The primary nozzle dam ring is composed of Alloy 600.	WNEP-8737
Steam Generator	1, 2	Tubes	Alloy 690	The steam generator tubes in both units are Alloy 690.	WNEP-8737 7803A035 7803E037
Steam Generator	2	Shell	Alloy 82	The cladding over the carbon steel shell (covering the girth seam weld) is Alloy 82.	WCAP-16198-P

Location	Unit	Method	Inspection Requirement	Frequency	Acceptance Criteria	Comments
RPV Hot and Cold Leg Nozzles	1	Volumetric	ASME Code Case N-770-1, Item D[1]	Per inspection requirement	Per inspection requirement	—
BMI Penetrations	1, 2	BMV	ASME Code Case N-722-1, Item B15.80[2]	Each refueling outage (only required every other refueling outage)	Per inspection requirement	CNP performs this inspection every refueling outage due to recurring refueling cavity leakage issues
Leak-off Monitor Tubes	1, 2	VT-2	ASME Section XI, Category B-P, Item B15.10	Each refueling outage (per inspection requirement)	Per inspection requirement	Inspected each refueling outage during NOP/NOT system leakage test
RVCH Nozzles and Welds	1, 2	Volumetric, Surface	ASME Code Case N-729-1, Item B4.40[3]	Per inspection requirement	Per inspection requirement	—
RVCH	1, 2	BMV	ASME Code Case N-729-1, Item B4.30[3]	Every third refueling outage (per inspection requirement)	Per inspection requirement	—
Core Support Pad Welds	1, 2	VT-3	ASME Section XI, Category B-N-2, Item B13.60	Once per interval (per inspection requirement)	Per inspection requirement	—
Clevis Inserts	1, 2	VT-3	ASME Section XI, Category B-N-2, Item B13.60	Once per interval (per inspection requirement)	Per inspection requirement	—
Pressurizer Nozzle (1-PRZ-23) to Safe-End Weld	1	Volumetric	ASME Code Case N-770-1, Item F[1]	Per inspection requirement	Per inspection requirement	A crack was discovered in this weld and later reinforced by full structural weld overlay

Location	Unit	Method	Inspection Requirement	Frequency	Acceptance Criteria	Comments
Pressurizer Nozzle to Safe-End Welds (except 1-PRZ-23)	1, 2	Volumetric	ASME Code Case N-770-1, Item C[1]	Per inspection requirement	Per inspection requirement	—
SG Divider Plate	1, 2	Visual	No requirement	Concurrent with SG Tube inspections[4]	Any signs of degradation or cracking	Remote, as-found visual scan (not an ASME Code examination)
SG Tubesheet Cladding	1, 2	Visual	No requirement	Concurrent with SG Tube inspections[4]	Any signs of degradation or cracking	Remote, as-found visual scan (not an ASME Code examination)
SG Primary Manway Diaphragm	1	Visual	No requirement	Concurrent with SG Tube inspections[4]	Any signs of degradation or cracking	Visual scan (not an ASME Code examination)
SG Primary Nozzle to Safe-End Weld	1	Volumetric	ASME Section XI, Category R-A, Item R1.20	Per inspection requirement	Per inspection requirement	CNP does not currently inspect this component as determined through the risk informed selection process in ASME Section XI.
SG Nozzle Dam Rings	1, 2	Visual	No requirement	Concurrent with SG Tube inspections[4]	Any signs of degradation or cracking	Remote, as-found visual scan (not an ASME Code examination)
SG Tubes	1, 2	Volumetric	Technical Specifications Section 5.5.7	Per inspection requirement	Per inspection requirement	—

Location	Unit	Method	Inspection Requirement	Frequency	Acceptance Criteria	Comments
SG Shell Cladding	2	Visual	No requirement	Concurrent with SG Tube inspections[4]	Any signs of degradation or cracking	Remote, as-found visual scan (not an ASME Code examination)

[1] Implementation of ASME Code Case N-770-1 is subject to the conditions specified in 10 CFR 50.55a (g)(6)(ii)(F)
[2] Implementation of ASME Code Case N-722-1 is subject to the conditions specified in 10 CFR 50.55a (g)(6)(ii)(E)
[3] Implementation of ASME Code Case N-729-1 is subject to the conditions specified in 10 CFR 50.55a (g)(6)(ii)(D)
[4] These components are inspected when tube inspections of the corresponding steam generator are performed in accordance with Donald C. Cook Technical Specifications.

