NMC
Committed to Nuclear Excellence

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U.S. Nuclear Regulatory Commission
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POINT BEACH NUCLEAR PLANT
DOCKETS 50-266 AND 50-301
BULLETIN 2002-01, "REACTOR PRESSURE VESSEL HEAD DEGRADATION AND REACTOR COOLANT PRESSURE BOUNDARY INTEGRITY," 60-DAY RESPONSE FOR THE POINT BEACH NUCLEAR PLANT, REQUEST FOR ADDITIONAL INFORMATION (TAC NOs. MB4566 and MB4567)

On March 18, 2002, the Nuclear Regulatory Commission (NRC) transmitted Bulletin (BL) 2002-01, "Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity." On May 16, 2002, Nuclear Management Company, LLC (NMC) provided the 60-day response to BL 2002-01. On November 18, 2002, the NRC issued a request for additional information (RAI) concerning the 60-day response to BL 2002-01. The NRC requested that the response be provided within 60 days of receipt of the RAI. NMC is providing the attached RAI response for the Point Beach Nuclear Plant. Attachment 1 provides the responses to the RAI questions. Attachment 2 provides the table as requested in the RAI.

This letter contains no new commitments and no revisions to existing commitments:

I declare under penalty of perjury that the foregoing is true and accurate. Executed on January 20, 2003.

A. J. Cayia

A. J. Cayia
Site Vice President, Point Beach Nuclear Plant

CC Regional Administrator, USNRC, Region III
Project Manager, Point Beach Plant, USNRC, NRR
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Attachments

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ATTACHMENT 1

NUCLEAR MANAGEMENT COMPANY, LLC
POINT BEACH NUCLEAR PLANT
DOCKETS 50-266 AND 50-301

JANUARY 20, 2003

BULLETIN 2002-01, "REACTOR PRESSURE VESSEL HEAD DEGRADATION AND REACTOR COOLANT PRESSURE BOUNDARY INTEGRITY," 60-DAY RESPONSE FOR THE POINT BEACH NUCLEAR PLANT, REQUEST FOR ADDITIONAL INFORMATION

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BULLETIN 2002-01, "REACTOR PRESSURE VESSEL HEAD DEGRADATION AND REACTOR COOLANT PRESSURE BOUNDARY INTEGRITY," 60 DAY RESPONSE FOR THE POINT BEACH NUCLEAR PLANT, REQUEST FOR ADDITIONAL INFORMATION

Requested Item

1. *Provide detailed information on, and the technical basis for, the inspection techniques, scope, extent of coverage, and frequency of inspections, personnel qualifications, and degree of insulation removal for examination of Alloy 600 pressure boundary material and dissimilar metal Alloy 82/182 welds and connection in the reactor coolant pressure boundary (RCPB). Include specific discussion of inspection of locations where reactor coolant leaks have the potential to come in contact with and degrade the subject material (e.g., reactor pressure vessel (RPV) bottom head).*

Response

The technical basis for, the inspection techniques, scope and extent of coverage, frequency of inspections, personnel qualifications and degree of insulation removal are as required to fulfill the requirements of American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code, Section XI, 1998 Edition, 2000 Addenda as modified by Nuclear Regulatory Commission (NRC) commitments. Refer to Attachment 2, Table 1 for a summary of these details.

Point Beach Nuclear Plant (PBNP) Unit 1 has 85 Alloy 600 (Inconel) penetrations and PBNP Unit 2 has 86 Alloy 600 penetrations and four Alloy 82/182 welds in the reactor coolant system (RCS). The RCS includes two identical heat transfer loops connected in parallel to the RPV. Each loop contains one steam generator, one circulating pump, flow and temperature instrumentation and connecting piping. A pressurizer is connected to one of the RPV outlets by means of a surge line.

Attachment 2, Table 1 contains a description of the Alloy 600 penetrations contained within the RCS and the inspection requirements for the penetrations.

The following sections describe the Alloy 600 penetration configurations, including discussion of inspection of locations where reactor coolant leaks have had the potential to come into contact with the RPV bottom head, and when applicable, repairs and augmented examination requirements.

Reactor Pressure Vessel Penetrations

The Unit 1 RPV shell was fabricated in three sections, the upper shell, the intermediate shell and the lower shell. The upper shell course is a machined forging fabricated of A-508 manganese-molybdenum steel conforming to ASME code case 1332-2 and is clad internally with weld deposited stainless steel. The lower and intermediate shells are cylindrical formed weldments fabricated of SA-302, Grade B, manganese-molybdenum steel and is clad internally with weld deposited austenitic stainless steel. The lower section of the lower shell course is weld deposited (clad) with Alloy 600.

