FPL Energy Point Beach, LLC, 6610 Nuclear Road, Two Rivers, WI 54241

FPL Energy
Point Beach Nuclear Plant

October 6, 2008

NRC 2008-0070
10 CFR 54

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

Point Beach Nuclear Plant, Units 1 and 2
Dockets 50-266 and 50-301
Renewed License Nos. DPR-24 and DPR-27

License Renewal Commitment
Alloy 600 Program Submittal

References: (1) Nuclear Management Company, LLC letter to NRC dated January 25, 2005, Response to Request for Additional Information Regarding the Point Beach Nuclear Plant License Renewal Application (ML050340198)
(2) NUREG-1839, Safety Evaluation Report Related to the License Renewal of Point Beach Nuclear Plant, Units 1 and 2, dated December 2005 (ML053420129)

In Reference 1, Nuclear Management Company, LLC (NMC), the former license holder for Point Beach Nuclear Plant (PBNP), committed to implementing an Alloy 600 Inspection Program as part of the license renewal process. Specifically, NMC committed to using the Electric Power Research Institute Materials Reliability Project (EPRI MRP) interim report on Alloy 600 (MRP-44), and its final version, as the basis for a Reactor Coolant System Alloy 600 Inspection Program. NMC also committed to submit a Reactor Coolant System Alloy 600 Inspection Program 24-36 months prior to the period of extended operation for NRR staff review and approval.

Enclosure 1 provides PBNP procedure AM 3-31, Alloy 600 Management Program. This program is based on MRP-126, Generic Guidance for Alloy 600 Management, which is the final version of MRP-44.

Summary of Regulatory Commitments

This letter contains no new commitments. This letter implements the following commitments:

NMC will use the interim report "PWR Materials Reliability Project Interim Alloy 600 Safety Assessment for US PWR plants (MRP-44), Part 1: Alloy 82/182 Pipe Butt Welds," and its final version as part of the basis for the Reactor Coolant System Alloy 600 Inspection Program.

NMC will submit the Reactor Coolant System Alloy 600 Inspection Program 24-36 months prior to the period of extended operation for staff review and approval to determine if the program demonstrates the ability to manage the effects of aging per 10 CFR 54.21(a)(3)

I declare under penalty of perjury that the foregoing is true and correct. Executed on October 6, 2008.

Very truly yours,

FPL Energy Point Beach, LLC

Larry Meyer
Site Vice President

Enclosure

cc: Region III Administrator, USNRC
Project Manager, Point Beach Nuclear Plant, USNRC
Resident Inspector, Point Beach Nuclear Plant, USNRC

ENCLOSURE

FPL ENERGY POINT BEACH, LLC
POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2

AM 3-31, REVISION 1
ALLOY 600 MANAGEMENT PROGRAM

28 pages follow

AM 3-31

ALLOY 600 MANAGEMENT PROGRAM

DOCUMENT TYPE:	Administrative
REVISION:	2
EFFECTIVE DATE:	October 3, 2008
APPROVAL AUTHORITY:	Department Manager
PROCEDURE OWNER (title):	Group Head
OWNER GROUP:	Program Engineering

TABLE OF CONTENTS

1.0 PURPOSE

1.1 Scope

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

This document also implements a commitment to the NRC to manage the effects of aging for SCC's within the scope of License Renewal (LR) as described in NP 7.7.25, PBNP Renewed License Program. Applicable LR commitments require the implementation of an Alloy 600 Program. (B-1 B-2)

The program is focused on both pressure and non-pressure boundary Reactor Coolant System (RCS) components constructed of Alloy 600 and welds constructed of the associated Alloy 82/182 filler metals. Industry experience has shown these materials to be susceptible to failure by primary water stress corrosion cracking (PWSCC). Steam generator tubing is excluded from this program because it is covered under the Steam Generator Integrity Program.

This program was developed utilizing the EPRI MRP-126 "Generic Guidance for Alloy 600 Management" industry guidance document, and NEI 03-08 "Guideline for the Management of Materials Issues." The Alloy 600 Management Program is a living document and will be revised periodically to reflect the latest plant configurations.

1.2 Objective

The overall objectives of the Alloy 600 Management Program are as follows:

1.2.1 Maintain the integrity and operability of Alloy 600/82/182 materials

1.2.2 Ensure regulatory compliance

1.2.3 Maintain plant safety

1.2.4 Minimize the impact of PWSCC on plant availability

2.0 DISCUSSION

2.1 Applicability

The Alloy 600 Management Program includes the management of short and long term examination, evaluation, mitigation, and repair/replacement activities. Implementation of these activities is controlled by other programs and procedures.

2.2 Definitions

None

3.0 RESPONSIBILITIES

Chief Nuclear Officer and Vice President, Nuclear Technical Services are ultimately responsible for the successful implementation of the Alloy 600 Program.

The overall responsibility for the development, revision and implementation of the Alloy 600 Management Program resides with the Program Engineering Department. Responsibilities of the various groups contained therein are described below.