WCAP-16198-P Susceptibility Rankings
Unit 1

*Items in **Bold** have been replaced by PWSCC resistant materials. The susceptibility data is no longer applicable.
*Items in *Italics* have been mitigated. The susceptibility data is no longer applicable.

Component	Alloy	Effective Stress (MPa)	Service Temp. (°F)	Susceptibility Index
Pressurizer Spray Nozzle to Safe-End Weld	*82/182*	*469.5*	*640.0*	*9.48E-09*
Pressurizer Safety & Relief Nozzle to Safe-End Weld	*82/182*	*438.5*	*640.0*	*7.21E-09*
Pressurizer Surge Nozzle to Safe-End Weld	*82/182*	*411.6*	*639.0*	*5.38E-09*
CRDM Nozzle to Head Welds	**82/182**	**650.8**	**581.0**	**3.36E-09**
Head Vent Nozzle	**600**	**485.3**	**581.0**	**2.78E-09**
CRDM Nozzles	**600**	**399.9**	**581.0**	**1.60E-09**
RPV Hot Leg Nozzle to Safe-End Welds	*82/182*	*390.2*	*589.8*	*6.26E-10*
BMI Nozzles	600	455.1	528.2	2.64E-10
Head Vent to Head Weld	**82/182**	**545.2**	**581.0**	**1.66E-10**
SG Tubesheet Cladding (Hot)	82	299.2	589.8	1.44E-10
BMI Nozzle to Vessel Welds	82/182	410.9	528.2	5.26E-11
BMI Nozzle to Guide Tube Welds	82/182	393.0	528.2	4.39E-11
RPV Cold Leg Nozzle to Safe-End Welds	*82/182*	*390.2*	*528.2*	*4.27E-11*
Core Support Pad (at weld)	600	306.8	528.2	2.72E-11
Core Support Pad Weld	82/182	348.1	528.2	2.71E-11
SG Tubesheet Cladding (Cold)	82	299.2	528.2	9.85E-12
Core Support Pad	600	29.2	528.2	2.24E-15
Head Vent to Housing Weld	**82/182**	**466.1**	**250.0**	**1.33E-18**
CRDM Nozzle to Housing Welds	**82/182**	**418.5**	**250.0**	**8.66E-19**
Head Vent Elbow	**600**	**70.0**	**250.0**	**2.26E-21**

WCAP-16198-P Susceptibility Rankings
Unit 2

*Items in **Bold** have been replaced by PWSCC resistant materials. The susceptibility data is no longer applicable.
*Items in *Italics* have been mitigated. The susceptibility data is no longer applicable.

Component	Alloy	Effective Stress (MPa)	Service Temp. (˚F)	Susceptibility Index
Pressurizer Spray Nozzle to Safe-End Weld	*82/182*	*469.5*	*650.7*	*1.40E-08*
Pressurizer Safety & Relief Nozzle to Safe-End Weld	*82/182*	*438.5*	*650.7*	*1.06E-08*
Pressurizer Surge Nozzle to Safe-End Weld	*82/182*	*411.6*	*654.8*	*9.64E-09*
CRDM Nozzle to Head Welds	**82/182**	**650.8**	**601.0**	**7.62E-09**
Head Vent Nozzle	**600**	**485.3**	**601.0**	**6.26E-09**
SG Divider Plate & Stub Runner (Hot)	600	434.4	607.2	3.91E-09
Head Vent to Head Weld	**82/182**	**545.2**	**601.0**	**3.76E-09**
CRDM Nozzles	**600**	**399.9**	**601.0**	**3.64E-09**
SG Stub Runner to Tube Sheet & Stub Runner to Divider Plate Welds (Hot)	82/182	503.3	607.2	3.52E-09
SG Divider Plate to Lower Bowl Weld (Hot)	82/182	415.7	607.2	1.64E-09
SG Shell Cladding (Hot)	82	415.7	607.2	1.09E-09
BMI Nozzles	600	455.1	544.0	5.38E-10
SG Tubesheet & Radius Cladding (Hot)	82/182	299.2	607.2	4.40E-10
SG Divider Plate & Stub Runner (Cold)	600	488.8	544.0	4.30E-10
SG Stub Runner to Tube Sheet & Stub Runner to Divider Plate Welds (Cold)	82/182	557.7	544.0	3.64E-10
SG Divider Plate to Lower Bowl Weld (Cold)	82/182	415.7	544.0	1.12E-10
BMI Nozzle to Vessel Welds	82/182	410.9	544.0	1.07E-10
BMI Nozzle to Guide Tube Welds	82/182	393.0	544.0	8.96E-11

