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Committed to Nuclear Excellence

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L-PI-03-007
10 CFR 50.54(f)

January 20, 2003

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

PRAIRIE ISLAND NUCLEAR GENERATING PLANT
DOCKET NOS. 50-282 AND 50-306
LICENSE NOS. DPR-42 AND DPR-60
BULLETIN 2002-01, "REACTOR PRESSURE VESSEL HEAD DEGRADATION AND REACTOR COOLANT PRESSURE BOUNDARY INTEGRITY," 60-DAY RESPONSE FOR THE PRAIRIE ISLAND NUCLEAR GENERATING PLANT, REQUEST FOR ADDITIONAL INFORMATION
(TAC NOS. MB4568 AND MB4569)

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 17, 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 Prairie Island Nuclear Generating 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.

M K N

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Attachments

ATTACHMENT 1

NUCLEAR MANAGEMENT COMPANY, LLC
PRAIRIE ISLAND NUCLEAR GENERATING PLANT
DOCKET 50-282 AND 50-306

JANUARY 2003

BULLETIN 2002-01, "REACTOR PRESSURE VESSEL HEAD DEGRADATION AND REACTOR COOLANT PRESSURE BOUNDARY INTEGRITY," 60-DAY RESPONSE FOR THE PRAIRIE ISLAND 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 PRAIRIE ISLAND NUCLEAR GENERATING 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, extent of coverage, frequency of inspections, personnel qualifications and degree of insulation removal are per the requirements of American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code, Section XI, 1989 Edition as modified by Nuclear Regulatory Commission (NRC) commitments. Refer to Attachment 2, Table 1 for a summary of these details.

Prairie Island Nuclear Generating Plant has 77 Alloy 600 (Inconel) penetrations (with Alloy 82/182 welds), all of which are contained within the Reactor Coolant System (RCS) on the Reactor Vessel. The RCS includes two identical heat transfer loops connected in parallel to the reactor pressure vessel (RPV). Each loop contains one steam generator, one reactor coolant pump, flow and temperature instrumentation and connecting piping.

Table 1 contains a description of the 77 Alloy 600 penetrations contained within the RCS and the inspection requirements for the penetrations.

The following sections describe the 77 Alloy 600 penetration configurations, including discussion of inspection of locations where reactor coolant leaks have the potential to degrade other components, and, when applicable, repairs and augmented examinations requirements.

Reactor Pressure Vessel Penetrations and Reactor Pressure Vessel Head Penetrations

The reactor vessel Alloy 600 (SB 166) penetrations are categorized as follows:

- (29) - Control Rod Drive Mechanisms penetrations on the Reactor Vessel Closure Head.
- (4) - Spare penetrations previously used for part length rods (Head)

- (4) - Plugged spare penetrations (Head)
- (3) - Core Exit Thermocouple penetrations (Head)
- (1) - Vent Line on the Reactor Vessel Closure Head.
- (36) - Instrument Tube Penetrations on the Reactor Vessel Bottom Head.

A bare metal RPV head visual inspection has been performed at least once each refueling outage for the Prairie Islands units since 1997 as reported in the Prairie Island 15-day response to NRC Bulletin 2002-01, "Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity." The most recent bare metal examinations of both unit 1 and 2 were performed during the 2002 refueling outages by a VT-2 qualified examiner as committed to in Prairie Island response to Bulletin 2001-01. Examinations of both units were acceptable with no indication of leakage or corrosion of either head.

The RPV and RPV closure head 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. Prairie Island's response to NRC Bulletin 2001-01 included a commitment to perform a bare metal examination of the reactor vessel closure head.

Pressurizer

The pressurizer maintains RCS operating pressure and compensates for changes in coolant volume during load changes. The pressurizer shell is constructed of SA 533, Grade A, Class 1 material and the upper and lower heads are constructed of SA 216 Grade WCC material. The interior surface of the cylindrical shell and upper head is clad with type 304 stainless steel or equivalent.

The pressurizer has no Alloy 600 penetrations or Alloy 82/182 welds.

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

Steam Generators

Prairie Island Nuclear Plant has two steam generators (SGs) in each unit. Reactor coolant enters the SGs through the inlet nozzle, flows through the 0.875-inch OD U-tubes and leaves through the outlet nozzle. A vertical partition plate in the lower head separates the inlet and outlet plenums. The plenums are stainless steel clad, while the primary side of the tube sheet is Inconel clad. The U-tubes are SB-163 material. The tubes are rolled in the partial depth of the tube sheet and then the tube-to-tubesheet joint is welded on the primary side.

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

The steam generators have no Alloy 600 penetrations or Alloy 82/182 welds on the external pressure boundary.

Reactor Coolant Pumps

The reactor coolant is circulated by two pumps, which are of the vertical single suction, centrifugal type. The pumps are constructed of high alloy casting (ASTM A 351, GR CF8) and stainless steel parts to minimize corrosion.

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

The reactor coolant pumps have no Alloy 600 penetrations or Alloy 82/182 welds.

Reactor Coolant Piping

The reactor coolant piping consists of the following: a 29-inch ID hot leg pipe from the reactor vessel outlet to the SG inlet, a 31-inch ID cold leg pipe between the SG outlet and the pump suction nozzle, a 27-1/2 inch cold leg pipe between the pump discharge and the reactor vessel inlets, and a 14-inch, schedule 140 surge line pipe between the pressurizer and one hot leg. Unit 1 reactor coolant loop pipe material is seamless, forged, ASTM A376, Type 316. Unit 2 reactor coolant loop pipe material is centrifugally cast ASTM A351 CF8M material. Smaller piping, including the pressurizer spray and relief lines, drains, and connections to other systems are austenitic stainless steel.

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

The reactor coolant piping has no Alloy 600 penetrations or Alloy 82/182 welds.

The following section provides a summary of personnel qualifications for examination 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

Prairie Island Nuclear Generating 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 10CFR55a, personnel performance demonstrations will be through the Electric Power Research Institute/Performance Demonstration Initiative. Qualifications will be per the requirements of ASME, B&PV Code, Section XI, 1995 Edition, with 1996 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 ASNT-SNT-TC-1A, 1984 edition and 1989 Edition, no Addenda of ASME B&PV Code, Section XI IWA-2300.
Visual (VT-1, VT-2, VT-3)	Non-destructive examination (NDE) personnel are qualified in accordance with their employers written practice, which meets the requirements of ASNT-SNT-TC-1A, 1984 Edition and 1989 Edition, no Addenda of ASME B&PV Code, Section XI IWA-2300.
Visual VT-2	Prairie Island Engineering personnel performing VT-2 examinations are

qualified in accordance with plant procedures, which meets the requirements of ANSI N45.2.6, ASNT-SNT-TC-1A, 1984 and 1992 Edition, and1989 Edition, no Addenda of ASME B&PV Code, Section XI IWA-2300.

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, is provided in the ASME B&PV Code, Section XI, 1989 Edition. 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. 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.

During regularly scheduled inservice inspection activities, insulation is removed from pressure retaining bolted connections on systems borated for the purpose of controlling reactivity in order to complete the VT-2 visual examinations required by ASME B&PV Code, Section XI and Nuclear Code Case N-533. Prairie Island procedures require the removal of insulation from bolted connections during the performance of the Reactor Coolant System pressure test or a four-hour hold time if insulation is present. These VT-2 examinations are performed at the frequency specified by ASME B&PV Code, Section XI for system pressure tests.