3.1 Programs Engineering Department

3.1.1 Preparation, maintenance and ownership of the Alloy 600 Management Program

3.1.2 Development of refueling outage examination plans

3.1.3 Development of a recommended strategy for the management of Alloy 600/82/182 materials

3.1.4 Ensuring compliance with regulatory requirements

3.1.5 Serving as the contact for outside technical communications (NEI, INPO, NRC, EPRI, ASME, PWR Owners Group, etc.)

3.1.6 Participating in industry owners groups

3.1.7 Providing analysis and response to significant industry events

3.1.8 Preparing periodic program health reports

3.1.9 Conducting periodic self-assessments of the Alloy 600 Management Program

3.2 Plant Engineering

3.2.1 Preparation of Design Change Packages (DCP) packages for repairs or modifications that would result in a configuration change to existing Alloy 600/82/182 components/welds.

3.2.2 Disposition of Condition Reports associated with examination results

3.3 Site Maintenance and Projects Departments

3.3.1 Performance of work orders for the implementation of examination, evaluation, mitigation and repair/replacement activities

4.0 PROCEDURE

4.1 Industry Experience

4.1.1 Construction

Alloy 600/82/182 materials were incorporated into the RCS of Westinghouse (PBNP 1 & 2) 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

The Westinghouse design utilized Alloy 600/82/182 for the RPV penetrations, the Bottom-Mounted Instrumentation (BMI) penetrations, and to a lesser extent some RCS piping connections. A complete listing of the Alloy 600/82/182 locations at PBNP 1 & 2 is located in Attachment C.

4.1.2 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

a. Thermo-mechanical processing variables utilized during the fabrication of Alloy 600 components directly affect the materials microstructure and degree of cold work. A high temperature mill anneal produces a microstructure that has been found to be more resistant to PWSCC than one resulting from lower mill anneal temperatures. High degrees of cold work and lower forging temperatures have also been found detrimental to PWSCC resistance.

b. PWSCC susceptibility is directly proportional to higher total stress levels, including both applied and residual. Furthermore, higher yield strength often correlates with a shorter PWSCC initiation time because it allows the material to retain higher residual stress levels from welding and machining processes.

c. The normal primary water environment is fully capable of supporting PWSCC. Additionally, contaminants or chemical additives such as sulfate, lead, and hydrogen in primary water may accelerate PWSCC. The Primary Chemistry Control Program alone is ineffective for prevention of PWSCC in Alloy 600/82/182 materials.

d. PWSCC susceptibility is directly proportional to temperature as the mechanism is a thermally activated process. While there is no proven minimum temperature for PWSCC, 561°F is the lowest temperature at which it has been observed in service.

4.1.3 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 compilation of domestic and foreign PWR components that were repaired/replaced due to PWSCC or concerns thereof is included in Table 3-1 of EPRI MRP 76. A listing of Alloy 600 components that have been replaced at PBNP due to PWSCC, or concerns thereof, is included as Attachment A.

4.1.4 NRC Communications

Since the early 90's, the NRC has issued a significant number of generic communications to PWR licensees concerning PWSCC of Alloy 600/82/182 materials. Given the generic nature of this issue, joint industry issue programs organized through NSSS owners groups, EPRI and NEI have been, and continue to be, instrumental in investigating the issue and addressing the NRC's safety concerns. Summaries of the generic communications and PBNP's responses, when they were required, are included in Attachment B.

4.2 Alloy 600/82/182 Locations

A comprehensive list of the Alloy 600/82/182 locations for PBNP Units 1 and 2 is provided in Attachment C. These locations include the following:

4.2.1 Reactor Pressure Vessel Shells

The Unit 1 lower shell course is internally clad with Alloy 82/182 on the bottom 11-7/8 inches. Four (4) core support guides made from Alloy 600 are welded to the bottom of the shell course. (Reference 5.1.4)

The Reactor Pressure Vessel Shells have no Alloy 600 penetrations,

4.2.2 Reactor Vessel Internals

The Reactor Vessel Clevis Insert Lock Keys and Clevis Inserts are manufactured from Alloy 600 material for both units.

4.2.3 Reactor Vessel Heads

The Reactor Vessel Upper Heads were replaced in 2005. The thirty-eight (38) upper vessel head penetrations are Alloy 690 material attached to the upper head with J-groove welds using Alloy 52/152 filler material. Only Alloy 52 weld filler metal is exposed to primary water. (References 5.1.5, 5.1.6)

Thirty-six (36) lower head Bottom-Mounted Instrumentation (BMI) penetrations are Alloy 600 penetrations. (References 5.1.3, 5.1.4)

4.2.4 Steam Generators

Each Unit 1 Steam Generator has one (1) Alloy 82/182 weld, which are around the coupling on the Channel Head Bowl Drain Lines. Alloy 600/82/182 is also located in the Unit 1 Steam Generator channel head divider plate weld and nozzle dam rings.