The Unit 2 RPV shell is a machined forging fabricated of A-508 manganese-molybdenum steel conforming to ASME code case 1332-2 and is clad internally with weld deposited stainless steel.

The shell course contains the four primary nozzles and the two safety injection nozzles with safe-ends. All six nozzles are forgings fabricated of A-508 manganese-molybdenum steel conforming to ASME code case 1332-2 and clad with austenitic stainless steel internally. The safety injection safe-ends are SA-182F-316 stainless steel.

The RPV shell has no Alloy 600 penetrations.

The RPV is contained within the required inspection boundary for the ASME B&PV Code, Section XI, "System Leakage Test," required each refueling outage. This is a visual inspection performed without the removal of insulation.

Reactor Pressure Vessel Head Penetrations

The RPV heads contains 85 Alloy 600 penetrations for Unit 1 and 86 Alloy 600 penetrations for Unit 2, which are categorized as follows:

- Forty-nine upper vessel head penetrations that are Alloy 600 (ASME SB-167), for both Unit 1 and Unit 2 upper heads. The housing bodies are welded into the inside of the closure head with weld deposited Alloy 600. The upper heads are fabricated of ASME SA-302 Grade B steel and internally clad with weld deposited austenitic stainless steel.

- One Unit 2 vent line which contains a stub welded to the upper head that is Alloy 600 (ASME SB-166-63).

- Thirty-six RPV lower head instrumentation penetrations which are Alloy 600 for both Unit 1 and Unit 2. The lower heads are fabricated of ASME SA-302 Grade B steel and internally clad with weld deposited austenitic stainless steel.

In response to NRC Bulletin 2001-01, "Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles," 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. This required removal of asbestos insulation that was directly in contact with the RPV upper head. A new insulation package was then installed which is elevated above the upper RPV head for future access.

In response to NRC Bulletin 2002-02, "Reactor Pressure Vessel Head and Vessel Head Penetration Nozzle Inspection," 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 was performed for Unit 1 during the Fall 2002 refueling outage. All examinations to date have been acceptable without repair. An above head visual examination and the VHP ultrasonic examinations are scheduled for the Unit 2 upper RPV head during the Unit 2 Fall 2003 refueling outage.

The RPV heads are contained within the required inspection boundary for the ASME B&PV Code, Section XI, "System Leakage Test," required each refueling outage. Six inspection ports in the RPV upper head insulation are removed for a limited direct visual of portions of the control rod drive mechanisms and upper head.

Pressurizer

The pressurizer maintains RCS operating pressure and compensates for changes in coolant volume during load changes. The pressurizer is constructed of ASME SA-302, Grade B. The heads are ASME SA-216 WCC. The interior surface is clad with type 304 stainless steel.

The pressurizer has no Alloy 600 penetrations.

The pressurizer is contained within the required inspection boundary for the ASME B&PV Code, Section XI, "System Leakage Test," required each refueling outage. This is a visual inspection performed with the removal of insulation for the bolted pressurizer manway cover.

Steam Generators

Each loop contains a vertical shell and U-tube steam generator (SG). Reactor coolant enters the inlet side of the channel head at the bottom of the SG through the inlet nozzle, flows through the U-tubes and leaves through an outlet channel and leaves the SG through another bottom nozzle. The inlet and outlet channels are separated by a partition. The SG is constructed primarily of carbon steel. The heat transfer tubes are Alloy 600. The interior surfaces of the channel heads and nozzle are clad with austenitic stainless steel and the side of the tube sheet in contact with the reactor coolant is clad with Ni-Cr-Fe Alloy. The tube sheet joint is welded. The Unit 1 SGs were replaced during an outage that occurred between October 1983 and April 1984 (there are no Alloy 600 penetrations in the Unit 1 SGs). The Unit 2 SGs were replaced during an outage that occurred between October 1996 and August 1997.

Each Unit 2 SG contains two Alloy 82/182 welds, which are categorized as follows:

- Four primary nozzle to safe-end welds on each hot and cold leg on each SG.

Each Unit 2 SG contains two Alloy 600 penetrations, which are categorized as follows:

- Four primary vent nozzles (ASME SB-167) on each cold and hot leg side of each SG channel head.

The Alloy 82/182 welds on the Unit 2 SG receive a volumetric and surface exam as required by ASME B&PV Code, Section XI. The baseline exams were completed in May 1996 and the inlet nozzles were examined in accordance with the Inservice Inspection Long Term Plan in December 1998 and January 1999.

