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Palisades Nuclear Plant
Operated by Nuclear Management Company, LLC

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PALISADES NUCLEAR PLANT
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BULLETIN 2002-01, "REACTOR PRESSURE VESSEL HEAD DEGRADATION AND REACTOR COOLANT PRESSURE BOUNDARY INTEGRITY," 60-DAY RESPONSE FOR THE PALISADES PLANT, REQUEST FOR ADDITIONAL INFORMATION (TAC NO. MB4562)

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 Palisades 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.

Douglas E. Cooper
Site Vice-President, Palisades Nuclear Plant

CC Regional Administrator, USNRC, Region III
Project Manager, Palisades Plant, USNRC, NRR
NRC Resident Inspector – Palisades Plant

Attachments

A095

ATTACHMENT 1

NUCLEAR MANAGEMENT COMPANY, LLC
PALISADES NUCLEAR PLANT
DOCKET 50-255

JANUARY 20, 2003

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

21 Pages Follow

BULLETIN 2002-01, "REACTOR PRESSURE VESSEL HEAD DEGRADATION AND REACTOR COOLANT PRESSURE BOUNDARY INTEGRITY," 60-DAY RESPONSE FOR THE PALISADES 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 connections 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 as required to fulfill the requirements of American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (BP&V) Code, Section XI, 1989 Edition as modified by Nuclear Regulatory Commission (NRC) commitments and an approved relief request for application of the inservice inspection program (ISI). Refer to Attachment 2, Table 1 for a summary of these details.

Palisades Nuclear Plant has 250 Alloy 600 (Inconel) penetrations, all of which are contained within the primary coolant system (PCS). The PCS includes two identical heat transfer loops connected in parallel to the reactor pressure vessel (RPV). Each loop contains one steam generator, two circulating pumps, flow and temperature instrumentation and connecting piping. A pressurizer is connected to one of the RPV outlets by means of a surge line.

As a result of Alloy 600 cracking issues associated with the pressurizer power-operated relief valve (PORV) nozzle, a project was initiated in 1993 to identify and rank all Alloy 600 penetrations contained within the PCS. This project ranked all 250 Alloy 600 penetrations based on four main criteria: primary water stress corrosion cracking (PWSCC) susceptibility, failure consequence, leakage detection margin and radiation dose rates. The examination requirements for the Alloy 600 penetrations are contained in the Palisades Third Interval Master Inservice Inspection Plan.

Attachment 2, Table 1 contains a description of the 250 Alloy 600 penetrations contained within the PCS and the inspection requirements for the penetrations.

The following sections describe the 250 Alloy 600 penetration configurations, including discussion of inspection of locations where reactor coolant leaks have had the potential to degrade other components, and when applicable, repairs and augmented examination requirements.

Reactor Pressure Vessel Penetrations

The RPV is a vertically mounted cylindrical vessel with a hemispherical lower head welded to the vessel and a removable hemispherical upper closure head. The RPV and top head assembly are constructed of ASTM SA-302 grade B material. The interior surfaces, which are in contact with the reactor coolant, are clad with 3/16-inch minimum type 308/309 austenitic stainless steel.

The RPV contains two Alloy 600 penetrations, which are categorized as follows:

- Two reactor flange leak detector taps.

These penetrations were given the lowest ranking because they are not primary system pressure boundaries due to their location downstream of the inner RPV head o-ring.

Palisades' RPV does not have any penetrations in the bottom head. Additionally, the RPV bottom head receives a remote bare metal visual examination each refueling outage.

Reactor Pressure Vessel Upper Head Penetrations

The RPV upper head has 54 Alloy 600 penetrations, which are categorized as follows:

- Forty-five control rod drive (CRD) Alloy 600 nozzles that are J-welded at the reactor head inner-diameter (ID) and then butt-welded to the CRD flanges above the RPV head.

- Eight incore instrumentation (ICI) nozzles that are J-welded at the RPV upper head ID and then butt-welded to the ICI flanges above the RPV upper head.

- One reactor vent line nozzle that is J-welded at the RPV upper head ID and then butt-welded to the reactor vent line above the RPV upper head.

A bare metal examination of the RPV upper head was performed during the 1995 refueling outage with acceptable results. Additionally, eddy current examination (ECT) of J-welds on the eight ICI penetrations was performed in 1995 with acceptable results.

The RPV and RPV upper 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.