Each Unit 2 Steam Generator has two (2) Alloy 82/182 welds and two (2) Alloy 600 penetrations.

- The Alloy 82/182 welds are for the primary nozzles to safe-end on each hot and cold leg. These welds are inlaid with Alloy 52/152.

- The Alloy 600 penetrations are the primary vent nozzles on each cold and hot leg side of the Steam Generator Channel Head.

4.2.5 Pressurizers, Reactor Coolant Pumps, and Reactor Coolant Piping

The pressurizers, reactor coolant pumps and reactor coolant piping have no Alloy 600 penetrations or Alloy 82/182 weld metal.

4.3 Inspection Requirements

Current examination requirements for the various Alloy 600/82/182 locations are included in Attachment C. Given the emergent nature of the Alloy 600 issue throughout the industry, these listings will likely require ongoing revision upon issuance of new examination requirements. Sources of these examination requirements include:

a. ASME Section XI In-Service Inspection (ISI) Program

b. NRC Orders

c. License Renewal Programs

d. Plant Procedures and Programs

e. Joint Industry Issues Programs (i.e. MRP-139)

f. NSSS Vendors

g. NRC Regulations (10 CFR 50.55a)

4.3.1 ASME Section XI ISI Program

10 CFR 50.55a requires that all power reactors maintain an ISI program in accordance with the ASME Boiler and Pressure Vessel Code, Section XI. Applicable requirements for Alloy 600/82/182 components addressed by this program (Class 1) are included in IWB-2500 of Section XI.

Code Case N-722 requires the performance of visual examinations (VT) of highly susceptible Alloy 600/82/182 components during each refueling outage (hot leg temperature and above). Other Alloy 600/82/182 components that are considered less susceptible to PWSCC cracking (e.g., cold leg instrument connections) are required to be examined by VT once per interval, with the exception of BMI's which are every other outage.

Code Case N-729-1 requires the performance of visual, surface, or volumetric examination of the Reactor Vessel Upper Head Nozzles having Pressure-Retaining Partial Penetration Welds. The examination technique used is based on the desired frequency of examination, material, and effective degradation years (EDY). Code Cases N-722 and N-729-1 were incorporated into 10 CFR 50.55a on September 10, 2008.

4.3.2 NRC Orders

Orders issued to date concerning examination of Alloy 600/82/182 materials include EA-03-009 (February 11, 2003) and EA-03-009, Rev. 1 (February 20, 2004). Both addressed reactor pressure vessel head penetrations. Once Code Case N-729-1 is incorporated into the PBNP ISI program, it will replace NRC Order EA-03-009.

4.3.3 License Renewal Programs

The license renewal processes conducted at PBNP 1 & 2 created a number of programs to ensure that the integrity of structures and components is maintained throughout the periods of extended operation at both sites. Specific programs concerning Alloy 600/82/182 materials include:

- LR-AMP-017-IWBCD, ASME Section XI, Subsections IWB, IWC, and IWD Inservice Inspection Program Basis Document for License Renewal
- LR-AMP-005-BAC, Boric Acid Corrosion Program Basis Document for License Renewal
- LR-AMP-013-RCA600, RCS Alloy 600 Inspection Program Basis Document for License Renewal

4.3.4 Procedures and Programs

Procedures and programs developed to verify the integrity of the RCS and minimize the chances of equipment degradation due to boric acid corrosion include: individual site program documents for Boric Acid Corrosion Control (BACC), Reactor Coolant System Leak Test, BMI Examination, and RPV Closure Head examinations.

4.3.5 Joint Industry Issues Programs

Joint industry issues programs are often utilized to address the degradation of Alloy 600/82/182 in the most cost effective and efficient manner. Applicable issues programs and their current examination requirements include:

a. EPRI Materials Reliability Program (MRP)

- MRP 2003-17, "Recommendation for PWR Owners with Alloy 600 Bottom Mounted Reactor Vessel Instrument Nozzles," (June 23, 2003) - Bare metal visual examination of any Alloy 600 nozzles penetrating the bottom head of the reactor pressure vessel (BMI's) during the current or next refueling outage *is recommended.*

- MRP 2003-039, "Recommendation for Inspection of Alloy 600/82/182 Pressure Boundary Components," (January 20, 2004) - Direct visual examination of the bare metal or equivalent alternative examination *shall be performed* on all Alloy 600/182/82 pressure boundary locations with normal operating temperatures $\geq 350^{\circ}$ F within next two refueling outages.

- MRP 2004-05, "Visual Inspection of Alloy 82/182 Butt Welds and Good Practice Recommendations for Weld Joint Configurations," (April 2, 2004) - Direct visual inspection of the bare metal or equivalent alternative examination *shall be performed* on all Alloy 82/182 pressure boundary butt weld locations with normal operating temperatures $\geq 350^{\circ}$ F within next two refueling outages. During examination, plant specific information on weld joint configurations and available access shall be gathered to prepare for future volumetric examinations.