WCAP-16198-P Susceptibility Rankings
Unit 2

*Items in **Bold** have been replaced by PWSCC resistant materials. The susceptibility data is no longer applicable.
*Items in *Italics* have been mitigated. The susceptibility data is no longer applicable.

Component	Alloy	Effective Stress (MPa)	Service Temp. (˚F)	Susceptibility Index
SG Shell Cladding (Cold)	82	420.5	544.0	7.87E-11
Core Support Pad (at weld)	600	306.8	544.0	5.56E-11
Core Support Pad Weld	82/182	348.1	544.0	5.53E-11
SG Tubesheet & Radius Cladding (Cold)	82/182	299.2	544.0	3.02E-11
Core Support Pad	600	29.2	544.0	4.56E-15
CRDM Nozzle to Housing Welds	**82/182**	**418.5**	**250.0**	**8.66E-19**

Location	Unit	Date	Method	Reference
Reactor Vessel Hot & Cold Leg Nozzles	1	U1C23 RFO (Spring 2010)	MSIP completed on all 8 RPV nozzle to safe-end DM welds	EC-0000048752
Reactor Vessel Closure Head	1	U1C21 RFO (Fall 2006)	Replacement with new head	1-MOD-55520
Reactor Vessel Closure Head	2	U2C17 RFO (Fall 2007)	Replacement with new head	2-MOD-55516
Pressurizer Nozzle (1-PRZ-23)	1	U1C20 RFO (Spring 2005)	Full structural weld overlay to repair a crack on 1-PRZ-23 safety nozzle to safe-end DM weld	WCAP-16428-P JO 05099030
Pressurizer Nozzles (all remaining nozzles)	1	U1C21 RFO (Fall 2006)	Full structural weld overlay on all remaining Pressurizer Spray, Safety, Relief, and Surge Nozzle to safe-end DM welds	EC-MOD-ECC-0000046930
Pressurizer Nozzles	2	U2C16 RFO (Spring 2006)	Full structural weld overlay on all Pressurizer Spray, Safety, Relief, and Surge Nozzle to safe-end DM welds	EC-0000050750

CONSTRUCTION

Alloy 600 materials were incorporated into the RCS of Westinghouse PWR designs for three primary reasons:

- Resistance to chloride stress corrosion cracking
- Corrosion resistance in high temperature water
- Compatible coefficient of thermal expansion to nuclear pressure vessel steels

MECHANISM

PWSCC is a form of stress corrosion cracking that affects Alloy 600/82/182 materials exposed to a primary water environment within chemistry specification limits. The primary susceptibility factors for PWSCC include:

- Thermo-mechanical processing
- Stress level
- Chemical environment
- Temperature

HISTORY

Stress corrosion cracking of nickel base materials in high purity water at elevated temperatures was first demonstrated in the laboratory in the late 1950s. In operating PWRs, PWSCC was initially observed on the primary side of Alloy 600 steam generator tubing. The first case of PWSCC involving a leaking Alloy 600 pressurizer instrument nozzle was discovered at San Onofre Unit 3 in 1986. The first instance in a RPV upper head Alloy 600 penetration was identified in France at Bugey Unit 3 in 1991. Finally, the first confirmed case of PWSCC in an Alloy 82/182 weld metal was discovered in 2000 at V.C. Summer, in a butt weld joining a reactor vessel hot leg nozzle to the RCS piping.