The primary side of the SGs is contained within the required inspection boundary for the ASME B&PV Code, Section XI, "System Leakage Test," required each refueling outage. This is a visual inspection performed with the removal of insulation for the bolted steam generator manway covers.

Reactor Coolant Pumps

The reactor coolant is circulated by four pumps, which are of the vertical single suction, centrifugal type. The pumps are constructed of high alloy casting (ASTM A 351, GR CF8M). All parts of the pumps in contact with the reactor coolant are austenitic stainless steel or equivalent corrosion resistant materials.

The reactor coolant pumps have no Alloy 600 penetrations.

The reactor coolant pumps are contained within the required inspection boundary for the ASME B&PV Code, Section XI, "System Leakage Test," required each refueling outage. This is a visual inspection performed without the removal of insulation.

Reactor Coolant Piping

The austenitic stainless steel reactor coolant piping and fittings, which make up the loops are a 29-inch ID in the hot legs, a 27.5-inch ID in the cold legs, and 31-inch ID between each steam generator outlet and its reactor coolant pump suction. Smaller piping, including the pressurizer surge spray and relief lines, drains, and connections to other systems are austenitic stainless steel.

The reactor coolant piping has no Alloy 600 penetrations.
The reactor coolant piping is contained within the required inspection boundary for the ASME B&PV Code, Section XI, "System Leakage Test," required each refueling outage. This is a visual inspection performed without the removal of insulation.

The following section provides a summary of personnel qualifications for examinations of Alloy 600 pressure boundary material and dissimilar metal weld connections. Personnel qualifications per component can be found in Attachment 2, Table 1.

Personnel Qualifications

Point Beach Nuclear Power Plant Personnel Qualifications For Examination of Alloy 600 Pressure Boundary Material And Dissimilar Metal Weld Connections

Examination Method	Personnel Qualification Requirements
Ultrasonic (UT)	Personnel are presently qualified in accordance with their employers written practice. In accordance with 10CFR50.55a, personnel performance demonstrations will be through Electric Power Research Institute / Performance Demonstration Initiative. Qualifications will be per the requirements of ASME B&PV Code, Section XI, 1998 Edition, with 2000 Addenda of Section XI, Appendix VIII, Supplement 10 for dissimilar metal welds.
Liquid Penetrant (PT)	Personnel are qualified in accordance with their employers written practice, which meets the requirements of ANSI/ASNT CP-189 and ASME Section XI, IWA-2300, 1998 Edition, 2000 Addenda.
Visual (VT-1, VT-2, VT-3)	Personnel are qualified in accordance with their employers written practice, which meets the requirements of ANSI/ASNT CP-189 and ASME Section XI, IWA-2300, 1998 Edition, 2000 Addenda.

Requested Item

2. *Provide the technical basis for determining whether or not insulation is removed to examine all locations where conditions exist that could cause high concentrations of boric acid on pressure boundary surfaces or locations that are susceptible to primary water stress corrosion cracking (Alloy 600 base metal and dissimilar metal Alloy 82/182 welds). Identify the type of insulation for each component examined, as well as any limitations to removal of insulation. Also, include in your response actions involving removal of insulation required by your procedures to identify the source of leakage when relevant conditions (e.g., rust stains, boric acid stains, or boric acid deposits) are found.*

Response

The technical basis for determining whether or not insulation is removed to examine all locations where conditions exist that could cause high concentrations of boric acid on pressure boundary surfaces or locations that are susceptible to primary water stress corrosion cracking (PWSCC), is provided in ASME B&PV Code, Section XI, 1998 Edition, 2000 Addenda. IWA-5242 requires that systems borated for the purpose of controlling reactivity shall have insulation removed from pressure retaining bolted connections in order to complete visual examination, VT-2. Nuclear Management Company, LLC (NMC) considers portions of the PBNP emergency core cooling system (ECCS), chemical and volume control system (CVCS), the RCS and sample system (SC) as systems borated for the purpose of controlling reactivity. These VT-2 examinations are performed at the frequency specified in ASME B&PV Code, Section XI for system pressure tests. During regularly scheduled inservice inspection activities, insulation is removed as necessary to complete the specified inspection or examination technique.

Indications of leakage that occur on insulated components require removal of insulation to determine the source of the leakage for evaluation. Any active leak requires repair, replacement or evaluation in accordance with IWA-5250.