  In response to MRP 2004-05 and MRP 2003-039, a bare metal visual examination of Alloy 82/182 pressure boundary butt welds was performed. These exams gathered weld joint configuration information and digital photos of the welds were taken. These exams were performed during the U1R29 Fall 2005 and U2R27 Spring 2005 refueling outages.

- MRP 2004-04, "BMI Inspection Plant," (May 14, 2004) - As a follow-up to MRP 2003-17, licensees/utilities with an upcoming 10 year ISI should plan to supplement the bare metal visual examinations of Alloy 600 BMI's with volumetric examinations.

- MRP 2005-014, "Primary System Piping Butt Weld Inspection and Evaluation Guideline (MRP-139)," (September 12, 2005) – As a follow-up to MRP 2004-05, for each butt weld location, MRP-139 visual examinations will be performed during the next RFO following successful completion of the initial ultrasonic examination.

- MRP 2007-038, "MRP-139 Interim Guidance on <4" Volumetric Exam Requirements," (November 1, 2007) - As a follow-up to MRP 2004-05, DM butt welds greater than or equal to 2" NPS in the following service conditions and not already included within the volumetric examination requirements of MRP-139 should be added:
  - Those at pressurizer temperatures
  - Those at hot leg temperatures
  - Those that serve an ECCS function
- MRP 2007-039, "MRP-139 Interim Guidance on Bare Metal Visual Exam Requirements," (November 1, 2007) – As a follow-up to MRP-2005-014, initial BMV inspection requirements were not provided for butt-welds ≥ 1" but < 4" not having additional volumetric inspections. Therefore, the interim guidance specifies requirements for these small diameter welds *and is mandatory* per NEI-03-08 guidelines.

4.4 Repair Methods

Since the first pressurizer instrumentation nozzle failure in 1986, PWR licensees have implemented a variety of repair methods. Selection of the optimum repair method is normally based upon available technology, ASME Code requirements, radiological conditions, and economic factors. Most repairs implemented since the mid 1990's have utilized only PWSCC resistant Alloy 690/52/152 materials. Summaries of the most common methods for the various Alloy 600 material locations are provided below. Repairs implemented to date at PBNP are included in Attachment A.

4.4.1 RPV Upper Head CRDM Nozzles

PBNP completed a program of RPV head replacements. The replacement RPV heads are constructed using Alloy 690 (vice Alloy 600) nozzles attached using A52/152 weld material vice (A82/182). Only A52 weld material comes in direct contact with the primary water The PBNP Unit 1 and 2 RPV Heads were replaced in 2005.

4.4.2 RPV Lower Head BMI Nozzles

The PBNP 1 and 2 Alloy 600 BMI nozzles are attached to the lower RPV vessel using J-groove welds.

a. Full Nozzle Repair

The failed nozzle is replaced in its original configuration. Radiological conditions would likely make this method impractical.

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.

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 an Alloy 690 (Filler Metal 52). 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 bottom mounted nuclear instrumentation nozzle BMI repairs.

4.5 <u>Mitigation Methods</u>

There are a number of mitigation methods for PWSCC that may provide cost effective alternatives to the replacement of Alloy 600 components. Most of these methods have previously been utilized to address IGSCC of austenitic SS in BWRs. Their functions vary from providing preventative benefits to total structural replacement. Mitigation methods include.

4.5.1 Mechanical Stress Improvement (MSIP)

4.5.2 Induction Heating Stress Improvement (IHSI)

4.5.3 Weld Overlay

4.5.4 Mechanical Nozzle Seal Assembly (MSNA)

4.5.5 Zinc Injection

4.5.6 Abrasive Water Jet (AWJ)

4.5.7 Nickel Plating

4.6 PWSCC Susceptibility Ranking

4.6.1 Historical Ranking Models

The NSSS owner's groups and EPRI developed a number of PWSCC ranking models following the discovery of PWSCC in RPV upper head nozzles at Bugey and several other foreign PWRs in the early 1990's. The models attempted to incorporate differences in operating time and temperature, water chemistry environment, surface stress, component geometry, material yield strength and microstructure, and fabrication practices (amount of cold work during machining). Unfortunately, uncertainties about surface stress state, microstructure and fabrication practices introduced significant error into all these models.

Subsequent to the discovery of circumferential cracking in CRDM nozzles at Oconee-3, the EPRI Materials Reliability Program (MRP) submitted the MRP-44, Part 2 report to provide an interim safety assessment for PWSCC of alloy RPV upper head nozzles and associated Alloy 182 J-groove welds in PWR plants. This report included a simplified ranking model based only upon the operating time and temperature of the RPVH penetrations, effective full power years (EFPY). This model was later challenged by the discovery of PWSCC in three RPV upper head penetrations at Millstone Unit 2 in 2002 which had been ranked as one of the least susceptible plants.