When boric acid residues are detected on components during a VT-2 examination, Prairie Island procedure requires that the leakage source and the areas of general corrosion be located.

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 Prairie Island's corrective action process, an action request (AR) shall be initiated upon discovery of equipment malfunction, damage, or degradation that is considered sudden or unexpected. The following conditions adverse to quality related to boric acid accumulations shall be documented by initiation of an AR:

a. Through-wall leakages identified from cracks or weld defects.
b. Degradation of fastener material, which may reduce cross sectional area greater than five percent.
c. Degradation of pressure boundaries, which may reduce wall thickness greater than or equal to 10 percent.
d. 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. Leak rate
b. Concentration (boric acid) of deposited fluid
c. Potential corrosion rate and related pressure boundary material thickness
d. Potential consequences on RCS integrity
e. Corrective actions/plan
f. Interim monitoring methods

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 the method for evaluating the potential for leakage in inaccessible are as is provided in ASME B&PV Code, Section XI. The inaccessible areas of the RCPB are the reactor vessel cylindrical shell and the six vessel nozzles and portions of the associated piping (2 cold legs, 2 hot legs, 2 safety injection). These components are contained within and surrounded by the concrete biological shield that encircles the vessel. The nozzles and piping include welds that undergo a surface examination once every 10 years per the ASME B&PV Code, Section XI 1989 Edition. These examinations require the removal of sand plug covers and/or sand plugs to facilitate access to the welds.

The RCPB is pressure tested each refueling outage as required by ASME B&PV Code, Section XI. Any leakage from the reactor vessel itself would drain by gravity to the sump directly beneath the reactor. Any leakage arising from the six vessel nozzles or associated piping would migrate either to the sump or through the biological shield penetration sleeve to a reactor coolant pump vault. During the test, a certified VT-2 Visual Examiner inspects the reactor vessel bottom head/ sump area and the vault penetration sleeves for indications of leakage.

The walls of the biological shield in the reactor vessel bottom head/ sump area are frequently wetted with refueling cavity water from cavity seal leaks and leaky sand plug cover joints and, therefore, exhibit boric acid staining. Potential through-wall leaks originating in the above-mentioned inaccessible areas of the RCPB and migrating toward the reactor sump would necessarily have to be of sufficient quantity to progress to the bottom of the vessel and then distinguish themselves amongst the existing staining (e.g. as wet or new boric acid) in order to be identified as a relevant indication.

The RCS is leak tested daily per plant Technical Specifications. Unidentified leakage is limited to 1 gallon per minute. Leaks to the sump under the reactor of sufficient quantity to cause the sump pump to run would be annunciated in the control room. Containment radiation monitors and humidity indicators are monitored on a daily basis to provide an indicator of leakage. These indicators are principally designed to check for much larger quantities of leakage than would normally be detectable during direct visual examination per ASME B&PV Code, Section XI.

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 Prairie Island corrective action program. Upon discovery of 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 Prairie Island. Any repair or replacement of a component shall satisfy ASME B&PV Code, Section XI Article IWA-4000 or IWA-7000, 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 acceptance criteria 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. If the acceptance of the component is by analytical evaluation, the evaluation analysis is to be 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 the leakage will be removed;

2. The bolt will receive a visual VT-3 examination;

3. The visual examination results shall be evaluated in accordance with IWA-3100.

In a situation where the leakage is not stopped, consideration of the variables above and an evaluation of structural integrity of the bolting, consequences of continued operation, and the effect on system operability for continued leakage will be the basis for deferral of removing the bolt closest to the source of leakage. Continued monitoring and trending shall occur as determined by the corrective action program and NRC commitments. If continued operation cannot be justified, then the component is repaired prior to resumption of service. The use of code case N-566-1 to justify continued operation with equipment degraded beyond ASME B&PV Code, Section XI criteria generally occurs less than once per fuel cycle.

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

Through wall cracking of the bottom reactor vessel head instrument tube penetration nozzles would be detected by the visual observation of leakage through the mirror insulation, accessible via the reactor vessel sump. The insulation is examined as part of the code-required system pressure test each refueling outage by a certified examiner. Evidence of leakage would be visible as boric acid deposits or wetted areas at insulation joints or at the cutouts where each thimble tube penetrates the insulation. Additionally, a remote VT-3 with a mini-sub is performed on the nozzles every 10 years from inside the reactor vessel when the core barrel is removed.

Detection of low levels of RCPB leakage that may result from through-wall cracking in certain components and configurations for other small diameter nozzles is provided by the automatic systems that monitor the containment atmosphere radioactivity and the containment sump level.

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

a) Containment Air Particulate Monitor - This system takes continuous air samples from the containment atmosphere and measures the air particulate gamma radioactivity. The samples, drawn from outside the containment, are in a closed, sealed system and are monitored by a scintillation counter -- filter paper detector assembly. The containment air particulate monitor is the most sensitive instrument of those available for detection of reactor coolant leakage into the containment.

b) Containment Radiogas Monitor - This system measures the gaseous radioactivity in the containment by taking continuous air samples from the containment atmosphere, after they pass through the air particulate monitors, and drawing the samples through a closed, sealed system to a gas monitor assembly. The containment radioactive gas monitor is inherently less sensitive than the containment air particulate monitor, and would function in the event that significant reactor coolant gaseous activity exists from fuel cladding defects.

c) Humidity Detector - The humidity detection instrumentation offers another means of detection of leakage into the containment. Although this instrumentation has not nearly the sensitivity of the air particulate monitor, it has the characteristics of being sensitive to vapor originating from all sources within the containment, including the reactor coolant and steam and feedwater systems. Plots of containment air dew point variations above a base-- line maximum established by the cooling water temperature to the air coolers should be sensitive to incremental leakage equivalent to 2 to 10 gpm. The sensitivity of this method depends on cooling water temperature, containment air temperature variation and containment air recirculation rate.

d) Condensate Measuring System - This leak detection method is based on the principle that the condensate collected by the cooling coils matches, under equilibrium conditions, the leakage of water and steam from systems within the containment. This principle applies because conditions within the containment promote complete evaporation of leaking water from hot systems. The air and internal structure temperatures are normally 80°F to 105°F, the relative humidity of the air is well below the saturation point, and the cooling coils provide the only significant surfaces at or below the dew point temperature.

The containment fan coil units are designed to remove the sensible heat generated within the containment. The resulting large coil surface area has the effect that the exit air from the coils has a dew point temperature which is very nearly equal to the cooling water temperature.