In accordance with the corrective action process at PBNP, an action request (AR) shall be initiated upon discovery of equipment malfunction, damage, or degradation that is considered sudden or unexpected. Per plant procedures, the following indications related to boric acid accumulations shall be documented by initiation of an AR:

a. Through-wall leakages identified from cracks or weld defects.
b. Any leakage from recirculation heat removal systems (high pressure safety injection (SI), low pressure safety injection (RHR) and containment spray system (CSS) outside of containment including leaks from seats,

seals, valve stems, pump seals, vessel flange gaskets, and other mechanical joints which result in total leakage greater than 400 ml/min.

c. Degradation of fastener material, which may reduce cross sectional area greater than or equal to five percent.
d. Degradation of pressure boundaries, which may reduce wall thickness greater than or equal to 10 percent.
e. Other conditions adverse to quality not specifically described in the procedure.

Examinations with recorded indications require an evaluation including an assessment of condition for areas contacted by boric acid. This evaluation includes the items summarized in the following statements:

a. Describe boric acid accumulation location(s).
b. Describe boric acid leak path(s).
c. Is degradation OR corrosion evident?
d. Is boric acid leak active (wet leakage) or inactive (minor dry residue)?
e. Has leakage contacted other components?
f. Is the source of boric acid accumulation due to a component failure other than packing, flange, OR threaded connection leaks (i.e., cracked fittings, welds, components)?
g. Describe recommended actions.

Based upon this evaluation, corrective actions are planned and implemented to address all equipment affected by boric acid, including removal of insulation and inspection of potentially affected carbon steel surfaces.

Insulation removal limitations are unique for each type of location and are dependent on the elevation of the location above floor level and nearness of the location to radiation sources, such as, the RCS. These limitations are considered when planning examinations for specific locations. Due to the nearness of each of the Alloy 600 locations to the RCS, radiation dose is of considerable concern.

The type of insulation for each component is provided in Attachment 2, Table 1.

Requested Item

3. *Describe the technical basis for the extent and frequency of walkdowns and the method for evaluating the potential for leakage in inaccessible areas. In addition, describe the degree of inaccessibility, and identify any leakage detection systems that are being used to detect potential leakage from components in inaccessible areas.*

Response

The technical basis for the extent and frequency of walkdowns and method for evaluating the potential for leakage is provided in ASME B&PV Code, Section XI. A leak test is performed on the primary system each refueling outage, as required by ASME B&PV Code, Section XI, Category B-P. This leak test is performed after reaching normal operating pressure during startup from the refueling outage. Additionally, the same procedure is performed at the beginning of the outage to provide an as found condition and provide time within the refueling outage for corrective action. NMC does not exclude any component from the leak test due to inaccessibility and meets ASME B&PV Code, Section XI requirements for insulated components.

The leak test procedure includes entry into the refueling cavity for a VT-2 examination of the primary pressure boundary for the upper area of the RPV, control rod drive mechanisms, head vent system and conoseals. Six inspection ports in the RPV upper head insulation are removed for a limited direct visual of portions of the control rod drive mechanisms and upper head. The leak test also includes entry into the instrumentation tunnel for VT-2 examination of the RPV lower head and instrumentation penetrations. The mirror insulation on the lower head is not removable as insulation was installed prior to welding the thimble tube conduits to the RPV nozzles.

The RPV hot and cold leg nozzles and the safety injection nozzles attached to the vertical shell of the RPV are below the cavity and are not readily accessible for direct visual examination. Limited access is available through the sand box covers, which are removed at a 10-year periodicity for surface examination of the nozzle welds. This surface examination was completed most recently for two nozzles on each unit in October 1999 for Unit 1 and in November 2000 for Unit 2 with acceptable results. The vertical surfaces of the insulated RPV are examined from the instrumentation tunnel in accordance with ASME B&PV Code, Section XI, IWA-5242.

All areas where leakage is likely to occur as listed in Attachment 2, Table 1 are accessible during the RCS leak test.

In accordance with technical specification surveillance requirement, the RCS is monitored by the daily determination of RCS leakage per directions contained in a system operating procedure. This procedure uses indicators, such as, charging versus letdown and containment sump level change to monitor for RCS leakage. Containment airborne activity and containment sump level are monitored to detect leakage. These methods would help identify reactor coolant leakage in an inaccessible area.