In response to NRC Bulletin 2001-01, "Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles," Nuclear Management Company (NMC) committed to "...perform a 100% effective visual examination of the reactor vessel head upper metal surface, for the Palisades plant, during the next refueling outage." This commitment was restated in the response to NRC Bulletin 2002-01, "Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity."

Pressurizer

The pressurizer maintains PCS operating pressure and compensates for changes in coolant volume during load changes. The pressurizer is constructed of ASTM A-533, grade B, Class 1 steel plate. The interior surface of the cylindrical shell and upper head is clad with type 304 stainless steel. The lower head is clad with a Ni-Cr-Fe alloy to facilitate welding of the Ni-Cr-Fe alloy heater sleeves to the shell.

The pressurizer contains 135 Alloy 600 penetrations, which are categorized as follows:

- One 3-inch ID X 6-inch outer diameter (OD) PORV nozzle located in the upper head, originally fabricated of ASTM-A-508-64 C1 2 forged steel with type 304 stainless steel cladding and fitted with a Ni-Cr-Fe schedule 120 safe-end. In 1993, the safe-end developed a through-wall crack. The safe-end was repaired by removing the cracked weld and heat affected zone, rewelding the stainless steel PORV line to the safe-end, and performing code required examinations of the repair. In 1995, the PORV safe-end was replaced with a new type 316 stainless steel safe-end/spool piece. Alloy 690 was used for the attachment weld to eliminate PCS contact with Alloy 600 from the pressurizer PORV nozzle. Baseline examinations of the repair welds were performed in 1995 using ultrasonic (UT) and liquid penetrant (PT) methods with acceptable results. The safe-end welds are UT/PT examined once each ten-year ISI interval.

- One 4-inch spray line nozzle assembly fabricated of ASTM-A-508-64 C1 2 forged alloy steel with type 304 stainless steel cladding and fitted with a schedule 140, Ni-Cr-Fe alloy safe-end. The elbow to safe-end and safe-end to nozzle welds were examined in 1995 with UT/PT methods with acceptable results. The welds associated with the pressurizer spray line safe-end are UT/PT examined every other refueling outage as required by a previous commitment to the NRC.

- One 12-inch surge line nozzle located in the bottom head of the pressurizer. The surge nozzle is fabricated of ASTM-A-508-64 C1 2 forged alloy steel with type 304 stainless steel cladding and fitted with a schedule 140, Ni-Cr-Fe alloy safe-end. The Mechanical Stress Improvement Process (MSIP) was applied to this weld in 1995. MSIP changed the residual stress patterns at these locations from tensile to compressive by plastically deforming the piping near the welds. The compressive residual stress is desired to help mitigate PWSCC. The safe-end welds were UT/PT examined in 1995 with acceptable results. The safe-end welds associated with the pressurizer surge line safe-end are UT/PT examined every other refueling outage as required by a previous commitment to the NRC.

- Three 3-inch ID X 6-inch OD valve nozzles, all in the upper head, fabricated of ASTM-A-508-64 C1 2 forged alloy steel with type 304 stainless steel cladding and fitted with Ni-Cr-Fe alloy nozzle flanges to provide a 3-inch, 2500# flange connection to the safety valves. Two of the three pressurizer relief valve nozzle to flange welds were examined in 1995 with acceptable results. All three nozzle to safe-end welds are volumetrically examined once each ten year ISI interval as a part of normal ISI inspections.

- Eight 1-inch level nozzles, four upper and four lower, fabricated of ASTM-A-508-64 C1 2 alloy steel forgings, clad and fitted with SA-182 type F-316 alloy ends. The alloy end is to facilitate welding the SA-182 (F-316) stainless steel socket weld safe-ends for connection of the water level instrumentation piping. The level nozzles were not examined in 1995, however they are pressure tested and examined as part of normal ISI inspections.

- Two 1-inch temperature element nozzle penetrations (TE-0101 and TE-0102), one in the upper head and the other in the lower shell, fabricated of Ni-Cr-Fe alloy with SA-182 (F-316) stainless steel socket weld safe-ends. These two penetrations were found to be leaking in 1993. Axial PWSCC, in the heat affected zone of these Alloy 600 nozzles, was determined to be the root cause for this leakage. A pad weld repair was performed in both nozzle locations. A visual VT-2 examination of TE-0101 is performed each refueling outage as required by a previous commitment to the NRC.