Measurement of the condensate drained from each of the fan coil units is made to determine condensation rate and thus leak rate. Should a leak occur, the condensation rate will increase above the previous steady state due to the increased vapor content of the fan coil air intake. A new equilibrium rate will be approached within approximately 200 minutes after the start of the leak. Detection of the increasing condensation rate is possible, however, within 5 to 10 minutes for initial condensation rates on the order of .05 gpm and larger (which correspond to leakage rates of .5 gpm and larger). Readout of each condensate flow measuring device is provided in the Auxiliary Building. A high flow alarm is provided in the control room to alert the operator to significant increases in the condensate flow rate.

e) Containment Sump A pump elapsed run time - The run time is recorded daily and reviewed weekly to check for abnormal water leakage in containment. Chemical analysis of the water is necessary to determine the system (RCS or other) source of the leakage.

f) Containment at Power Inspection – This inspection is conducted monthly to check for leakage. It is important to note that this inspection, although all areas cannot be viewed, is more sensitive to cool areas where sump run times may not be as sensitive in detecting small leakage and is effective at finding very small leaks.

g) Reactor Coolant System Leak Test – This test is conducted daily to monitor RCS (also includes charging and letdown) leakage. Although this test is principally designed to check for leakage greater than Technical Specification-allowable, it provides additional trending of leakage.

Plant procedures require that whenever the unidentified leak rate exceeds 0.1 gpm for 3 consecutive days, that a leakage investigation is initiated per a controlled procedure. If the unidentified leakage remains above 0.1 gpm following the investigation, the boric acid corrosion control (BACC) program becomes controlling. This procedure would direct visual walkdowns of containment, local air grab samples, sump water chemical analyses, and evaluation of condensate collection pot flow rates. The possible consequences of continued operation with the leak, including boric acid corrosion, are required by this procedure to be evaluated.

Leakage from the RCS is maintained as low as reasonably possible. Review of recent plant records shows that Prairie Island operates with low unidentified RCS leakage. Through-wall leakage of the RCPB, excluding SG tubing, is not permitted per Technical Specifications and must be corrected.

Unidentified RCS leakage is limited to one gpm per Technical Specifications. One gpm of unidentified leakage from within the RCPB is allowed as a reasonable minimum detectable amount that the containment air monitoring and containment sump level monitoring equipment can detect within a reasonable time period. Leak rate calculations, monitoring, and trending are routinely performed during normal operation to identify increases in RCS leakage at levels less than the Technical Specification limits. When an increase in RCS leakage is detected, NMC implements administrative controls and investigates the source of leakage. The investigation involves performing appropriate chemistry sampling and radiation monitoring, additional trending, generating a corrective action and attempting to locate the source of RCS leakage.

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 effect 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 Prairie Island BACC program does not use susceptibility models or consequence models. The BACC program in use at the Prairie Island 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

Prairie Island's RPV vendor, Westinghouse, has made no recommendations for visual inspections on nozzles with Alloy 600/82/182 material.

Prairie Island implements an Operating Experience Program to ensure that lessons learned from industry operating experience (OE) are translated into appropriate action to improve nuclear safety, personnel safety, and plant reliability. The expectation is that effective use of operating experience information is the responsibility of each employee.

Personnel screen and review OE documents for applicability to any plant activity or program, 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 OE.

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 Integrity Test procedure, specifically in relation to system leakage and hydrostatic pressure testing.

NMC has concluded that the inspections and evaluations described above comply with PINGP Technical Specifications (TS) based on the above-noted compliance with ASME B&PV Code, Section XI. PINGP TS 5.5 requires that a number of programs shall be established, implemented, and maintained. Specifically included among these programs, as described in TS 5.5.7, is the Inservice Testing Program, which provides controls for inservice testing of ASME B&PV Code, Code Class 1, 2, and 3 components. PINGP TS do not invoke any additional requirements or programs to control reactor coolant pressure boundary (RCPB) corrosion caused by leakage of boric acid, thus, compliance with ASME B&PV Code, Section XI is all that is required to comply with TS requirements. That is, there is not a TS RCPB BACC program separate from the Inservice Testing Program of TS 5.5.7.

The PINGP BACC program, particularly with respect to RCPB components other than the reactor pressure vessel 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:

- PINGP Procedure H2, "Program for Identification and Disposition of Small Reactor Coolant Leakage on Low Alloy Reactor Coolant Pressure Boundary Components"
  This procedure describes the boric acid leak inspection program at the PINGP. This program is designed to fulfill the requirements of Generic Letter 88-05. Additionally, the boric acid leak inspection program complements the ASME B&PV Code, Section XI pressure testing program by providing suitable methods for dispositioning boric acid leaks discovered during pressure testing. H2 specifically references SP 1070 (Unit 1) and SP 2070 (Unit 2), "Reactor Coolant System Integrity Test,"

and SP 1070 [2070] specifically invoke IWA-5250.

- 5AWI 3.16.0, "Plant Surveillance Program"
  The purpose of this procedure is to establish the requirements for the PINGP surveillance program.

- 5AWI 15.0.0, "Work Control Process"
  The purpose of this procedure is to define Work Control at the PINGP, provide overall process requirements and assign major responsibilities. It includes a roadmap to lower-tier implementing instructions. This Instruction provides goals, objectives, expectations and guidelines and assigns responsibilities for administering the Work Control Process.

- 5AWI 16.0.0, "Action Request Process"
  The purpose of this Instruction is to establish the standardized NMC Fleet Corrective Action (Action Request) process at the PINGP. It establishes expectations for documenting and tracking the resolution of issues and requests. It provides the framework to ensure that conditions adverse to quality, operability issues, and reportability issues are promptly identified, evaluated if necessary, and corrected as appropriate.

ATTACHMENT 2

NUCLEAR MANAGEMENT COMPANY, LLC
PRAIRIE ISLAND NUCLEAR GENERATING PLANT
DOCKET 50-282 AND 50-306

JANUARY 2003

TABLE 1 – ALLOY 600 PENETRATIONS,BOLTED CONNECTION INSPECTIONS AND CARBON STEEL PRESSURE VESSELS

15 Pages Follow

TABLE 1
ALLOY 600 PENETRATIONS,
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						
Reactor Vessel Head Penetrations 40 Penetrations	Visual ---------- Supplemental visual	VT-2	Insulated VT-2 ---------- 100%	Each Refueling Outage	Insulation Not Removed / Mirror insulation ---------- Through viewports in mirror insulation	H2 Program for Identification and Disposition of Small Reactor Coolant Leakage on Low Alloy Reactor Coolant Pressure Boundary Components
Reactor Vessel Head Vent 1 Penetration	Visual ---------- Supplemental visual	VT-2	Insulated VT-2 ---------- 100%	Each Refueling Outage	Insulation Not Removed / Mirror insulation ---------- Through viewports in mirror insulation	H2
Reactor Vessel Lower Head Incore Instrument Penetrations 36 Penetrations	Visual	VT-2	Insulated VT-2	Each Refueling Outage	Insulation Not Removed / Mirror insulation	H2