Requested Item

4. *Describe the evaluations that would be conducted upon discovery of leakage from mechanical joints (e.g., bolted connections) to demonstrate that continued operation with the observed leakage is acceptable. Also describe the acceptance criteria that was established to make such a determination. Provide the technical basis used to establish the acceptance criteria. In addition,*

    a. *if observed leakage is determined to be acceptable for continued operation, describe what inspection/ monitoring actions are taken to trend/evaluate changes in leakage, or*
    b. *if observed leakage is not determined to be acceptable, describe what corrective actions are taken to address the leakage.*

Response

Leakage from safety related mechanical joints is documented in the PBNP corrective action program. Upon discovery of active leakage, a corrective action document is initiated and an operability determination is completed. The first objective is to repair the leaking component connection. In accordance with ASME B&PV Code, Section XI, subparagraph IWA-5250(a)(2), bolting is removed and a VT-3 visual exam for corrosion shall be performed and evaluated in accordance with ASME B&PV Code, Section XI, paragraph IWA-3100. As an alternative to the requirements of subparagraph IWA-5250(a)(2), an evaluation can be performed to assess the integrity of the joint per code case N-566-1, "Corrective Action for Leakage Identified at Bolted Connections, Section XI, Division 1," as approved for use by the NRC for PBNP. Any repair or replacement of a component shall satisfy ASME B&PV Code, Section XI, Article IWA-4000, as applicable.

Code case N-566-1 requires one of the following actions as an alterative to the requirements of IWA-5250(a)(2) when leakage is detected at a bolted connection:

(a) The leakage shall be stopped, and the bolting and component material shall be evaluated for joint integrity as described in (c) below.

(b) If the leakage is not stopped, the joint shall be evaluated in accordance with IWB-3142.4 for joint integrity. This evaluation shall include the considerations listed in (c) below.

(c) The evaluation for (a) and (b) above is to determine the susceptibility of the bolting to corrosion and failure. This evaluation shall include the following:

(1) the number and service age of bolts;
(2) bolt and component material;
(3) corrosiveness of process fluid;
(4) leakage location and system function;
(5) leakage history at the connection or other system components;
(6) visual evidence of corrosion at the assembled connection.

No immediate action is necessary when the evaluation required by code case N-566-1 determines that the leaking condition has not degraded the fasteners or the connection, or that the joint integrity will remain acceptable until corrective action for the leak is completed. However, reasonable attempts shall be made to stop the leakage as appropriate.

Acceptance criteria used during the IWB-3142.4 evaluation will generally be in accordance with ASME Section III for allowable stresses. In special cases, alternative acceptance criteria may be developed and included in the IWB-3142.4 evaluation and submitted to the NRC in accordance with IWB-3144(b).

If the evaluation of the variables above indicates the joint integrity is indeterminate or that there is a need for further evaluation, the following actions shall be taken:

1. The bolt closest to the source of leakage and any bolts that have been degraded due to the leakage shall be removed;

2. The bolt(s) shall receive a visual VT-1 examination;

3. The visual VT-1 examination results shall be evaluated and dispositioned in accordance with IWB-3517. If the removed bolting shows evidence of rejectable degradation, the bolts adjacent to the removed bolting shall be removed and VT-1 examined and evaluated per IWB-3517.

The bolting removal and visual VT-1 examination(s) may be deferred to the next outage of sufficient duration if justified. If continued operation cannot be justified, then the component is repaired prior to resumption of service.

Leakage of the PBNP RCS pressure boundary to date has not required application of code case N-566-1.

Requested Item

5. *Explain the capabilities of your program to detect the low levels of reactor coolant pressure boundary leakage that may result from through-wall cracking in the bottom reactor pressure vessel head incore instrumentation nozzles. Low levels of leakage may call into question reliance on visual detection techniques or installed leakage detection instrumentation, but has the potential for causing boric acid corrosion. The NRC has had a concern with bottom reactor pressure vessel head incore instrumentation nozzles because of the high consequences associated with loss of integrity of the bottom head nozzles. Describe how your program would evaluate evidence of possible leakage in this instance. In addition, explain how your program addresses leakage that may impact components that are in the leak path.*

Response

Detection of low levels of RCPB leakage that may result from through-wall cracking in the bottom reactor pressure vessel head incore instrumentation nozzles is provided by the automatic systems that monitor the containment atmosphere radioactivity and the containment sump level. These leakage detection systems are specified in plant Technical Specifications.