- One hundred twenty pressurizer heater penetrations, which are J-welded to the internal Alloy 600 cladding of the vessel lower head. Palisades operates with heaters in full-time service to improve equalization of the PCS loop and pressurizer chemistry, reduce thermal stratification and cycling in the surge and spray lines, and maintain lower pressurizer spray inlet safe-end temperatures. A visual VT-2 examination is performed each refueling outage as required by the ISI program.

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 without the removal of insulation.

Steam Generators

Palisades Nuclear Plant has two steam generators (SGs) that were replaced in 1991. Primary coolant enters the SGs through the inlet nozzle, flows through the ¾-inch OD U-tubes and leaves through two outlet nozzles. Vertical partition plates in the lower head separate the inlet and outlet plenums. The plenums are stainless steel clad, while the primary side of the tube sheet is Ni-Cr-Fe clad. The U-tubes are Alloy 600. The tubes are rolled in the full depth of the tube sheet and the tube-to-tube sheet joint is welded on the primary side. The SG tubes are included in the SG Tube Surveillance Program and are inspected in accordance with Technical Specifications.

Each SG contains two Alloy 600 penetrations, which are the primary bowl plugs. Insulation is not removed to inspect the SG primary bowl plugs due to dose concerns. If a leak develops, it can be identified during the leak test visual inspection by boric acid crystal buildup at the primary bowl insulation seams.

The SGs 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.

Primary Coolant Pumps

Primary coolant is circulated by four vertical, single suction, centrifugal type pumps. The pumps and suction elbows are constructed of a high alloy steel casting (ASTM A-351, GR CF8M) and stainless steel components to minimize corrosion. The pump discharge and the suction elbows are welded to carbon steel transition pieces which are then welded to the ASTM A 516, Grade 70, carbon steel primary coolant piping.

The primary coolant pumps have no Alloy 600 penetrations.

The primary 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.

Primary Coolant Piping

Each primary coolant loop consists of the following: a 42-inch ID hot leg pipe from the RPV outlet to the SG inlet, a 30-inch ID cold leg pipe between the SG outlet and each pump suction nozzle, a 30-inch cold leg pipe between each pump discharge and the RPV inlets, and a 12-inch, schedule 140 surge line pipe between the pressurizer and one hot leg.

The primary coolant piping is of rolled bond clad plate construction, with a base metal of ASTM A-516, grade 70, with a cladding of 304L stainless steel with a nominal thickness of ¼-inch. The 12-inch surge line is type 316 stainless steel. The 12-inch safety injection nozzles on the 30-inch ID cold leg pipes are constructed of carbon steel with stainless steel clad interior. The 12-inch shutdown cooling nozzle on the 42-inch ID hot leg pipe is of the same construction.

The primary coolant piping contains 55 Alloy 600 penetrations, which are categorized as follows:

- Four 12-inch, schedule 140, safety injection and shutdown cooling inlet nozzles, consisting of carbon steel forging, clad with type 304 stainless steel. A SB-166 safe-end is welded to the nozzle forging to permit welding to the connecting piping. These nozzles were not examined in 1995, however they are pressure tested and examined as part of normal ISI inspections.

- One 12-inch, schedule 140, shutdown cooling outlet nozzle, consisting of an alloy steel forging, clad with type 304 stainless steel. A SB-166 safe-end is welded to the nozzle forging to permit welding to the connecting piping. MSIP was applied to this weld in 1995. The compressive residual stress is desired to help mitigate PWSCC. The welds associated with the shutdown cooling outlet safe-end were UT/PT examined in 1995 with acceptable results. These welds are also UT/PT examined every other refueling outage as required by a previous commitment to the NRC.

- One 12-inch, schedule 140 surge nozzle consisting of a carbon steel forging clad with type 304 stainless steel. A SB-166 safe-end is welded to the nozzle forging to permit welding to the connecting piping. MSIP was applied to this weld in 1995. The welds associated with the surge line safe-end were examined in 1995 with acceptable results.

  These welds are additionally UT/PT examined every other refueling outage as required by a previous commitment to the NRC.

- Twenty-two temperature measurement, SB-166 nozzles on the primary loops that are welded on the ID with a J-weld.

- One 2-inch, schedule 160 hot leg drain, SB-166 nozzle. In 1995, the PCS pipe to nozzle weld was PT examined. The nozzle to pipe weld was examined by UT/PT methods. All examinations performed in 1995 had acceptable results.