Component	Inspection Techniques	Personnel Qualifications	Extent of Coverage	Frequency	Degree of Insulation Removal / Insulation Type	Corrective Action
Bolted Connection Inspections, Unit 1						
MV-32166 1 Reactor Excess Letdown Line Isolation Motor Valve A	Visual	VT-2	100%	Each Refueling Outage	Non-Insulated	H2 Program for Identification and Disposition of Small Reactor Coolant Leakage on Low Alloy Reactor Coolant Pressure Boundary Components SP 1392 Unit 1 RCS Bolting Inspection
MV-32073 1 Safety Injection Cold Leg Injection Isolation Motor Valve	Visual	VT-2	100%	Each Refueling Outage	Non-Insulated	H2 SP 1392
MV-32074 11 Safety Injection Reactor Vessel Injection Isolation Motor Valve	Visual	VT-2	100%	Each Refueling Outage	Non-Insulated	H2 SP 1392
CV-31232 1 Pressurizer Power Operated Relief Valve A Control Valve	Visual	VT-2	100%	Each Refueling Outage	Insulation Removed/Mirror Insulation	H2 SP 1392
CV-31231 1 Pressurizer Power Operated Relief Valve B Control Valve	Visual	VT-2	100%	Each Refueling Outage	Insulation Removed/Mirror Insulation	H2 SP 1392
MV-32196 1 Pressurizer Power Operated Relief Valve Isolation B Motor Valve	Visual	VT-2	100%	Each Refueling Outage	Insulation Removed/Mirror Insulation	H2 SP 1392
MV-32195 1 Pressurizer Power Operated Relief Valve Isolation A Motor Valve	Visual	VT-2	100%	Each Refueling Outage	Insulation Removed/Mirror Insulation	H2 SP 1392
RC-10-1 11 Pressurizer Relief	Visual	VT-2	100%	Each Refueling Outage	Non-Insulated	H2 SP 1392
RC-10-2 11 Pressurizer Relief	Visual	VT-2	100%	Each Refueling Outage	Non-Insulated	H2 SP 1392
CV-31224 11 Reactor Coolant Pump Loop A Pressurizer Spray Control Valve	Visual	VT-2	100%	Each Refueling Outage	Insulation Removed/Mirror Insulation	H2 SP 1392
RC-2-1, Loop A RTD Loops Outlet	Visual	VT-2	100%	Each Refueling Outage	Insulation Removed/Mirror Insulation	H2 SP 1392
FE-458 Loop A RTD Flow Element FLG	Visual	VT-2	100%	Each Refueling Outage	Insulation Removed/Mirror Insulation	H2 SP 1392

TABLE 1
ALLOY 600 PENETRATIONS,
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
RC-18-1 Reactor Coolant System Loop A Hot Leg Drain to Reactor Coolant Drain Tank	Visual	VT-2	100%	Each Refueling Outage	Non-Insulated	H2 SP 1392
RC-18-2 Reactor Coolant System Loop A Hot Leg Drain to Reactor Coolant Drain Tank	Visual	VT-2	100%	Each Refueling Outage	Non-Insulated	H2 SP 1392
SI-6-2 12 Accumulator Loop B Check Valve Downstream of SI-6-1	Visual	VT-2	100%	Each Refueling Outage	Insulation Removed/Mirror Insulation	H2 SP 1392
SI-9-2 Cold Leg Injection Line to Loop A Cold Leg-Check Valve	Visual	VT-2	100%	Each Refueling Outage	Insulation Removed/Mirror Insulation	H2 SP 1392
RC-7-2 Loop A to Pressurizer CV-31224 Bypass	Visual	VT-2	100%	Each Refueling Outage	Insulation Removed/Mirror Insulation	H2 SP 1392
2 - 2" Flanges 11 Reactor Coolant Pump Seal Injection	Visual	VT-2	100%	Each Refueling Outage	Non-Insulated	H2 SP 1392
2 - 3/4" Flanges 11 Reactor Coolant Pump Seal Bypass	Visual	VT-2	100%	Each Refueling Outage	Non-Insulated	H2 SP 1392
1 - 2" Flange 11 Reactor Coolant Pump #1 Seal Outlet	Visual	VT-2	100%	Each Refueling Outage	Non-Insulated	H2 SP 1392
1 - 1.5" Flange 11 Reactor Coolant Pump #1 Seal Outlet	Visual	VT-2	100%	Each Refueling Outage	Non-Insulated	H2 SP 1392
CV-31225 12 Reactor Coolant Pump Loop B Pressurizer Spray Control Valve	Visual	VT-2	100%	Each Refueling Outage	Insulation Removed/Mirror Insulation	H2 SP 1392
SI-6-4 11 Accumulator Loop A Check Valve Downstream of SI-6-3	Visual	VT-2	100%	Each Refueling Outage	Insulation Removed/Mirror Insulation	H2 SP 1392
SI-9-1 Cold Leg Injection Line to Loop B Cold Leg Check	Visual	VT-2	100%	Each Refueling Outage	Insulation Removed/Mirror Insulation	H2 SP 1392
RC-2-2 Loop B RTD Loops Outlet	Visual	VT-2	100%	Each Refueling Outage	Insulation Removed/Mirror Insulation	H2 SP 1392
RC-7-1 Loop B to Pressurizer CV-31225 Bypass	Visual	VT-2	100%	Each Refueling Outage	Insulation Removed/Mirror Insulation	H2 SP 1392
RC-16-2 Reactor Vessel Level Indication System Loop B Hot Leg Isolation FOR RC-17-4	Visual	VT-2	100%	Each Refueling Outage	Non-Insulated	H2 SP 1392
FE-459 Loop B RTD Outlet Flow Element Flange	Visual	VT-2	100%	Each Refueling Outage	Insulation Removed/Mirror Insulation	H2 SP 1392

TABLE 1
ALLOY 600 PENETRATIONS,
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
2 - 2" Flanges 12 Reactor Coolant Pump Seal Injection	Visual	VT-2	100%	Each Refueling Outage	Insulation Removed/Mirror Insulation	H2 SP 1392
2 - 3/4" Flanges 12 Reactor Coolant Pump Seal Bypass	Visual	VT-2	100%	Each Refueling Outage	Insulation Removed/Mirror Insulation	H2 SP 1392
1 - 2" Flange 12 Reactor Coolant Pump #1 Seal Outlet	Visual	VT-2	100%	Each Refueling Outage	Insulation Removed/Mirror Insulation	H2 SP 1392
1 - 1.5" Flange 12 Reactor Coolant Pump #1 Seal Outlet	Visual	VT-2	100%	Each Refueling Outage	Insulation Removed/Mirror Insulation	H2 SP 1392
RC-9-1 Reactor Vessel Flange Outer Seal Leak-off	Visual	VT-2	100%	Each Refueling Outage	Insulation Removed/Mirror Insulation	H2 SP 1392
RC-9-2 Reactor Vessel Flange Inner Seal Leak-off	Visual	VT-2	100%	Each Refueling Outage	Insulation Removed/Mirror Insulation	H2 SP 1392
MV-32164 1 Reactor Coolant System Loop A Hot Leg Residual Heat Removal Supply (Inside) Motor Valve	Visual	VT-2	100%	Each Refueling Outage	Insulation Removed/Mirror Insulation	H2 SP 1392
MV-32230 1 Reactor Coolant System Loop B Hot Leg Residual Heat Removal Supply (Inside) Motor Valve	Visual	VT-2	100%	Each Refueling Outage	Insulation Removed/Mirror Insulation	H2 SP 1392
CV-31328 11 Regenerative Heat Exchanger CHG Line to Reactor Coolant System Control Valve	Visual	VT-2	100%	Each Refueling Outage	Insulation Removed/Mirror Insulation	H2 SP 1392
CV-31329 11 Regenerative Heat Exchanger Auxiliary Spray to 11 Pressurizer Control Valve	Visual	VT-2	100%	Each Refueling Outage	Insulation Removed/Mirror Insulation	H2 SP 1392
CV-31255 U1 Letdown Line Isolation Train B Control Valve	Visual	VT-2	100%	Each Refueling Outage	Insulation Removed/Mirror Insulation	H2 SP 1392
CV-31226 U1 Letdown Line Isolation Train A Control Valve	Visual	VT-2	100%	Each Refueling Outage	Insulation Removed/Mirror Insulation	H2 SP 1392
RC-19-1 Loop A Pressurizer Spray to Reactor Coolant Drain Tank Shutdown Purification	Visual	VT-2	100%	Each Refueling Outage	Non-Insulated	H2 SP 1392
RC-19-2 Loop A Pressurizer Spray to Reactor Coolant Drain Tank Shutdown Purification	Visual	VT-2	100%	Each Refueling Outage	Non-Insulated	H2 SP 1392
RC-18-5 Reactor Coolant System Loop A Pressurizer Spray Manual Isolation	Visual	VT-2	100%	Each Refueling Outage	Insulation Removed/Mirror Insulation	H2 SP 1392