In Bulletin 2002-02, the NRC described of a comprehensive RPV upper head examination program that addressed a combination of visual and non-visual examinations on a graded approach based upon plant susceptibilities to PWSCC. This Bulletin introduced a time at temperature model, effective degradation years (EDY), to characterize plant susceptibility. This same model was included in the subsequent Orders (EA-03-009 Revs. 0 & 1) to determine the frequency and type of examinations for RPV head penetrations at individual plants.

5.0 REFERENCES

5.1 Source Documents

5.1.1 CIM-00109, WEST, Unit 2 Steam Generators

5.1.2 CIM-00112, CE, Unit 2 Reactor Vessel

5.1.3 CIM-00210, B/W, Unit 1 Reactor Vessel

5.1.4 WCAP-16345-P, Nuclear Management Company – Point Beach Unit 1 Nuclear Power Plant Replacement Reactor Vessel Closure Head – Design Report

5.1.5 WCAP-16266-P, Nuclear Management Company – Point Beach Unit 2 Nuclear Power Plant Replacement Reactor Vessel Closure Head – Design Report

5.1.6 Westinghouse Letter WEP-02-8, "Point Beach Units 1 and 2 Reactor Internals CMTR Summary," dated September 12, 2002

5.2 Reference Documents

5.2.1 EPRI MRP-126, "Generic Guidance for Alloy 600 Management"

5.2.2 NEI 03-08, "Guideline for the Management of Materials Issues."

5.2.3 EPRI MRP-227, "PWR Internals Inspection and Evaluation Guidelines"

5.2.4 EPRI MRP-44, "PWR Materials Reliability Project Interim Alloy 600 Safety Assessments for US PWR Plants"

5.2.5 EPRI MRP-139, "Materials Reliability Program: Primary System Piping Butt Weld Inspection and Evaluation Guidelines"

5.3 Records

None

6.0 BASES

B-1 LR-AMP-0113-RCA600, RCS Alloy 600 Inspection Program Basis Document for License Renewal

B-2 NUREG-1839, "US NRC Safety Evaluation Report Related to the License Renewal of the Point Beach Nuclear Plant, Unit 1 and 2."

ATTACHMENT A
ALLOY 600 REPAIRS/REPLACEMENTS

Unit	Location	Tag ID	Replacement Date	Replacement Method	Inconel Buildup Pad	Design Document	Reason for Replacement
PB-1	Reactor Pressure Vessel Head		11/05	New Head	-	-	Modification MR 03-047
PB-2	Reactor Pressure Vessel Head		7/05	New Head	-	-	Modification MR 03-056

ATTACHMENT B
NRC GENERIC COMMUNICATIONS

NRC Information Notice (IN) 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 PWSCCC 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 (GL) 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.

PBNP Response:

Letter No. NPL 97-0420, dated July 30, 1997

NRC IN 2000-17 (October 18, 2002), "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 ID initiated axial indication the Alloy 182/82 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.

ATTACHMENT B
NRC GENERIC COMMUNICATIONS

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 IN 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 August 16, 2002 that PBNP provided the requested information.

In response to NRC Bulletin 2001-01, a bare metal visual examination of the RPV upper head was performed during the Unit 2 Spring 2002 outage and the Unit 1 Fall 2002 refueling outage with acceptable results. Reactor Vessel Head Inspection Findings were provided to the NRC in letters NRC 2002-0050 and NRC 2002-0102.

PBNP Responses:

Attachment 2 to Letter No. NRC 2001-060, dated September 4, 2001

Attachment 2 to Letter No. NRC 2002-0002, dated January 3, 2002

Letter No. NRC 2002-0011, dated January 28, 2002

Letter No. NRC 2002-0038, dated May 09, 2002

Letter No. NRC 2002-0050, dated June 12, 2002 – Unit 2 Reactor Vessel Head Inspection Findings

Letter No. NRC 2002-0102, dated November 15, 2002 – Unit 1 Reactor Vessel Head Inspection Findings.

ATTACHMENT B
NRC GENERIC COMMUNICATIONS

NRC IN 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" SS 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.

PBNP Responses:

Letter No. NRC 2002-0027, dated April 2, 2002

Letter No. NRC 2002-0029, dated April 18, 2002

Letter No. NRC 2002-0037, dated May 9, 2002

Attachment 2 to Letter No. NRC 2002-043, dated May 16, 2002

Letter No. NRC 2002-0050, dated June 12, 2002

Letter No. NRC 2002-0102, dated November 15, 2002

Letter No. NRC 2003-0006, dated January 20, 2003

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NRC GENERIC COMMUNICATIONS

NRC IN 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.

