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Indiana Michigan Power
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AEP-NRC-2011-39
10 CFR 50.54

Docket No.: 50-315
50-316

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

Donald C. Cook Nuclear Plant Units 1 and 2
LICENSE RENEWAL SUBMITTAL - ALLOY 600 AGING MANAGEMENT PROGRAM

References:

1. Letter from M. W. Rencheck, Indiana Michigan Power Company, to U. S. Nuclear Regulatory Commission Document Control Desk, "Donald C. Cook Nuclear Plant Units 1 and 2, Docket No. 50-315 and 50-316, Application for Renewed Operating License," AEP:NRC:3034, dated October 31, 2003 (ADAMS Accession Number ML033070177)

2. Letter from J. Rowley, U. S. Nuclear Regulatory Commission Office of Nuclear Reactor Regulation, to M. K. Nazar, Indiana Michigan Power Company, "Request for Additional Information for the Review of Donald C. Cook Nuclear Plant, Units 1 and 2, License Renewal Application," dated July 2, 2004 (ADAMS Accession Number ML041840194)

3. Letter from J. N. Jensen, Indiana Michigan Power Company, to U. S. Nuclear Regulatory Commission Document Control Desk, "Donald C. Cook Nuclear Plant Units 1 and 2, Docket No. 50-315 and 50-316, License Renewal Application – Response to Requests for Additional Information on Aging Management Programs, (TAC Nos. MC1202 and MC1203)," AEP:NRC:4034-10, dated August 11, 2004 (ADAMS Accession Number ML042470410)

By Reference 1, Indiana Michigan Power Company (I&M) submitted an application pursuant to 10 CFR Part 54, to renew the operating licenses for Donald C. Cook Nuclear Plant (CNP) Units 1 and 2, for review by the U. S. Nuclear Regulatory Commission (NRC). As a part of that application, I&M committed to implement an Alloy 600 Aging Management Program prior to the period of extended operation. By Reference 2, the NRC requested that I&M modify the commitment to state that lessons learned from industry initiatives and research would be used as a part of the Alloy 600 Aging Management Program and to submit the program to the NRC for staff review and approval three years prior to the period of extended operation. By Reference 3, I&M agreed to the commitment change request and modified the commitment accordingly.

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The enclosure to this letter provides CNP procedure EHI-5070-ALLOY600, Alloy 600 Material Management Program. This program is based on Electric Power Research Institute (EPRI) MRP-126, Generic Guidance for Alloy 600 Management, November 2004. Submittal of this procedure implements the commitment described above.

There are no new or revised commitments in this letter. Should you have any questions, please contact Mr. Michael K. Scarpello, Regulatory Affairs Manager, at (269) 466-2649.

Sincerely,

Joel P. Gebbie
Site Vice President

MCS/jen

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

Enclosure: EHI-5070-ALLOY600, Alloy 600 Material Management Program

Enclosure to AEP-NRC-2011-39

EHI-5070-ALLOY600
ALLOY 600 MATERIAL MANAGEMENT PROGRAM

Alloy 600 Material Management Program

Information

Debra J Poupard	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 Cook Nuclear Plant (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 (supersede EPRI MRP-139, Material Reliability Program: Primary System Piping Butt Weld Inspection and Evaluation Guideline, December 2008).

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, phenomena 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 Determines appropriate sample locations, inspection techniques, and acceptance standards in accordance with industry guidelines.

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 Inspections

4.3.1 The current inspection frequency for components in the Alloy 600 Material Management Program is included in Attachment 2, Inspections.

4.3.2 Sources of inspection requirements may include:

- ASME Boiler and Pressure Vessel Code, Section XI
- NRC Orders
- License Renewal Programs
- Plant Procedures and Programs
- Joint Industry Issues Programs (EPRI Material Reliability Program)
- NRC Regulations (10 CFR 50.55a)

4.4 PWSCC Susceptibility Rankings

4.4.1 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.

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

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

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 Mitigation

4.6.1 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.

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

4.6.3 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

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

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

4.7 Corrective Measures

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

4.7.2 **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].