TABLE 1
ALLOY 600 PENETRATIONS,
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
CV-31325 11 Regenerative Heat Exchanger Letdown Line Isolation Control Valve A	Visual	VT-2	100%	Each Refueling Outage	Insulation Removed/Mirror Insulation	H2 SP 1392
CV-31326 11 Regenerative Heat Exchanger Letdown Line Isolation Control Valve B	Visual	VT-2	100%	Each Refueling Outage	Insulation Removed/Mirror Insulation	H2 SP 1392
CV-31327 11 Regenerative Heat Exchanger Letdown Line Isolation Control Valve C	Visual	VT-2	100%	Each Refueling Outage	Insulation Removed/Mirror Insulation	H2 SP 1392
CV-31330 11 Excess Letdown Heat Exchanger Inlet Loop A Isolation Control Valve	Visual	VT-2	100%	Each Refueling Outage	Insulation Removed/Mirror Insulation	H2 SP 1392
CV-31210 11 Excess Letdown Heat Exchanger Outlet Flow Control Valve	Visual	VT-2	100%	Each Refueling Outage	Insulation Removed/Mirror Insulation	H2 SP 1392
CV-31339 11 Letdown Heat Exchanger Inlet Isolation Control Valve	Visual	VT-2	100%	Each Refueling Outage	Insulation Removed/Mirror Insulation	H2 SP 1392
CV-31333 11 Excess Letdown Diversion to Volume Control Tank/Reactor Coolant Drain Tank Control Valve	Visual	VT-2	100%	Each Refueling Outage	Non-Insulated	H2 SP 1392
Excess Letdown Heat Exchanger Bolting	Visual	VT-2	100%	Each Refueling Outage	Insulation Removed/Mirror Insulation	H2 SP 1392
CV-31334 11/12 Reactor Coolant Pump Seal Bypass Return Control Valve	Visual	VT-2	100%	Each Refueling Outage	Non-Insulated	H2 SP 1392
CV-31445 12 Accumulator Makeup Isolation Control Valve	Visual	VT-2	100%	Each Refueling Outage	Non-Insulated	H2 SP 1392
CV-31450 12 Accumulator Reactor Coolant Test (Before Check) Isolation Control Valve	Visual	VT-2	100%	Each Refueling Outage	Non-Insulated	H2 SP 1392
CV-31449 12 Accumulator Reactor Coolant Test (After Check) Isolation Control Valve	Visual	VT-2	100%	Each Refueling Outage	Non-Insulated	H2 SP 1392
SI-6-1 12 Accumulator to Reactor Coolant System Loop B Cold Leg Check	Visual	VT-2	100%	Each Refueling Outage	Non-Insulated	H2 SP 1392

TABLE 1
ALLOY 600 PENETRATIONS,
BOLTED CONNECTION INSPECTIONS
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Component	Inspection Techniques	Personnel Qualifications	Extent of Coverage	Frequency	Degree of Insulation Removal / Insulation Type	Corrective Action
SI-9-3 Low Head Safety Injection to 12 Reactor Vessel Nozzle Check	Visual	VT-2	100%	Each Refueling Outage	Non-Insulated	H2 SP 1392
SI-9-5 High/Low Head Safety Injection to 12 Reactor Vessel Check	Visual	VT-2	100%	Each Refueling Outage	Insulation Removed/Mirror Insulation	H2 SP 1392
VC-8-1 11 Regenerative Heat Exchanger Charging Inlet - Check Valve (Spool Piece)	Visual	VT-2	100%	Each Refueling Outage	Non-Insulated	H2 SP 1392
VC-8-5 11 Reactor Coolant Pump Seal Water Injection Check (Spool Piece)	Visual	VT-2	100%	Each Refueling Outage	Non-Insulated	H2 SP 1392
VC-8-4 12 Reactor Coolant Pump Seal Water Injection Check (Spool Piece)	Visual	VT-2	100%	Each Refueling Outage	Non-Insulated	H2 SP 1392
MV-32199 1 Reactor Coolant Pump Seal Return/Excess Letdown Isolation Train B Motor Valve	Visual	VT-2	100%	Each Refueling Outage	Non-Insulated	H2 SP 1392
VC-25-1 11/12 Reactor Coolant Pump Discharge Line to Seal Water Filter Relief	Visual	VT-2	100%	Each Refueling Outage	Non-Insulated	H2 SP 1392
CV-31442 11 Accumulator Makeup Isolation Control Valve	Visual	VT-2	100%	Each Refueling Outage	Non-Insulated	H2 SP 1392
CV-31447 11 Accumulator Reactor Coolant Test (After Check) Isolation Control Valve	Visual	VT-2	100%	Each Refueling Outage	Non-Insulated	H2 SP 1392
CV-31448 11 Accumulator Reactor Coolant Test (Before Check) Isolation Control Valve	Visual	VT-2	100%	Each Refueling Outage	Non-Insulated	H2 SP 1392
SI-6-3 11 Accumulator to Reactor Coolant System Loop A Cold Leg Check	Visual	VT-2	100%	Each Refueling Outage	Non-Insulated	H2 SP 1392
SI-9-6 High/Low Head Safety Injection to 11 Reactor Vessel Check	Visual	VT-2	100%	Each Refueling Outage	Insulation Removed/Mirror Insulation	H2 SP 1392
SI-9-4 Low Head Safety Injection to 11 Reactor Vessel Nozzle Check	Visual	VT-2	100%	Each Refueling Outage	Non-Insulated	H2 SP 1392