In response to NRC Bulletin 2002-02, an ultrasonic examination of the vessel head penetration (VHP) nozzle base material and a supplemental ultrasonic leak path examination of the interference region of the VHP penetrations will be performed. These examinations were started for Unit 1, during the U1R27 Fall 2002 refueling outage and were performed on a refueling outage interval until the reactor pressure vessel head was replaced during U1R29 (Fall 2005). These examinations will continue to be performed in accordance with the Revised NRC Order (EA-03-009). During the Unit 1 refueling outage (U1R28) Spring 2004, the UT examinations showed a flaw in penetration 26 that exceeded the acceptance criteria of the original design and repairs were made under modification MR 03-041. These examinations were also started for Unit 2, during the U2R26 Fall 2003 refueling outage and were performed on a refueling outage interval until the reactor pressure vessel head was replaced during U2R27 (Fall 2005). These examinations will continue to be performed in accordance with the Revised NRC Order (EA-03-009).

PBNP Responses:

Letter No. NRC 2002-0082, dated September 12, 2002

Letter No. NRC 2002-0102, dated November 15, 2002

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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.

NRC Regulatory Issue Summary (RIS) 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 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) penetration 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 NRC 2004-0006 and NRC 2004-0077. The NRC replied in a letter dated November 22, 2004 that PBNP met the reporting requirements of this Bulletin.

In response to NRC Bulletin 2003-02, a bare metal visual examination of the RPV lower head and BMI nozzles were performed during the Unit 2 October 2003 outage and the Unit 1 April 2004 refueling outage with acceptable results. Each of the 36 BMI nozzles per head were examined with VT-1 quality resolution 360 degrees around their circumference, as well as all bare metal for at least six (6) to twelve (12) inches above the highest BMI.

PBNP Responses:

Letter No. NRC 2003-0089, dated September 22, 2003

Letter No. NRC 2004-0006, dated January 15, 2004 – Unit 2 Reactor Vessel Inspections

Letter No. NRC 2004-0077, dated August 6, 2004 – Unit 1 Reactor Vessel Inspections

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NRC IN 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 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.

PBNP Response:

Letter No. NRC 2004-0023, dated March 10, 2004

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.

PBNP responded to Bulletin 2004-01 indicating that no Alloy 82/182/600 materials exist in the PBNP Unit 1 and Unit 2 pressurizers. The NRC replied in a letter dated March 7, 2006 that, based on the responses to items 1a, 1b, 1c, and 1d of the Bulletin, the NRC staff no longer requires a specific response for PBNP for item 2 of the Bulletin.

PBNP Response:

Letter No. NRC 2004-0075, dated July 23, 2004

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NRC IN 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-1). 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 IN 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.

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Information		Description			PWSCC Susceptibility		
Weld number	**Location**	**Material**	**Pipe Dia.**	**Thickness**	**Operating Temperature**	**Failure Consequence**	**Ranking (Low, Moderate, High, Very High)**
Unit 1							
Vessel Head Penetrations (38)	Reactor Vessel Head	A690 A52/152	4"	.625"	598°F	B,E,G	**Low** – Resistant Material
BMI Nozzles (36)	Bottom Mounted Instrumentation	SB-166	1.5"	.264"	540°F	B,E,G	**Moderate** -(Low Temperature, Low Probability of failure, good industry exam record)
Internal clad	Bottom 11-7/8 inches of lower shell course	A82/182	N/A	N/A	540°F	None	**Low** - (No Industry OE of Failure, Low Consequence, Not Pressure Boundary)
SG Channel Head Drains	1 per SG	A82/182	.375"	0.091"	584°F	B,E,G	**High** - (High Temperature, Some Industry OE or recent failures, Low Consequence)
SG Channel Head Divider Plate	1 per SG	SB-168	N/A	2.0"	584°F	None (under evaluation)	**Low** – (Foreign OE exists, No domestic OE of failure, Not Pressure Boundary]
RV Clevis Insert Lock Keys	Reactor Vessel	A600	N/A	N/A	540°F	G	**Low** - (No Industry OE of Failure, Low Consequence, Not Pressure Boundary)
RV Clevis Inserts	Reactor Vessel Internals	A600	N/A	N/A	540°F	G	**Low** - (No Industry OE of Failure, Low Consequence, Not Pressure Boundary)
SG Nozzle Dam Rings	SG Nozzles	A600	N/A	NA	540°F / 584°F	G	**Low** - (No Industry OE of Failure, Low Consequence, Not Pressure Boundary)

B – Causes a design-basis accident
E – Breaches reactor coolant pressure boundary integrity
G – Causes a significant economic impact. Significant events are those for which we do not have a proven fix and would result in significant regulatory and/or public scrutiny, such as first-of-a-kind consideration would be a suitable test. It can be considered that "non-significant" events are those for which it is expected that a proven fix exists that will require minimal regulatory and/or public scrutiny.