Since the above mentioned events, there have been numerous failures at foreign and domestic PWRs, involving Alloy 600 pressurizer heater sleeves, instrument nozzles, thermocouple nozzles, CRDM nozzles and safe ends, and buttering welds of piping exposed to the RCS. A Summary of key industry events involving PWSCC of Alloy 600/82/182 is included in EPRI MRP-126, Material Reliability Program: Generic Guidance for Alloy 600 Management, November 2004, Appendix A, Summary of Key Industry Events Involving PWSCC of Alloy 600/82/182.

CNP OPERATING EXPERIENCE

CRDM Penetration Nozzle:

In 1994, CNP examined the majority of accessible CRDMs on Unit 2 using eddy current techniques. Penetration #75 was found to have three axial cracks located in the Alloy 600 CRDM penetration base material. One of the cracks extended about 43% through-wall.

Using ASME Section XI flaw evaluation standards the NRC approved the operation of the Unit for one fuel cycle. The cracks were removed to an acceptable level and subsequently embedded in accordance with approved ASME Section XI repair techniques. The Unit 2 Reactor Vessel Closure Head was replaced during the U2C17 refueling outage (Fall 2007).

Pressurizer Safety Nozzle:

During the U1C20 refueling outage in April 2005, an axially oriented indication was detected in the Pressurizer Safety Nozzle (1-PRZ-23) to safe-end dissimilar metal weld at CNP Unit 1. The indication was found to initiate at the inside surface of the nozzle, extending approximately 1.23" into the Alloy 82/182 weld and spanning 0.4" along the axis of the nozzle. The indication was confined in the Alloy 82/182 weld material. There was no evidence that the indication was present in the adjacent stainless steel or carbon steel. A full structural weld overlay repair was performed to maintain weld integrity.

NRC GENERIC COMMUNICATIONS

NRC Information Notice 90-10 (February 23, 1990), "Primary Water Stress Corrosion Cracking (PWSCC) of Inconel 600," was issued to alert PWR licensees of the potential problems associated with PWSCC of Alloy 600 that had occurred at several domestic and foreign PWR plants. During the 1989 RFO at Calvert Cliffs Unit 2, visual examination detected leakage in 20 pressurizer heater sleeves and 1 upper-level pressurizer instrument nozzle. Subsequent NDE confirmed the presence of axially oriented, crack-like indications in these components and 4 additional heater sleeves. The causative failure mechanism was postulated to be PWSCC.

On February 27, 1986 leakage was detected in an upper-level pressurizer instrument nozzle at San Onofre Nuclear Generating Station Unit 3. Subsequent NDE and metallurgical examination revealed the leak path to be axially oriented PWSCC.

In spring 1989, leakage from pressurizer instrument nozzles was observed in two foreign PWRs. NDE revealed crack like indications that were both axially and circumferentially oriented. NDE of five additional PWRs revealed 12 more nozzles with crack-like indications.

NRC Generic Letter 97-01 (April 1, 1997), "Degradation of Control Rod Drive Mechanism Nozzle and Other Vessel Closure Head Penetrations," requested PWR licensees to describe their program for ensuring the timely inspection of the control rod drive mechanisms (CRDMs) and other reactor vessel head penetrations (RVHPs). In addition, licensees were asked to assess and provide a description of any resin bead intrusion, as described in NRC Information Notice (IN) 96-11, which would have resulted in sulfate levels exceeding the EPRI primary water chemistry guidelines.

CNP Responses:

- Letter No. AEP:NRC:1218B, dated April 29, 1997
- Letter No. AEP:NRC:1218C, dated August 1, 1997
- Letter No. AEP:NRC:1218D, dated November 4, 1997
- Letter No. AEP:NRC:1218F, dated June 11, 1999

NRC Information Notice 2000-17 (October 18, 2000), "Crack in Weld Area of Reactor Coolant System Hot Leg Piping at V.C. Summer," described the licensees discovery of leakage from the air boot around the A loop RCS hot leg pipe on 10/7/2000. Subsequent NDE revealed that the leak path was an inner diameter (ID) initiated axial indication in the Alloy 82/182 weld metals. A metallurgical failure analysis determined that the causative failure mechanism was PWSCC. High residual tensile stresses resulting from extensive weld repairs during original construction were determined to have been a significant contributor. The "A" loop hot leg weld was removed and replaced in its entirety. The licensee also identified other ECT indications in four of the other five reactor coolant system nozzle to pipe welds. Westinghouse performed an evaluation to justify continued operation of the "B" and "C" hot legs without repair of these ECT indications.