TABLE 1
ALLOY 600 PENETRATIONS,
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
MV-32064 Residual Heat Removal to 1 Reactor Vessel Injection Isolation Train A Motor Valve	Visual	VT-2	100%	Each Refueling Outage	Non-Insulated	H2 SP 1392
MV-32065 Residual Heat Removal to 1 Reactor Vessel Injection Isolation Train B Motor Valve	Visual	VT-2	100%	Each Refueling Outage	Non-Insulated	H2 SP 1392
MV-32066 1 Reactor Coolant System Loop B Cold Leg Residual Heat Removal InjectionMotor Valve	Visual	VT-2	100%	Each Refueling Outage	Insulation Removed/Mirror Insulation	H2 SP 1392
MV-32165 1 Reactor Coolant System Loop A Hot Leg Residual Heat Removal Supply (Outside) Motor Valve	Visual	VT-2	100%	Each Refueling Outage	Insulation Removed/Mirror Insulation	H2 SP 1392
MV-32231 1 Reactor Coolant System Loop B Hot Leg Residual Heat Removal Supply (Outside) Motor Valve	Visual	VT-2	100%	Each Refueling Outage	Insulation Removed/Mirror Insulation	H2 SP 1392
MV-32234 U1 Residual Heat Removal to 11 Letdown Heat Exchanger Inlet Motor Valve	Visual	VT-3	100%	Each Refueling Outage	Non-Insulated	H2 SP 1392
SI-26-1 Residual Heat Removal Heat Exchanger to 1 Reactor Vessel Loop A Relief Valve to Pressurizer Relief Tank	Visual	VT-2	100%	Each Refueling Outage	Non-Insulated	H2 SP 1392
CV-31335 11 Reactor Coolant Pump SL Water Outlet Isolation Control Valve	Visual	VT-2	100%	Each Refueling Outage	Non-Insulated	H2 SP 1392
1 - 3/4" Flange 11 Reactor Coolant Pump Seal Bypass orifice	Visual	VT-2	100%	Each Refueling Outage	Non-Insulated	H2 SP 1392
2 - 3/4" Flanges 11 Reactor Coolant Pump Seal Bypass FE-179	Visual	VT-2	100%	Each Refueling Outage	Non-Insulated	H2 SP 1392
CV-31336 12 Reactor Coolant Pump SL Water Outlet Isolation Control Valve	Visual	VT-2	100%	Each Refueling Outage	Non-Insulated	H2 SP 1392
1 - 3/4" Flange 12 Reactor Coolant Pump Seal Bypass Orifice	Visual	VT-2	100%	Each Refueling Outage	Non-Insulated	H2 SP 1392

TABLE 1
ALLOY 600 PENETRATIONS,
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
2 - 3/4" Flanges 12 Reactor Coolant Pump Seal Bypass FE-180	Visual	VT-2	100%	Each Refueling Outage	Non-Insulated	H2 SP 1392
2 - Reactor Head Vent PPG Flanges	Visual	VT-2	100%	Each Refueling Outage	Non-Insulated	H2 SP 1392
3 - Reactor Head Thermocouple Instrument Port Mechanical Seal	Visual	VT-2	100%	Each Refueling Outage	Non-Insulated	H2 SP 1392
1 - Reactor Head Flange	Visual	VT-2	100%	Each Refueling Outage	Insulation Not Removed/Mirror Insulation	H2 SP 1392
2 - Steam Generator Primary Manways	Visual	VT-2	100%	Each Refueling Outage	Insulation Removed/Mirror Insulation	H2 SP 1392
1 - Pressurizer Manways	Visual	VT-2	100%	Each Refueling Outage	Insulation Removed/Mirror Insulation	H2 SP 1392
2 - Reactor Coolant Pump Flange and Seal Areas	Visual	VT-2	100%	Each Refueling Outage	Non-Insulated	H2 SP 1392
Bolted Connection Inspections, Unit 2						
MV-32194 2 Reactor Excess Letdown Line Isolation Motor Valve A	Visual	VT-2	100%	Each Refueling Outage	Non-Insulated	H2 Program for Identification and Disposition of Small Reactor Coolant Leakage on Low Alloy Reactor Coolant Pressure Boundary Components SP 2392 Unit 2 RCS Bolting Inspection
MV-32176 2 Safety Injection Cold Leg InjectionIsolation Motor Valve	Visual	VT-2	100%	Each Refueling Outage	Non-Insulated	H2 SP 2392
MV-32177 21 Safety Injection Reactor Vessel InjectionIsolation Motor Valve	Visual	VT-2	100%	Each Refueling Outage	Non-Insulated	H2 SP 2392
CV-31233 2 Pressurizer Power Operated Relief Valve B Control Valve	Visual	VT-2	100%	Each Refueling Outage	Insulation Removed/Mirror Insulation	H2 SP 2392
CV-31234 2 Pressurizer Power Operated Relief Valve A Control Valve	Visual	VT-2	100%	Each Refueling Outage	Insulation Removed/Mirror Insulation	H2 SP 2392
MV-32197 2 Pressurizer Power Operated Relief Valve Isolation A Motor Valve	Visual	VT-2	100%	Each Refueling Outage	Insulation Removed/Mirror Insulation	H2 SP 2392

TABLE 1
ALLOY 600 PENETRATIONS,
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
MV-32198 2 Pressurizer Power Operated Relief Valve Isolation B Motor Valve	Visual	VT-2	100%	Each Refueling Outage	Insulation Removed/Mirror Insulation	H2 SP 2392
CV-31228 21 Reactor Coolant Pump Loop A Pressurizer Spray Control Valve	Visual	VT-2	100%	Each Refueling Outage	Insulation Removed/Mirror Insulation	H2 SP 2392
2RC-2-1 Loop A RTD Loops - Outlet	Visual	VT-2	100%	Each Refueling Outage	Insulation Removed/Mirror Insulation	H2 SP 2392
2FE-458 2 Reactor Coolant System Loop A RTD Bypass Flow Orifice	Visual	VT-2	100%	Each Refueling Outage	Insulation Removed/Mirror Insulation	H2 SP 2392
2SI-6-2 22 Accumulator Loop B Check Valve Downstream of 2SI-6-1	Visual	VT-2	100%	Each Refueling Outage	Insulation Removed/Mirror Insulation	H2 SP 2392
2SI-9-1 Cold Leg Injection Line to Loop B Cold Leg Check	Visual	VT-2	100%	Each Refueling Outage	Insulation Removed/Mirror Insulation	H2 SP 2392
2RC-7-2 Loop A to Pressurizer Control Valve 31228 Bypass	Visual	VT-2	100%	Each Refueling Outage	Insulation Removed/Mirror Insulation	H2 SP 2392
2 - 2" Flanges 21 Reactor Coolant Pump Seal Injection	Visual	VT-2	100%	Each Refueling Outage	Non-Insulated	H2 SP 2392
2 - 3/4" Flanges 21 Reactor Coolant Pump Seal Bypass	Visual	VT-2	100%	Each Refueling Outage	Non-Insulated	H2 SP 2392
1 - 2" Flange 21 Reactor Coolant Pump #1 Seal Outlet	Visual	VT-2	100%	Each Refueling Outage	Non-Insulated	H2 SP 2392
1 - 1.5" Flange 21 Reactor Coolant Pump #1 Seal Outlet	Visual	VT-2	100%	Each Refueling Outage	Non-Insulated	H2 SP 2392
CV-31229 22 Reactor Coolant Pump Loop B Pressurizer Spray Control Valve	Visual	VT-2	100%	Each Refueling Outage	Insulation Removed/Mirror Insulation	H2 SP 2392
2SI-6-4 21 Accumulator Loop A Check Valve Downstream of 2SI-6-3	Visual	VT-2	100%	Each Refueling Outage	Insulation Removed/Mirror Insulation	H2 SP 2392
2SI-9-2 Cold Leg InjectionLine to Loop A Cold Leg Check	Visual	VT-2	100%	Each Refueling Outage	Insulation Removed/Mirror Insulation	H2 SP 2392
2RC-2-2 Loop B RTD Loops - Outlet	Visual	VT-2	100%	Each Refueling Outage	Insulation Removed/Mirror Insulation	H2 SP 2392