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Information		Description			PWSCC Susceptibility		
Weld number	**Location**	**Material**	**Pipe Dia.**	**Thickness**	**Operating Temperature**	**Failure Consequence**	**Ranking (Low, Moderate, High, Very High)**
Unit 2							
Vessel Head Penetrations (38)	Reactor Vessel Head	A690 A52/152	4"	.625"	598°F	B,E,G	**Low** – Resistant Material
BMI Nozzles (36)	Bottom Mounted Instrumentation	SB-166	1.5"	.264"	540°F	B,E,G	**Moderate** - (Low Temperature, Low Probability of failure, good industry exam record)
RC-34-MRCL-AI-05	SG 'A' Hot Leg S/G Primary Nozzle Safe-End Weld	A82/182	34"	3" nominal	584°F	B,E,G	**Low** - (Clad with Alloy 52 during initial fabrication)
RC-36-MRCL-AII-01A	SG 'A' Cold Leg S/G Primary Nozzle Safe-End Weld	A82/182	36"	3" nominal	540°F	B,E,G	**Low** - (Clad with Alloy 52 during initial fabrication)
RC-34-MRCL-BI-05	SG 'B' Hot Leg S/G Primary Nozzle Safe-End Weld	A82/182	34"	3" nominal	584°F	B,E,G	**Low** - (Clad with Alloy 52 during initial fabrication)
RC-36-MRCL-BII-01A	SG 'B' Cold Leg S/G Primary Nozzle Safe-End Weld	A82/182	36"	3" nominal	540°F	B,E,G	**Low** - (Clad with Alloy 52 during initial fabrication)
Primary Vent Nozzles (4) 2 per SG	Steam Generator	A82	0.75"	0.154"	540 °F & 584°F	B,E,G	**Moderate** - (Low Temperature, Low Probability)
RV Clevis Insert Lock Keys	Reactor Vessel	A600	N/A	N/A	540°F	G	**Low** - (No Industry OE of Failure, Low Consequence, Not Pressure Boundary)
RV Clevis Inserts	Reactor Vessel Internals	A600	N/A	N/A	540°F	G	**Low** -(No Industry OE of Failure, Low Consequence, Not Pressure Boundary)

B – Causes a design-basis accident

E – Breaches reactor coolant pressure boundary integrity

G – Causes a significant economic impact. Significant events are those for which we do not have a proven fix and would result in significant regulatory and/or public scrutiny, such as first-of-a-kind consideration would be a suitable test. It can be considered that "non-significant" events are those for which it is expected that a proven fix exists that will require minimal regulatory and/or public scrutiny.

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	Inspections [1, 2]							
Weld number	**Most Recent BMV Exam**	**BMV Results**	**Current BMV Frequency**	**Next Scheduled BMV**	**As-Built Geometry Acquired**	**PWSCC Category**	**Volumetric**	**Inspection Comments**
Unit 1								
Vessel Head Penetrations (38)	Spring 2007	NRI [3]	Each RFO	Fall 2008	Yes	A [4, 6]	Fall 2011	Per NRC Order 03-009. BMV per PBNP Letter NRC 2002-0082.
BMI Nozzles (36)	Spring 2007	NRI [3]	Each RFO	Fall 2008	N/A	N/A	N/A	No commitment at this time for UT. BMV per PBNP Letter NRC 2003-0089.
SG Channel Head Drains	2005 [5]	NRI [3]	Every other RFO	Fall 2008	N/A	N/A	N/A	OE and Code Case N-722
SG Channel Head Divider Plate	None	N/A	None Required	Fall 2008	N/A	N/A	N/A	Visual per CIM #104 with Eddy Current (ET) of Steam Generator
RV Clevis Insert Lock Keys	To Follow EPRI Materials Reliability Program (MRP) Reactor Vessel Internals ITG Program Requirements							
RV Clevis Inserts	To Follow EPRI Materials Reliability Program (MRP) Reactor Vessel Internals ITG Program Requirements							
SG Nozzle Dam Rings	None	N/A	None Required	Fall 2008	N/A	N/A	N/A	Visual per CIM #104 with Eddy Current (ET) of Steam Generator

NOTES:
1. 10CFR50.55a mandates that PDI techniques are used. For those welds with single-sided access, we can take credit for only that side (50%), even though we may be able to penetrate the weld and see some of the other side.
2. Our Risk-Informed ISI Program does not require every DM weld to be examined, however, due to NEI 03-08 guidance and MRP-139, DM Welds may put them back into the schedule. Thickness for DM safe-end welds must be determined by actual measurement, these have all been performed.
3. NRI – No recordable indications
4. PWSCC Category A – Resistant Materials
5. Welds were repaired/replaced during 2005 outage (U1R29). Prior to welding, examinations were performed on the original welds to assure no flaws existed.
6. Alloy 690/52/152