As a result of their evaluation of this event, the NRC identified several generic issues: 1) potential weaknesses in the ability of the ASME Code-required non-destructive examination techniques to detect and size small inner-diameter stress corrosion cracks; 2) potential weaknesses in the ASME Code that allows multiple weld repairs which affect residual weld stress and PWSCC; and 3) potential weaknesses in RCS leak detection systems; and 4) questions regarding the continued applicability of "leak before break" analyses.

NRC Information Notice 2001-05 (April 30, 2001), "Through-Wall Circumferential Cracking of Reactor Pressure Vessel Head Control Rod Drive Mechanism Penetration Nozzles at Oconee Nuclear Station, Unit 3," was issued to alert addressees to the recent detection of through wall circumferential cracks in two of the control rod drive mechanism (CRDM) penetration nozzles and weldments at the Oconee nuclear Station, Unit 3 (ONS3). On February 18, 2001, nine leaking CRDM nozzles at ONS3 were detected by visual examinations during a planned maintenance outage. All of the flaws were initially characterized as either axial or below-the-weld circumferential indications by NDE. However, subsequent NDE and metallurgical examinations revealed the presence of OD initiated PWSCC, located above the welds and with circumferential orientation in two of the nozzles. The discovery of such flaws challenged previous safety assessments conducted by the PWR owners groups and the NRC that had assumed PWSCC of RPVH penetrations would be predominantly axial in orientation.

NRC Bulletin 2001-01 (August 3, 2001), "Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles," was issued following the discovery of circumferential cracks in two CRDM nozzles at Oconee Nuclear Station Unit 3 (ONS3). The bulletin requested PWR licensees to provide information related to the structural integrity of the RPVH penetration nozzles. The requested data included the results of previous inspections, the inspections and repairs undertaken to satisfy applicable regulatory requirements, and the basis for concluding that future inspections would ensure compliance with applicable regulatory requirements. This information was provided to the NRC in the letters listed below. The NRC responded in a letter dated January 14, 2002 that CNP provided the requested information.

In response to NRC Bulletin 2001-01, reactor vessel head penetration (VHP) examinations were performed during the Unit 1 and Unit 2 refueling outages in 2002. No nozzle leakage and no cracks were identified on Unit 1. No nozzle leakage was identified on Unit 2. However, three small axial cracks were identified on the inside diameter of Penetration #74 on Unit 2. Reactor vessel head inspection results were provided to the NRC in letters AEP:NRC:2054 and AEP:NRC:2054-04.

CNP Responses:

- Letter No. C0801-20, dated September 4, 2001
- Letter No. C1001-08, dated October 12, 2001
- Letter No. C1001-05, dated November 5, 2001
- Letter No. C1101-16, dated November 30, 2001
- Letter No. C1201-05, dated December 6, 2001
- Letter No. AEP:NRC:2054, dated March 28, 2002 – Unit 2 Reactor Vessel Head Inspection Findings
- Letter No. AEP:NRC:2054-04, dated July 3, 2002 – Unit 1 Reactor Vessel Head Inspection Findings

NRC Information Notice 2002-11 (March 12, 2002), "Recent Experience with Degradation of Reactor Pressure Vessel Head," was issued following the discovery of severe degradation of the RPVH at Davis-Besse Nuclear Power Station. On February 27, 2002 while conducting RPVH inspections in response to Bulletin 2001-01, the licensee discovered axially oriented PWSCC in three CRDM nozzles in the RPVH. Part way through the repair process on one of the nozzles, a cavity in RPVH was discovered. Leaking boric acid had consumed the ferritic steel in a localized region on the downstream side of the nozzle, leaving only the 3/8" stainless steel cladding still intact.