TABLE 1
ALLOY 600 PENETRATIONS,
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
2RC-7-1 Loop B to Pressurizer Control Valve 31229 Bypass	Visual	VT-2	100%	Each Refueling Outage	Insulation Removed/Mirror Insulation	H2 SP 2392
2RC-16-2 Reactor Vessel Level Indication System Loop B Hot Leg Isolation FOR 2RC-17-4	Visual	VT-2	100%	Each Refueling Outage	Non-Insulated	H2 SP 2392
2FE-459 2 Reactor Coolant System Loop B RTD Bypass Flow Orifice	Visual	VT-2	100%	Each Refueling Outage	Insulation Removed/Mirror Insulation	H2 SP 2392
2RC-18-1 Reactor Coolant System Loop B Hot Leg Drain to Reactor Coolant Drain Tank	Visual	VT-2	100%	Each Refueling Outage	Non-Insulated	H2 SP 2392
2RC-18-2 Reactor Coolant System Loop B Hot Leg Drain to Reactor Coolant Drain Tank	Visual	VT-2	100%	Each Refueling Outage	Non-Insulated	H2 SP 2392
2RC-10-1 21 Pressurizer Relief	Visual	VT-2	100%	Each Refueling Outage	Non-Insulated	H2 SP 2392
2RC-10-2 21 Pressurizer Relief	Visual	VT-2	100%	Each Refueling Outage	Non-Insulated	H2 SP 2392
2 - 2" Flanges 22 Reactor Coolant Pump Seal Injection	Visual	VT-2	100%	Each Refueling Outage	Non-Insulated	H2 SP 2392
2 - 3/4" Flanges 22 Reactor Coolant Pump Seal Bypass	Visual	VT-2	100%	Each Refueling Outage	Non-Insulated	H2 SP 2392
1 - 2" Flange 22 Reactor Coolant Pump #1 Seal Outlet	Visual	VT-2	100%	Each Refueling Outage	Non-Insulated	H2 SP 2392
1 - 1.5" Flange 22 Reactor Coolant Pump #1 Seal Outlet	Visual	VT-2	100%	Each Refueling Outage	Non-Insulated	H2 SP 2392
2RC-9-1 Reactor Vessel Flange InnerSeal Leak Off	Visual	VT-2	100%	Each Refueling Outage	Non-Insulated	H2 SP 2392
2RC-9-2 Reactor Vessel Flange OuterSeal Leak Off	Visual	VT-2	100%	Each Refueling Outage	Non-Insulated	H2 SP 2392
MV-32192 2 Reactor Coolant System Loop A Hot Leg Residual Heat Removal Supply (Inside) Motor Valve	Visual	VT-2	100%	Each Refueling Outage	Insulation Removed/Mirror Insulation	H2 SP 2392
MV-32232 2 Reactor Coolant System Loop B Hot Leg Residual Heat Removal Supply (Inside) Motor Valve	Visual	VT-2	100%	Each Refueling Outage	Insulation Removed/Mirror Insulation	H2 SP 2392

TABLE 1
ALLOY 600 PENETRATIONS, 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
CV-31420 21 Regenerative Heat Exchanger CHG Line to Reactor Coolant System Control Valve	Visual	VT-2	100%	Each Refueling Outage	Insulation Removed/Mirror Insulation	H2 SP 2392
CV-31421 21 Regenerative Heat Exchanger Auxiliary Spray to 21 Pressurizer Control Valve	Visual	VT-2	100%	Each Refueling Outage	Insulation Removed/Mirror Insulation	H2 SP 2392
CV-31279 U2 Letdown Line Isolation Train B Control Valve	Visual	VT-2	100%	Each Refueling Outage	Insulation Removed/Mirror Insulation	H2 SP 2392
CV-31230 U2 Letdown Line Isolation Train A Control Valve	Visual	VT-2	100%	Each Refueling Outage	Insulation Removed/Mirror Insulation	H2 SP 2392
2RC-19-1 Loop A Pressurizer Spray to Reactor Coolant Drain Tank Shutdown Purification	Visual	VT-2	100%	Each Refueling Outage	Non-Insulated	H2 SP 2392
2RC-19-2 Loop A Pressurizer Spray to Reactor Coolant Drain Tank Shutdown Purification	Visual	VT-2	100%	Each Refueling Outage	Non-Insulated	H2 SP 2392
2RC-18-5 Reactor Coolant System Loop A Pressurizer Spray Manual Isolation	Visual	VT-2	100%	Each Refueling Outage	Insulation Removed/Mirror Insulation	H2 SP 2392
CV-31347 21 Letdown Orifice Isolation A 40 GPM Control Valve	Visual	VT-2	100%	Each Refueling Outage	Insulation Removed/Mirror Insulation	H2 SP 2392
CV-31348 21 Letdown Orifice Isolation B 40 GPM Control Valve	Visual	VT-2	100%	Each Refueling Outage	Insulation Removed/Mirror Insulation	H2 SP 2392
CV-31349 21 Letdown Orifice Isolation C 80 GPM Control Valve	Visual	VT-2	100%	Each Refueling Outage	Insulation Removed/Mirror Insulation	H2 SP 2392
CV-31422 21 Excess Letdown Heat Exchanger Inlet Isolation Control Valve	Visual	VT-2	100%	Each Refueling Outage	Insulation Removed/Mirror Insulation	H2 SP 2392
CV-31222 21 Excess Letdown Heat Exchanger Outlet Flow Control Valve	Visual	VT-2	100%	Each Refueling Outage	Insulation Removed/Mirror Insulation	H2 SP 2392
Excess Letdown Heat Exchanger Bolting	Visual	VT-2	100%	Each Refueling Outage	Insulation Removed/Mirror Insulation	H2 SP 2392

TABLE 1
ALLOY 600 PENETRATIONS,
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
CV-31424 21 Excess Letdown Diversion to Volume Control Tank/Reactor Coolant Drain Tank Control Valve	Visual	VT-2	100%	Each Refueling Outage	Non-Insulated	H2 SP 2392
CV-31425 21/22 Reactor Coolant Pump Seal Bypass Return Control Valve	Visual	VT-2	100%	Each Refueling Outage	Non-Insulated	H2 SP 2392
CV-31518 22 Accumulator Make-up Isolation Control Valve	Visual	VT-2	100%	Each Refueling Outage	Non-Insulated	H2 SP 2392
CV-31462 22 Accumulator Reactor Coolant Test (Before Check) Isolation Control Valve	Visual	VT-2	100%	Each Refueling Outage	Non-Insulated	H2 SP 2392
CV-31461 22 Accumulator Reactor Coolant Test (After Check) Isolation Control Valve	Visual	VT-2	100%	Each Refueling Outage	Non-Insulated	H2 SP 2392
2SI-6-1 22 Accumulator to Reactor Coolant System Loop B Cold Leg Check	Visual	VT-2	100%	Each Refueling Outage	Non-Insulated	H2 SP 2392
2SI-9-3 Low Head Safety Injection to 22 Reactor Vessel Nozzle Check	Visual	VT-2	100%	Each Refueling Outage	Non-Insulated	H2 SP 2392
2SI-9-5 High/Low Head Safety Injection to 22 Reactor Vessel Nozzle Check	Visual	VT-2	100%	Each Refueling Outage	Insulation Removed/Mirror Insulation	H2 SP 2392
CV-31517 21 Accumulator Make-up Isolation Control Valve	Visual	VT-2	100%	Each Refueling Outage	Non-Insulated	H2 SP 2392
CV-31459 21 Accumulator Reactor Coolant Test (After Check) Isolation Control Valve	Visual	VT-2	100%	Each Refueling Outage	Non-Insulated	H2 SP 2392
CV-31460 21 Accumulator Reactor Coolant Test (Before Check) Isolation Control Valve	Visual	VT-2	100%	Each Refueling Outage	Non-Insulated	H2 SP 2392
2SI-6-3 21 Accumulator to Reactor Coolant System Loop A Cold Leg Check	Visual	VT-2	100%	Each Refueling Outage	Non-Insulated	H2 SP 2392
2SI-9-6 High/Low Head Safety Injection to 21 Reactor Vessel Nozzle Check	Visual	VT-2	100%	Each Refueling Outage	Insulation Removed/Mirror Insulation	H2 SP 2392