Small leaks from the RCS can be detected by one or a combination of the following systems:

a) Containment Atmosphere Relative Humidity - One containment humidity detector is available to detect a small leak in containment. Humidity readings are taken each shift. This provides a plot of containment air dewpoint variations above a baseline maximum established by cooling water temperature to the air coolers. This is sensitive to incremental leakage equivalent to 2 to 10 gpm.

b) Containment Sump Level - Containment sump water level indication is provided in the main control room by one level indicator that can be used to detect RCS leakage. Sump level is recorded each shift. The sump is drained when the high level alarm is reached. The times at which the sump is drained are recorded in station logs. From this information a volume and leak rate can be determined. This method is sensitive to incremental leakage equivalent to 0.5 to 10 gpm.

c) Containment Sample and Continuous Vent - Air from containment is drawn through a particulate and a noble gas monitor and is pump backed into containment. In addition, a path for the discharge of containment air to the atmosphere through the containment purge exhaust stack filters is open. The particulate monitor is capable of detecting particulate activity in concentrations as low as 10^{-8} µCi/cc with a range of 10^{-8} to 10^{-3} µCi/cc. The noble gas monitor will sense gaseous activity in the range of 10^{-7}µCi/cc to 10^{-1} µCi/cc. This method is sensitive to 0.013 gpm within 20 minutes.

One charging pump is in automatic under normal operation and will automatically increase in speed to try to maintain the equivalence between letdown flow and combined charging line flow across the reactor coolant pump seals. If the pump reaches a high speed limit, an alarm is actuated.

A review of the PBNP leak rate capabilities has also been performed relative to this issue. NMC routinely checks the leakage of the primary system daily via plant surveillance procedure when the reactor is at power or in the hot shutdown condition. Typical leakage at PBNP during an operating cycle is less than 0.1 gpm. An increase to 0.2 gpm would signify a possible problem and result in increased monitoring and inspections for RCS leakage.

Through wall cracking of the RPV lower head incore instrumentation nozzles would be detected by the visual observation of leakage through the mirror insulation. The insulation is examined as evidence of leakage would be visible with boric acid deposits or wetted areas. An as-found condition is documented prior to refueling activities that shows where potential leakage occurred during power operation since the previous outage. Boric acid deposits that occur as a result of leaking cavity seals during refueling are cleaned prior to the RCS pressure test in Mode 5 during startup to eliminate boric acid deposits that could mask new leakage. Both the as-found and ASME pressure test are performed with the same procedure. Additionally, a remote VT-3 with a remotely operated vehicle is performed on the lower head nozzles every 10 years from the inside the RPV.

Evaluation of boric acid indications are performed by experienced engineers. Evidence of leakage is compared against previous examination results. Component materials, estimated length of time of leak, operating temperature of component, leak path, and existing indications of potential corrosion products are considered in the evaluation.

NMC performs routine inspections of the containment during power operation. Detailed visual inspections are performed of the reactor coolant system each refueling outage prior to returning the unit to power.

The combination of activities performed at PBNP including daily leak rate measurements, as found and post refueling visual examinations, and routine containment entries provide assurance that any potential degradation will be detected and corrected by the plant staff.

Once the source of leakage is discovered, it is documented in accordance with the site's corrective action program. Part of the evaluation of cause and determination of corrective actions includes a determination of affect on other plant structures, systems and components (SSCs) in the leak path. Once a determination is made whether any SSCs are affected, the SSCs are assessed for damage and necessary corrective actions.

NDE techniques are described in the response to question 1 and included in Attachment 2, Table 1.

Requested Item

6. *Explain the capabilities of your program to detect the low levels of reactor coolant pressure boundary leakage that may result from through-wall cracking in certain components and configurations for other small diameter nozzles. Low levels of leakage may call into question reliance on visual detection techniques or installed leakage detection instrumentation, but has the potential for causing boric acid corrosion. Describe how your program would evaluate evidence of possible leakage in this instance. In addition, explain how your program addresses leakage that may impact components that are in the leak path.*

Response

Detection of low levels of RCPB leakage that may result from through-wall cracking in certain components and configurations for other small nozzles is provided by the automatic systems that monitor the containment atmosphere radioactivity and the containment sump level. These methods are the same as answered above in response to Question 5.

Requested Item

7. *Explain how any aspects of your program (e.g., insulation removal, inaccessible areas, low levels of leakage, evaluation of relevant conditions) make use of susceptibility models or consequence models.*

Response

The PBNP boric acid corrosion control (BACC) program does not use susceptibility models or consequence models. The BACC program in use at PBNP is per the response to NRC Generic Letter 88-05, "Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components in PWR Plants."