NRC Bulletin 2002-01 (March 18, 2002), "Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity," was issued following the discovery by Davis-Besse of cracking in several CRDM nozzles and significant reactor head degradation associated with one of these leaking nozzles. The bulletin requested PWR licensees to provide: 1) information related to the integrity of the reactor coolant pressure boundary including the reactor pressure vessel head and the extent to which inspection and maintenance programs have been undertaken to satisfy applicable regulatory requirements, and 2) the basis for concluding that plants satisfy applicable regulatory requirements related to the structural integrity of the reactor coolant pressure boundary and future inspections will ensure continued compliance with applicable regulatory requirements. A Request for Additional Information (RAI) was later issued by the NRC in a letter dated November 18, 2002 to obtain more detailed information regarding licensees' boric acid corrosion control (BACC) programs.

CNP Responses:

- Letter No. AEP:NRC:2054-01, dated April 1, 2002
- Letter No. AEP:NRC:2054-02, dated May 10, 2002
- Letter No. AEP:NRC:2054-03, dated July 3, 2002
- Letter No. AEP:NRC:3054, dated January 17, 2003

NRC Information Notice 2002-13 (April 4, 2002), "Possible Indicators of Ongoing Reactor Pressure Vessel Head Degradation," was issued to report the findings of an augmented inspection team (AIT) sent by the NRC to investigate the circumstances of the degradation of the Davis-Besse RPVH material. This AIT identified several possible indicators of the observed reactor pressure boundary degradation. These included: 1) unidentified RCS leakage; 2) containment air cooler fouling; and 3) radiation element filter fouling. Licensees were advised to be aware of such indicators even though they do not provide clear evidence of ongoing degradation.

NRC Bulletin 2002-02 (August 9, 2002), "Reactor Pressure Vessel Head and Vessel Head Penetration Nozzle Inspection Programs," was issued in response to the discoveries of circumferential cracking of VHP nozzles at Oconee Nuclear Station 3 and other PWR facilities, the RPV head material degradation at Davis-Besse, and the NRC's review of licensees' responses to Bulletins 2001-01 and 2002-01. These issues raised concerns about the adequacy of current inspection programs that rely solely on visual examinations as the primary inspection method to ensure RPVH and VHP nozzle structural integrity and compliance with applicable regulations. PWR licensees were strongly encouraged to supplement their inspection programs with non-visual methods and to provide technical justification for the efficacy of these programs.

CNP Response:

- Letter No. AEP:NRC:2054-05, dated September 6, 2002

NRC Order EA-03-009 (February 11, 2003) modified PWR licenses by establishing required inspections of RPV heads and associated penetration nozzles. The NRC felt that these requirements were necessary to provide reasonable assurance that plant operations did not pose an undue risk to the public health and safety. The inspection requirements included: 1) bare metal visual (BMV) inspections of the RPVH surface, including 360° around each penetration nozzle, and 2) volumetric (UT) or surface (ECT or PT) inspections of the wetted surface of each J-Groove weld and RPVH penetration nozzle base material. The frequency of these examinations was determined by a reactor's susceptibility category, calculated as effective degradation years (EDY) based upon operating time and RVH temperature. The requirements of the Order were expected to remain in effect pending long-term changes to the NRC regulations, specifically 10 CFR 50.55a.

CNP Responses:

- Letter No. AEP:NRC:3054-03, dated March 3, 2003
- Letter No. AEP:NRC:3054-04, dated March 26, 2003
- Letter No. AEP:NRC:3054-06, dated May 13, 2003
- Letter No. AEP:NRC:3054-08, dated June 2, 2003
- Letter No. AEP:NRC:3054-11, dated August 13, 2003
- Letter No. AEP:NRC:4054, dated January 26, 2004

NRC Regulatory Issue Summary 2003-13 (July 29, 2003), "NRC Review of Responses to Bulletin 2002-01, 'Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity'," provided the conclusions of the NRC staff's review of PWR licensees' responses to Bulletin 2002-01. In it, they concluded that: 1) most licensees do not perform inspections of Inconel Alloy 600/82/182 materials beyond those required by Section XI of the ASME Code, 2) such inspections are generally performed without removing insulation and are not capable, in many cases, of detecting through-wall leakage, and 3) existing monitoring programs may need to be enhanced to ensure early detection and prevention of leakage from the RCPB. No responses to the RIS from PWR licensees were required.