- Four 2-inch, schedule 160 cold leg drain, SB-166 nozzles. In 1995, PT examinations were performed on the four PCS pipe to nozzle welds and the four nozzle to pipe welds associated with these drains. All examinations performed in 1995 had acceptable results.

- Ten ¾-inch, schedule 160, pressure measurement and sampling nozzles of SB-166. Each nozzle is provided with a stainless steel (SA-182, TP316) socket weld safe-end. In 1995, PT exams were performed on all ten PCS pipe to nozzle welds and UT/PT exams were performed on all ten nozzle to safe-end welds. All examinations performed in 1995 had acceptable results.

- Eight ¾-inch, schedule 160, pressure measurement nozzles of SB-166 with a stainless steel (SA-182, TP 316) socket weld safe-end. The PCS pipe to nozzle and the nozzle to safe-end welds for all eight cold leg pressure measurement penetrations were examined by the liquid penetrant method with acceptable results in 1995.

- Two 3-inch, schedule 160 spray nozzles of SB-166.

- Two 2-inch, schedule 160 charging inlet nozzles of SB-166.

The primary coolant piping, including the penetrations identified above, 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 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

Palisades 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 10CFR 50.55a, personnel performance demonstrations will be through the Electric Power Research Institute / Performance Demonstration Initiative. Qualifications will be through the requirements of ASME B&PV Code, Section XI, 1995 Edition, with 1998 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 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 of ASME B&PV Code, Section XI IWA-2300.
Visual VT-2	Palisades plant operations personnel performing VT-2 examinations are qualified in accordance with plant procedures, which meet the requirements of ASNT-SNT-TC-1A, 1984 edition and 1989 Edition 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 PWSCC 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. NMC considers portions of the Palisades engineered safeguards system (ESS), chemical and volume control system (CVCS), the PCS and spent fuel pool (SFP) 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.

In accordance with Palisades' procedures, new boric acid accumulations shall be documented by a work order request until the boric acid accumulation is removed and necessary repairs are completed.

In accordance with Palisades' corrective action process, 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 (HPSI), low pressure safety injection (LPSI) 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 0.2 gpm (756 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.

Inspections conducted during corrective action planning and implementation activities include an assessment of condition for areas contacted by boric acid. This assessment includes the following:

a. Boric acid accumulation location(s).
b. Boric acid accumulation source(s).
c. If degradation OR corrosion is evident.
d. If boric acid leak is active (wet leakage) or inactive (minor dry residue).
e. If leakage has contacted other components.
f. If the source of boric acid accumulation is due to a component failure other than packing, flange, OR threaded connection leaks (i.e., cracked fittings, welds, components).
g. Actions taken.

Based upon this assessment, and in accordance with Palisades' work planning processes, 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 proximity of the location to radiation sources, such as the PCS. These limitations are considered when planning examinations for specific locations. Due to the proximity of each of the Alloy 600 locations to the PCS, radiation dose is of primary 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 areas is provided in accordance with the safety evaluation (SE), dated June 28, 1996, entitled, "Palisades Plant - Evaluation of the Third 10-Year Inspection Program Plan Requests for Relief NOS PR-02 and PR-04." The SE specifies an alternative to ASME B&PV Code, Section XI, 1989 Edition, which requires VT-2 visual examination at nominal operating pressure and temperature for all portions of the PCS. In the SE, the NRC approved determination of leakage from piping and components under the RPV in accordance with paragraph IWA-5244, "Buried Components," of ASME B&PV Code, Section XI, 1989 Edition, no Addenda. This requirement is satisfied by conducting PCS leak rate calculations in accordance with plant procedures. The detection systems used to conduct PCS leakrate calculations include the containment sump level and RPV flange leak off.

In addition to the leak rate calculation, and as a condition of relief request approval, the NRC invoked the performance of a VT-2 visual examination for evidence of leakage in the RPV cavity, during Mode 5 or 6, once per refueling cycle. The cylindrical portion of the RPV located within the reactor cavity is inaccessible for direct visual examination due to obstructions, heat stress and radiation hazards. Connections located in this area include four cold leg and two hot leg connections. The bottom head of the RPV has no penetrations and is accessible for visual examination. Procedures require that evidence of leakage be documented in accordance with the site's corrective action process and dispositioned prior to unit restart.

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 Palisades' 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 Palisades. 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 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).

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 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 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 joint integrity is justified. 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

This question is not applicable to the Palisades Nuclear Plant, due to the absence of penetrations in the RPV lower head.