Requested Item

8. *Provide a summary of recommendations made by your reactor vendor on visual inspections of nozzles with Alloy 600/82/182 material, actions you have taken or plan to take regarding vendor recommendations, and the basis for any recommendations that are not followed.*

Response

PBNP Unit 1 RPV was fabricated by Babcock and Wilcox and the Unit 2 RPV was fabricated by Combustion Engineering. Westinghouse has made no recommendations for visual inspections on nozzles with Alloy 600/82/182 material.

NMC implements an Operating Experience Program to ensure that lessons learned from industry experience are translated into appropriate action to improve plant safety, reliability and availability. Personnel screen and review industry experience documents for applicability to plant activities or programs, including boric acid corrosion control. As potential relevant reports or documents are identified, the sites corrective action program is used as a vehicle for completing assessments and implementing changes based on the particular industry experience.

Requested Item

9. *Provide the basis for concluding that the inspections and evaluations described in your responses to the above questions comply with your plant Technical Specifications and Title 10 of the Code of Federal Regulations (10 CFR), Section 50.55(a), which incorporates Section XI of the American Society of Mechanical Engineers (ASME) Code by reference. Specifically, address how your boric acid corrosion control program complies with ASME Section XI; paragraph IWA-5250 (b) on corrective actions. Include a description of the procedures used to implement the corrective actions.*

Response

NMC has concluded that the inspections and evaluations described above comply with ASME B&PV Code, Section XI, paragraph IWA-5250 (b), (as invoked by 10CFR 50.55(a)). This conclusion is based on IWA-5250 being invoked as applicable in the RCS Pressure Test procedure, specifically in relation to system leakage and hydrostatic pressure testing.

NMC has concluded that the inspections and evaluations described above comply with ASME B&PV Code, Section XI. 10CFR50.55a require compliance with ASME B&PV Code Section XI. PBNP TS do not invoke any additional requirements or programs to control RCPB corrosion caused by leakage of boric acid, thus, compliance with ASME B&PV Code, Section XI is all that is required. That is, there is not a TS RCPB BACC program separate from the Inservice Inspection Program.

The PBNP BACC program, particularly with respect to RCPB components other than the RPV head, is based on Generic Letter 88-05 and the subject of a controlled plant procedure. The requirements of this procedure are implemented through the site's surveillance, corrective action, and work control processes. The requirements of the BACC Program are implemented through the following procedures:

- ISI CL 1, 2, 3, Program, "PBNP Class 1, 2, and 3 Inservice Inspection Program, This procedure describes the Inservice Inspection Program and provides the requirements for a BACC Program.

- 1-PT-RCS-1, Reactor Coolant System (RCS) Pressure Test – Inside / Outside Containment Unit 1, This procedure provides the reactor coolant leak test procedure and method of evaluation of indication for Unit 1.

- SEM 7.11.6, RCS Leak Test for Unit 2, This procedure provides the reactor coolant leak test procedure for Unit 1. This procedure is under revision to update standards established in 1-PT-RCS-1.

- SEM 7.11.2, Indication Disposition Report, This procedure provides the ASME B&PV Code, B&PV Code, Section XI disposition of indications including supplemental and augmented examinations. Initiation of the correct action process occurs within this procedure.

- NP 7.4.10, Periodic Pressure Test, This procedure provides the programmatic content of the ASME B&PV Code, Section XI Pressure Test Program.

- NDE 753, Visual Examination (VT-2) Leakage Detection of Nuclear Power Plants Components, Examination requirements for a VT-2 examination are provided within this procedure to ensure CP-189, 1994 standards are met.

- NDE 757, Visual Examination for Leakage of Reactor Vessel Closure Head Penetrations, This procedure provides requirements for examination of the reactor pressure vessel upper head meeting equivalent VT-2 requirements.

- NP 5.3.1, Action Request Process, This procedure provides description of the corrective action process.