NRC Information Notice 2003-11 (August 13, 2003), "Leakage Found on Bottom Mounted Instrumentation Nozzles," described indications of leakage in the form of boron deposits discovered on two bottom-mounted instrumentation (BMI) nozzles at South Texas Project Unit 1 (STP Unit 1). These deposits were discovered while performing BACC walkdowns during the Unit's 1RE11 RFO. Similar inspections performed during the prior RFO had not detected any evidence of leakage.

NRC Bulletin 2003-02 (August 21, 2003), "Leakage from Reactor Pressure Vessel Lower Head Penetrations and Reactor Coolant Pressure Boundary Integrity," was issued subsequent to the discovery of two leaking bottom mounted instrumentation (BMI) penetrations in the RPV lower head at South Texas Project Unit 1 on April 12, 2003. The NRC advised PWR licensees that current methods of inspecting the RPV lower head penetrations may need to be supplemented with additional measures (e.g., bare-metal visual inspections (BMV)) to detect RCPB leakage. Licensees were requested to provide a description and findings of the RPV lower head inspection program that has been performed in the past, and a description of the program that will be implemented during future refueling outages. Inspection results were provided in letters AEP:NRC:4054-04 and AEP:NRC:4054-11. The NRC replied in letters dated October 15, 2004 and July 28, 2005 that CNP met the reporting requirements of this Bulletin for Units 1 and 2 respectively.

In response to NRC Bulletin 2003-02, CNP performed a 360-degree bare metal visual examination on all 58 RPV lower head penetrations during the Unit 1 Fall 2003 refueling outage and the Unit 2 Fall 2004 refueling outage. No evidence of penetration leakage was observed.

CNP Responses:

- Letter No. AEP:NRC:3054-14, dated September 17, 2003
- Letter No. AEP:NRC:4054-04, dated March 25, 2004
- Letter No. AEP:NRC:4054-11, dated January 6, 2005
- Letter No. AEP:NRC:5054-07, dated June 3, 2005

NRC Information Notice 2003-11 Supplement 1 (January 8, 2004), "Leakage Found on Bottom Mounted Instrumentation Nozzles," provided the destructive examination results of the boat sample extracted from the STP Unit 1 BMI nozzle: 1) the nozzle exhibited OD initiated, axially oriented PWSCC in the vicinity of the J-groove weld; 2) there was evidence of LOF at the tube-to-weld interface; 3) the leak path in the weld metal was a crack-like defect that was thought to be an initial fabrication flaw. The 561°F operating temperature of the BMIs was the lowest recorded temperature for PWSCC of an Alloy 600 component in an operating PWR to date.

NRC First Revised Order EA-03-009 (February 20, 2004) was issued to address revisions to bare metal visual inspections, penetration nozzle inspection coverage, flexibility in combination of non-destructive examination methods, flaw evaluation and requirements for plants which had replaced their RPV heads. These were common issues that had emerged in numerous relaxation requests from licensees since original issuance of the Order.

CNP Responses:

- Letter No. AEP:NRC:4054-03, dated March 9, 2004
- Letter No. AEP:NRC:5054-03, dated January 20, 2005
- Letter No. AEP:NRC:5054-09, dated June 27, 2005

NRC Information Notice 2004-11 (May 6, 2004), "Cracking in Pressurizer Safety and Relief Nozzles and in Surge Line Nozzle," described the discovery of PWSCC in several bimetallic nozzle-to-safe end welds. In September 2003, axially oriented cracks were discovered in the Alloy 132 weld metal joining the 316 SS safe ends to the low alloy steel pressurizer safety and relief nozzles at Tsuruga Unit 2. In October 2003, a similar indication was discovered by UT in Alloy 82/182 weld metal joining the carbon steel surge line nozzle to cast 316 SS safe end at Three Mile Island, Unit 1 (TMI-I). Investigations conducted by both utilities revealed evidence of previous weld repairs during construction on the safety nozzle at Tsuruga and the surge line nozzle at TMI-1. TMI-1 performed a full structural weld overlay repair to maintain weld integrity.