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 primary coolant pressure boundary (PCPB) 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 and the containment sump level. These leakage detection systems are specified in plant Technical Specifications.

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

a) Containment sump level - Containment sump water level indication is provided in the main control room by four level indicators that can be used to detect PCS leakage. Two level indicators actuate high water level alarms at 12 inches. Additionally, containment sump level is also monitored (and plotted) continuously by the Plant Process Computer. This method has proven effective in discovering small leakage rates.

b) RPV flange leak off - The inner seal leakage goes to a closed drain line. Leakage is detected by a pressure alarm set at 1,500 psig that will be activated by steam leakage from the RPV flange of approximately 130 in^3. The outer seal liquid leakage is collected and drained to a closed drain line and will be detected by action of a level switch set at 120 inches which will result from a liquid accumulation of approximately 35 in^3.

Water flow changes of 0.5 gpm to 1.0 gpm can readily be detected in contained volumes by monitoring changes in water level, in flow rate, or in the operating frequency of plant equipment.

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

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

Unidentified PCS leakage is maintained less than one gpm per Technical Specifications. One gpm of unidentified leakage from within the PCPB 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 PCS leakage at levels less than the Technical Specification limits. When an increase in PCS 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, conducting leak rate calculations at an increased frequency, generating a corrective action and attempting to locate the source of PCS leakage.

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

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

NMC has developed a comprehensive inspection program to support safe, reliable and cost-effective operation of all 251 Alloy 600 penetration nozzles and safe-ends in the PCS. To support this comprehensive Alloy 600 program an inspection prioritization scheme was developed. This prioritization scheme formed the basis for ranking the 251 Alloy 600 components with respect to PWSCC susceptibility. Four criteria were used to develop the susceptibility rank of each component. A brief explanation of these criteria is provided below.

Susceptibility: All Alloy 600 PWSCC susceptibility contributors are fairly well understood from considerable worldwide studies and field experiences. The susceptibility ranking considers material heat treatment temperatures, carbon content, fabrication process, material yield strength, weld configuration, postweld heat treatment and service temperature as main susceptibility variables. The method yields results comparable to those of other ranking schemes.

Consequence: Postulated PWSCC induced failures in Alloy 600 are bounded by existing large and small break loss-of-coolant accidents. PWSCC induced control rod ejection is bounded by analysis. Failure consequences range from permanent plant shutdown to simple repair during an unforced outage. Contributing factors include core damage potential, plant conditions required for repair, axial or circumferential cracking, component location, difficulty of post-event cleanup prior to restart, and public and regulatory perception.

Detectability: A simplified ranking criteria based on component leak detection margin using the leak-before-break concept was developed for potential circumferential cracking in Alloy 600 girth butt-welded components. Leakage through axial cracks at J-groove welded penetrations can be effectively detected by the ISI Program well before the cracks become critical.

ALARA: Radiation exposure to personnel is generally lower for inspection than repair. However, in-depth inspection of components with low PWSCC susceptibility and low consequence profiles can incur unnecessary exposure in high dose areas.

The table below summarizes the prioritization results. The components are ranked from high to low priority for inspection. On a relative scale, Group I represents the most susceptibility, Group II represents average susceptibility and Group III represents the least susceptibility to PWSCC of all Alloy 600 components at the Palisades Nuclear Plant.

PWSCC INSPECTION PRIORITIZATION INDEX FOR PALISADES	
GROUP	COMPONENT DESCRIPTION
I	Pressurizer PORV Nozzle Safe-End
	Pressurizer and Hot Leg Surge Nozzle Safe-Ends
	Pressurizer Temperature Element (TE) Nozzles
	Primary Relief Valve Mounting Flanges Pressurizer Heater (HE) Sleeves with 44° < Setup Angles < 58°
	Pressurizer Heater (HE) Sleeves with Setup Angles < 44°
	Hot Leg Shutdown Cooling (SDC) Outlet Nozzle Safe-End Pressurizer Level Indicator (LT) Tap Nozzles
II	Pressurizer Spray Safe-End Reactor Head Control Rod Drive Mechanism (CRDM) Nozzles with Setup Angles > 45° Reactor Head Incore Instrument (ICI) Nozzles
	Cold Leg Safety Injection/SDC Inlet Nozzle Safe-Ends Hot Leg Pressure (DPT) and Sampling (SX) Tap Penetrations Reactor Head CRDM Nozzles with 22.5° < Setup Angle < 45° Reactor Head Gas Vent Nozzle
III	Hot Leg Drain Penetration Hot Leg RTD Nozzles Reactor Head CRDM Nozzles with Setup Angles < 22.5°
	Cold Leg RTD Nozzles
	Cold Leg Drain, Charging, Letdown and Spray Penetrations Cold Leg Pressure (DPT) and Sampling (SX) Tap Penetrations Reactor Flange Leak Detector Taps

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

Palisades' reactor vendor is Combustion Engineering (CE). Combustion Engineering Owners Group (CEOG) made the following recommendations based upon CE plant experience with PWSCC in pressurizer heater sleeve nozzles.