4.7.3 Such an evaluation is to consider the significance of the test results, the operability of the component, the reportability of the event, the extent of the concern, the potential root causes for not meeting the test acceptance criteria, the corrective actions required, and the likelihood of recurrence.

4.7.4 When an unacceptable condition or situation is identified, a determination is made as to whether the same condition or situation is applicable to other components within the scope of the program.

4.7.5 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.

4.8 Records

4.8.1 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)

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	Required Inspection Frequency	Current CNP Inspection Frequency
Hot and Cold Leg Nozzles	1	Volumetric	ASME Code Case N-770-1, Item D	Per required inspection frequency
BMI Penetrations	1, 2	BMV	ASME Code Case N-722-1, Item B15.80: every other refueling outage	Every refueling outage due to recurring refueling cavity leakage issues during refueling activities
Leak-off Monitor Tubes	1, 2	VT-2	No requirement	Visual inspection every refueling outage during initial depressurized VT-2 inspection under the ISI Program
RVCH Nozzles and Welds	1, 2	Volumetric, Surface	ASME Code Case N-729-1, Item B4.40: once per interval	Per required inspection frequency
RVCH	1, 2	BMV	ASME Code Case N-729-1, Item B4.30: every third refueling outage or 5 calendar years, whichever is less	Per required inspection frequency
Core Support Pad Welds	1, 2	VT-3	ASME Section XI, Category B-N-2, Item B13.60: once per interval	Per required inspection frequency
Clevis Inserts	1, 2	VT-3	ASME Section XI, Category B-N-2, Item B13.60: once per interval	Per required inspection frequency
Pressurizer Nozzle (1-PRZ-23) to Safe-End Weld	1	Volumetric	ASME Code Case N-770-1, Item F	Per required inspection frequency
Pressurizer Nozzle to Safe-End Welds (except 1-PRZ-23)	1, 2	Volumetric	ASME Code Case N-770-1, Item C	Per required inspection frequency

Location	Unit	Method	Required Inspection Frequency	Current CNP Inspection Frequency
SG Divider Plate	1, 2	Visual	No requirement	Remote as-found visual scan during scheduled steam generator primary side activities
SG Tubesheet Cladding	1, 2	Visual	No requirement	Remote as-found visual scan during scheduled steam generator primary side activities
SG Primary Manway Diaphragm	1	Visual	No requirement	Visual scan during scheduled steam generator primary side activities
SG Primary Nozzle to Safe-End Weld	1	Volumetric, Surface	ASME Code Program: each inspection interval	Per required inspection frequency
SG Nozzle Dam Rings	1, 2	Visual	No requirement	Remote as-found visual scan during scheduled steam generator primary side activities
SG Tubes	1, 2	Volumetric	Technical Specifications Section 5.5.7: Inspect 100% of the tubes at sequential periods of 144, 108, 72, and, thereafter, 60 effective full power months. Inspect 50% of the tubes by the refueling outage nearest the midpoint of the period and the remaining 50% by the refueling outage nearest the end of the period.	Per required inspection frequency
SG Shell Cladding	2	Visual	No requirement	Remote as-found visual scan during scheduled steam generator primary side activities

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 U2C16 refueling outage (Spring 2006).

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.: 3
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.	
This revision is a major revision to update and enhance EHI-5070-ALLOY600, Material Management Program, to address NRC requirements in Commitment #8244. No margin marks are used for this revision.	NRC Commitment #8244 requires that an Alloy 600 inspection plan be submitted for staff review and approval three years prior to the period of extended operation. Cook Nuclear Plant will submit Revision 3 of EHI-5070-ALLOY600, Material Management Program, to the NRC to meet this commitment. Revision 3 of EHI-5070-ALLOY600 aligns the CNP Alloy 600 Material Management Program with inspection plans from other plants already submitted to the NRC. The requirement to develop and submit an Alloy 600 inspection plan is tracked under GT 00113534.

Office Information for Form Tracking Only - Not Part of Form

This is a free-form as called out in PMP-2010-PRC-002, Procedure Alteration, Review, and Approval.