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	Inspections [1, 2]							
Weld number	**Most Recent BMV Exam**	**BMV Results**	**Current BMV Frequency**	**Next Scheduled BMV**	**As-Built Geometry Acquired**	**PWSCC Category**	**Volumetric**	**Inspection Comments**
Unit 2								
Vessel Head Penetrations (38)	Spring 2008	NRI [3]	Each RFO	Fall 2009	Yes	A [4, 5]	Spring 2011	Per NRC Order 03-009. BMV per PBNP Letter NRC 2002-0082.
BMI Nozzles (36)	Spring 2008	NRI [3]	Each RFO	Fall 2009	N/A	N/A	N/A	No commitment at this time for UT. BMV per PBNP Letter NRC 20003-0089.
RC-34-MRCL-AI-05	4/20/2005	NRI [3]	None	N/A	Yes	A [4]	Fall 2009	No App VIII UT Exam to date – Inspection required per ISI Program
RC-36-MRCL-AII-01A	4/20/2005	NRI [3]	None	N/A	Yes	A [4]	Fall 2009	Same as above
RC-34-MRCL-BI-05	4/20/2005	NRI [3]	None	N/A	Yes	A [4]	Fall 2009	Same as above
RC-36-MRCL-BII-01A	4/20/2005	NRI [3]	None	N/A	Yes	A [4]	Fall 2009	Same as above
Primary Vent Nozzles (4) 2 per SG	N/A	N/A	Each RFO [6]	Fall 2009	N/A	N/A	N/A	Code Case N-722
RV Clevis Insert Lock Keys	To Follow EPRI Materials Reliability Program (MRP) Reactor Vessel Internals ITG Program Requirements							
RV Clevis Inserts	To Follow EPRI Materials Reliability Program (MRP) Reactor Vessel Internals ITG Program Requirements							

NOTES:
1. 10CFR50.55a mandates that PDI techniques are used. For those welds with single-sided access, we can take credit for only that side (50%), even though we may be able to penetrate the weld and see some of the other side.
2. Our Risk-Informed ISI Program does not require every DM weld to be examined, however, due to NEI 03-08 guidance and MRP-139, DM Welds may put them back into the schedule. Thickness for DM safe-end welds must be determined by actual measurement, these have all been performed.
3. NRI – No recordable indications
4. PWSCC Category A – Resistant Materials
5. Alloy 690/52/152
6. For vent nozzles at hot leg temperature. Otherwise, once per ISI interval

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Weld number	Mitigation Options	Repair Options	Plan Summary
Unit 1			
Vessel Head Penetrations (38)	....	....	Head replaced in Fall 2005 with resistant material
BMI Nozzles (36)	....	Preventative or Repair via a Half Nozzle Repair	Follow future guidance from the EPRI MRP on inspections of BMIs.
SG Channel Head Drains	....	Preventative or Repair via a Half Nozzle Repair	Prepare half nozzle repair package materials. Perform preventive repair at earliest opportunity consistent with high susceptibility.
SG Channel Head Divider Plate		Grind indications smooth. Re-inspect with PT.	Visual examination concurrent with planned ECT inspections. Follow future guidance from EPRI regarding inspection techniques, acceptance criteria and repair methods.
RV Clevis Insert Lock Keys	....	....	Follow PWR Owners Group recommendations on repair strategies. Follow EPRI RVI-ITG recommendations on inspections.
RV Clevis Inserts	....	....	Follow PWR Owners Group recommendations on repair strategies. Follow EPRI RVI-ITG recommendations on inspections.
SG Nozzle Dam Rings	....	PWR OG Developing Repair Options	Follow PWR Owners Group recommendations on possible replacement

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Weld number	Mitigation Options	Repair Options	Plan Summary
Unit 2			
Vessel Head Penetrations (38)	....	....	Head replaced in Spring 2005 with resistant material
BMI Nozzles (36)	....	Preventative or Repair via a Half Nozzle Repair	Follow future guidance from the EPRI MRP on inspections of BMIs.
RC-34-MRCL-AI-05	Mitigated during fabrication with inlay of A52/152 prior to installation.	None	Perform UT during U2R30. PWSCC Category A (Resistant Materials) - No follow-up inspections needed
RC-36-MRCL-AII-01A	Mitigated during fabrication with inlay of A52/152 prior to installation.	None	Perform UT during U2R30. PWSCC Category A (Resistant Materials) - No follow-up inspections needed
RC-34-MRCL-BI-05	Mitigated during fabrication with inlay of A52/152 prior to installation.	None	Perform UT during U2R30. PWSCC Category A (Resistant Materials) - No follow-up inspections needed
RC-36-MRCL-BII-01A	Mitigated during fabrication with inlay of A52/152 prior to installation.	None	Perform UT during U2R30. PWSCC Category A (Resistant Materials) - No follow-up inspections needed
Primary Vent Nozzles (4) 2 per SG	....	Preventative or Repair via a Half Nozzle Repair	Prepare half nozzle repair package materials. Perform preventive repair at earliest opportunity consistent with moderate susceptibility.
RV Clevis Insert Lock Keys	....	....	Follow PWR Owners Group recommendations on repair strategies. Follow EPRI RVI-ITG recommendations on inspections.
RV Clevis Inserts	....	....	Follow PWR Owners Group recommendations on repair strategies. Follow EPRI RVI-ITG recommendations on inspections.