TABLE 1
ALLOY 600 PENETRATIONS,
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
2SI-9-4 Low Head Safety Injection to 21 Reactor Vesse Nozzle Check	Visual	VT-2	100%	Each Refueling Outage	Non-Insulated	H2 SP 2392
MV-32167 Residual Heat Removal to 2 Reactor Vessel Injection Isolation Train A Motor Valve	Visual	VT-2	100%	Each Refueling Outage	Non-Insulated	H2 SP 2392
MV-32168 Residual Heat Removal to 2 Reactor Vessel Injection Isolation Train B Motor Valve	Visual	VT-2	100%	Each Refueling Outage	Non-Insulated	H2 SP 2392
MV-32169 2 Reactor Coolant System Loop B Cold Leg Residual Heat Removal InjectionMotor Valve	Visual	VT-2	100%	Each Refueling Outage	Insulation Removed/Mirror Insulation	H2 SP 2392
MV-32193 2 Reactor Coolant System Loop A Hot Leg Residual Heat Removal Supply (Outside) Motor Valve	Visual	VT-2	100%	Each Refueling Outage	Insulation Removed/Mirror Insulation	H2 SP 2392
MV-32233 2 Reactor Coolant System Loop B Hot Leg Residual Heat Removal Supply (Outside) Motor Valve	Visual	VT-2	100%	Each Refueling Outage	Insulation Removed/Mirror Insulation	H2 SP 2392
MV-32235 U2 Residual Heat Removal to 21 Letdown Heat Exchanger Inlet Motor Valve	Visual	VT-2	100%	Each Refueling Outage	Non-Insulated	H2 SP 2392
2SI-26-1 Residual Heat Removal Heat Exchanger to 2 Reactor Vessel Loop B Relief Valve to Pressurizer Relief Tank	Visual	VT-2	100%	Each Refueling Outage	Non-Insulated	H2 SP 2392
CV-31426 21 Reactor Coolant Pump Seal Water Outlet Isolation Control Valve	Visual	VT-2	100%	Each Refueling Outage	Non-Insulated	H2 SP 2392
1 - 3/4" Flange 21 Reactor Coolant Pump Seal Bypass orifice	Visual	VT-2	100%	Each Refueling Outage	Non-Insulated	H2 SP 2392
2 - 3/4" Flanges 21 Reactor Coolant Pump Seal Bypass 2FE-179	Visual	VT-2	100%	Each Refueling Outage	Non-Insulated	H2 SP 2392
CV-31427 22 Reactor Coolant Pump Seal Water Outlet Isolation Control Valve	Visual	VT-2	100%	Each Refueling Outage	Non-Insulated	H2 SP 2392

TABLE 1
ALLOY 600 PENETRATIONS,
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
1 - 3/4" Flange 22 Reactor Coolant Pump Seal Bypass orifice	Visual	VT-2	100%	Each Refueling Outage	Non-Insulated	H2 SP 2392
2 - 3/4" Flanges 22 Reactor Coolant Pump Seal Bypass 2FE-18(	Visual	VT-2	100%	Each Refueling Outage	Non-Insulated	H2 SP 2392
2VC-8-1 21 Regenerative Heat Exchanger Charging Inlet Check	Visual	VT-2	100%	Each Refueling Outage	Non-Insulated	H2 SP 2392
MV-32210 2 Reactor Coolant Pump Seal Return/Excess Letdown Isolation Train B Motor Valve	Visual	VT-2	100%	Each Refueling Outage	Non-Insulated	H2 SP 2392
2VC-25-1 21/22 Reactor Coolant Pump Discharge Line to Seal Water Filter Relief	Visual	VT-2	100%	Each Refueling Outage	Non-Insulated	H2 SP 2392
2VC-8-4 22 Reactor Coolant Pump Seal Water Injection Check	Visual	VT-2	100%	Each Refueling Outage	Non-Insulated	H2 SP 2392
2VC-8-5 21 Reactor Coolant Pump Seal Water Injection Check	Visual	VT-2	100%	Each Refueling Outage	Non-Insulated	H2 SP 2392
2 - Reactor Head Vent PPG Flanges	Visual	VT-2	100%	Each Refueling Outage	Non-Insulated	H2 SP 2392
CV-31430 21 Letdown Heat Exchanger Inlet Control Valve	Visual	VT-2	100%	Each Refueling Outage	Insulation Removed/Mirror Insulation	H2 SP 2392
3 - Reactor Head Thermocouple Instrument Port Mechanical Seal	Visual	VT-2	100%	Each Refueling Outage	Non-Insulated	H2 SP 2392
1 - Reactor Head Flange	Visual	VT-2	100%	Each Refueling Outage	Insulation Not Removed/Mirror Insulation	H2 SP 2392
2 - Steam Generator Primary Manways	Visual	VT-2	100%	Each Refueling Outage	Insulation Removed/Mirror Insulation	H2 SP 2392
1 - Pressurizer Manways	Visual	VT-2	100%	Each Refueling Outage	Insulation Removed/Mirror Insulation	H2 SP 2392
2 - Reactor Coolant Pump Flange and Seal Areas	Visual	VT-2	100%	Each Refueling Outage	Non-Insulated	H2 SP 2392

TABLE 1
ALLOY 600 PENETRATIONS,
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						
Reactor Vessel	Visual	VT-2	Insulated VT-2	Each Refueling Outage	Insulation not Removed/Mirror Insulation	H2 Program for Identification and Disposition of Small Reactor Coolant Leakage on Low Alloy Reactor Coolant Pressure Boundary Components SP 1070 [SP 2070] Reactor Coolant System Integrity Test
Pressurizer	Visual	VT-2	Insulated VT-2	Each Refueling Outage	Insulation not Removed/Mirror Insulation	H2 SP 1070 [SP 2070]
Steam Generator A	Visual	VT-2	Insulated VT-2	Each Refueling Outage	Insulation not Removed/Nukon Blanket with Metal Cover	H2 SP 1070 [SP 2070]
Steam Generator B	Visual	VT-2	Insulated VT-2	Each Refueling Outage	Insulation not Removed/Nukon Blanket with Metal Cover	H2 SP 1070 [SP 2070]