ATTACHMENT 2

NUCLEAR MANAGEMENT COMPANY, LLC
POINT BEACH NUCLEAR PLANT
DOCKETS 50-266 AND 50-301

JANUARY 20, 2003

TABLE 1 – ALLOY 600 PENETRATIONS, ALLOY 82/182 WELDS, BOLTED CONNECTION INSPECTIONS AND CARBON STEEL PRESSURE VESSELS

4 Pages Follow

TABLE 1
ALLOY 600 PENETRATIONS,
ALLOY 82/182 WELDS,
BOLTED CONNECTION INSPECTIONS
AND CARBON STEEL PRESSURE VESSELS

Component	Inspection Techniques	Personnel Qualifications	Extent of Coverage	Frequency	Degree of Insulation Removal / Insulation Type	Corrective Action
Alloy 600 Penetrations and Alloy 82/182 Welds						
Vessel Head Penetrations (VHP)-49	Volumetric	UT	Interference Fit Region and Nozzle Base Metal	U1R27 (SEP 02) U2R26 (SEP 03)	Non insulated	SEM 7.11.2, Indication Disposition Report
	Visual	VT-2	100% Bare Metal	Each Refueling Outage	Removed / Panel Insulation	SEM 7.11.2
	Visual	VT-2	Insulated VT-2	Each Refueling Outage	Not Removed- 6 Inspection Ports Opened / Panel Insulation	PT-RCS-1, Reactor Coolant System (RCS) Pressure Test – Inside/Outside Containment Unit 1; SEM 7.11 2
Head Vent Line Stub (Unit 2 Only)	Visual	VT-3	100% Bare Metal	Each Refueling Outage	Removed / Panel Insulation	SEM 7.11.2
	Visual	VT-2	Insulated VT-2	Each Refueling Outage	Not Removed- 6 Inspection Ports Opened / Panel Insulation	PT-RCS-1, SEM 7.11 2
Instrument Tube Penetrations – 36	Visual	Remote VT-3	100% from RPV interior	Once Every 10 Years	Non insulated	SEM 7 11 2
	Visual	VT-2	Insulated VT-2	Each Refueling Outage	Not Removed / Mirror Insulation	PT-RCS-1, SEM 7.11.2
Unit 2 Primary Vent Nozzles (2 ea S/G)	Visual	VT-2	Insulated VT-2	Each Refueling Outage	Not Removed / Panel Insulation	PT-RCS-1, SEM 7.11.2

TABLE 1
ALLOY 600 PENETRATIONS,
ALLOY 82/182 WELDS,
BOLTED CONNECTION INSPECTIONS
AND CARBON STEEL PRESSURE VESSELS

Component	Inspection Techniques	Personnel Qualifications	Extent of Coverage	Frequency	Degree of Insulation Removal / Insulation Type	Corrective Action
Unit 2 Hot / Cold Leg S/G Primary Nozzles Safe-End Welds	Volumetric and surface	UT/PT	100%	Once Every 10 Years	Removed / Panel Insulation	SEM 7.11 2
	Visual	VT-2	Insulated VT-2	Each Refueling Outage	Not Removed / Panel Insulation	PT-RCS-1, SEM 7.11.2

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ALLOY 82/182 WELDS,
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Component	Inspection Techniques	Personnel Qualifications	Extent of Coverage	Frequency	Degree of Insulation Removal / Insulation Type	Corrective Action
Bolted Connection Inspection IWA-5242						
Pressurizer Manway	Visual	VT-2	100%	Each Refueling Outage	Removed / Blanket Insulation	PT-RCS-1, SEM 7.11.2
"A" Steam Generator Primary Manways (2)	Visual	VT-2	100%	Each Refueling Outage	Removed / Blanket Insulation	PT-RCS-1, SEM 7.11.2
"B" Steam Generator Primary Manways (2)	Visual	VT-2	100%	Each Refueling Outage	Removed / Blanket Insulation	PT-RCS-1, SEM 7.11.2
SI-867A, SI System to RC Loop A Cold Leg Check	Visual	VT-2	100%	Each Refueling Outage	Removed / Blanket Insulation	PT-RCS-1, SEM 7.11.2
SI-867B, SI System to RC Loop B Cold Leg Check	Visual	VT-2	100%	Each Refueling Outage	Removed / Blanket Insulation	PT-RCS-1, SEM 7.11 2

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ALLOY 600 PENETRATIONS,
ALLOY 82/182 WELDS,
BOLTED CONNECTION INSPECTIONS
AND CARBON STEEL PRESSURE VESSELS

Component	Inspection Techniques	Personnel Qualifications	Extent of Coverage	Frequency	Degree of Insulation Removal / Insulation Type	Corrective Action
Carbon Steel Pressure Vessels						
RPV (1 ea unit)	Visual	VT-2	Insulated VT-2	Each Refueling Outage	Not Removed / Mirror Insulation	PT-RCS-1, SEM 7.11.2
Pressurizer (1 ea unit)	Visual	VT-2	Insulated VT-2	Each Refueling Outage	Not Removed / Panel Insulation	PT-RCS-1, SEM 7.11.2
Steam Generators (2 each unit)	Visual	VT-2	Insulated VT-2	Each Refueling Outage	Not Removed / Panel Insulation	PT-RCS-1, SEM 7.11.2