NRC Bulletin 2004-01 (May 28, 2004), "Inspection of Alloy 82/182/600 Materials Used in the Fabrication of Pressurizer Penetrations and Steam Space Piping Connections at Pressurized Water Reactors," was issued to advise PWR licensees that existing inspection methods may need to be supplemented to detect and characterize PWSCC flaws. Licensees were requested to provide descriptions of the pressurizer penetrations and steam space piping, as well as past and future inspections that will be performed to ensure that degradation of Alloy 600/82/182 materials used in the fabrication of the pressurizer penetrations and steam space piping connection will be identified, adequately characterized and repaired. Inspection results were provided in letters AEP:NRC:5054 and AEP:NRC:5054-08. The NRC replied in a letter dated April 17, 2007 that CNP's responses to NRC Bulletin 2004-01 were acceptable.

CNP Responses:

- Letter No. AEP:NRC:4054-07, dated July 26, 2004
- Letter No. AEP:NRC:4054-10, dated October 28, 2004
- Letter No. AEP:NRC:5054, dated January 6, 2005
- Letter No. AEP:NRC:5054-08, dated June 24, 2005

NRC Information Notice 2005-02 (February 4, 2005), "Catawba SG Bowl Drain Cracking," described the discovery of boric acid deposits in the vicinity of a SG bowl drain line while conducting bare metal visual examinations of the plants Alloy 600/82/182 components during the Fall 2004 Unit 2 RFO. The hot and cold leg temperatures were reported to be 617°F and 588°F, respectively. It was noted that the leakage would have gone undetected if the surrounding insulation had not been removed to facilitate the inspections. No response from PWR Licensees was requested.

REVISION SUMMARY

Procedure No.: EHI-5070-ALLOY600 Rev. No.: 4
Title: Alloy 600 Material Management Program

Alteration	Justification
10 CFR 50.59 is not applicable to this procedure due to Inservice Inspection Program governing requirements in this procedure and the procedure is a Managerial/Administrative procedure governing the conduct of facility operations per Attachment 1 of PMP-2010-PRC-002.	
Step 1.7 – Added reference to 10 CFR 50.55a.	In response to an RAI requesting CNP provide this information when referencing these Code Cases.
Step 1.7 – Removed reference to previous inspection requirements (MRP-139).	This information is not necessary in the procedure.
Section 2, PWSCC – Corrected 'phenomena' to 'phenomenon'	Correct to word usage Editorial Correction Criteria 'a'
Step 3.5.4 – Added reference to NRC regulations and generic communications.	In response to an RAI requesting CNP include this information.
Section 4 (Details) – Reordered the previous subsections (4.3 Inspections, 4.4 PWSCC Susceptibility Rankings, 4.6 Mitigation)	Placed subsections in a more appropriate order. Margin marks not used.
Step 4.6.1 – Reworded step.	To clarify the information actually contained in Attachment 2.
Step 4.6.2, bulleted steps – Added reference to 'CNP's Inservice Inspection Program' and 'other relevant NRC communications' and reordered the list	To include additional sources of inspection requirements.
Section 4.7 – Added new section for acceptance criteria.	In response to an RAI requesting CNP include a section for acceptance criteria.
Section 4.8 – Revised Steps 4.8.2, 4.8.3, and 4.8.4 and deleted former Step 4.7.5	To make information more concise and accurate.
Added References 5.2.1i and 5.2.1j	Added references Editorial Correction Criteria 'n'
Attachment 2 – Completely revised table. Added references to 10 CFR 50.55a. Added 'acceptance criteria' and 'comments' columns. Added description of several Steam Generator inspections.	In response to an RAI requesting CNP include this information. Due to the magnitude of the changes, margin marks not used.
Attachment 5 (page 25) – Corrected the timeframe listed for U2 RVCH replacement.	Unit 2 RVCH was replaced during U2C17 not U2C16.