1) Inspect pressurizer small diameter Alloy 600 nozzles and heater sleeves during each refueling outage for signs of primary coolant leakage.
2) Inspect with the insulation in place or removed (either approach is acceptable) the presence of boric acid deposits or corrosion products should be assumed to be an indication of leakage until proven otherwise and appropriate actions taken to stop the leakage.
3) Inspect low alloy steels exposed to boric acid and promptly repair primary coolant leaks.

VT-2 visual inspections are performed each refueling outage of the pressurizer bottom head including all 120 heater sleeve nozzles in accordance with the boric acid program. Insulation is not removed for this inspection.

The Westinghouse Owners Group has made no recommendations for CE units concerning inspections for boric acid corrosion of low alloy steel components.

After a review of CEOG recommendations, it is concluded that Palisades Nuclear Plant is in compliance with these CE recommendations for boric acid corrosion of low alloy steel components.

Palisades implements an Industry Experience Review Program to ensure that lessons learned from industry experience (IE) are translated into appropriate action to improve plant safety, reliability and availability. The expectation is that personnel will utilize IE in their daily activities.

Personnel screen and review IE 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 IE.

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 PCS 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 plant Technical Specifications based on the above-noted compliance with ASME B&PV Code, Section XI. Palisades Technical Specifications require that a number of programs be established, implemented, and maintained. Specifically included among these programs, as described in plant Technical Specifications, is the Inservice Testing Program, which provides controls for inservice testing of ASME Code Class 1, 2, and 3 components. Palisades Technical Specifications do not invoke any additional requirements or programs to control PCPB corrosion caused by leakage of boric acid, thus, compliance with ASME B&PV Code, Section XI is all that is required to comply with Technical Specification requirements. There is not a Technical Specification PCPB boric acid corrosion control (BACC) program separate from the Inservice Testing Program.

The Palisades BACC program, particularly with respect to PCPB components other than the RPV head, is based on Generic Letter 88-05, "Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components in PWR Plants," 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:

- Engineering Manual Procedure, EM-09-13, "Inservice Inspection Pressure Testing Program"
  This procedure provides general requirements for the performance of the ASME B&PV Code, Section XI, 1989 Edition and Technical Specification 5.5.7 pressure testing program.

- Engineering Manual Procedure EM-09-14, "VT-2 Examinations"
  This procedure defines the requirements, responsibilities and techniques for performance of VT-2 examinations on ASME Class 1, 2 and 3 systems and components.

- Engineering Manual Procedure EM-09-03, "Inservice Inspection"
  This procedure provides general requirements for the performance of the ASME B&PV Code, Section XI, 1989 Edition and Technical Specification 5.5.7, Inservice Inspection Program for ASME Class 1, 2 and 3 systems and components.

- Engineering Manual Procedure EM-26, "Boric Acid Leak Inspection Program"
  This procedure describes the boric acid leak inspection program at Palisades. 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 described in EM-09-13 by providing suitable methods for dispositioning boric acid leaks discovered during pressure testing.

- Administrative Procedure 5.01, "Work Requests/ Work Orders"
  The purpose of this procedure is to identify the responsibilities and administrative controls required during the initiation, planning, performance and completion of maintenance authorized via a work request/work order at the Palisades Nuclear Plant.

- Administrative Procedure 3.03, "Corrective Action Process"
  The purpose of the corrective action process is to assure timely identification of problems, appropriate evaluation of problems, timely and effective implementation of corrective actions and review of corrective action effectiveness.

ATTACHMENT 2

NUCLEAR MANAGEMENT COMPANY, LLC
PALISADES NUCLEAR PLANT
DOCKET 50-255

JANUARY 20, 2003

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

10 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 Incore Instrument Penetrations ICI-1 thru ICI-8	Visual	VT-2	Insulated VT-2	Each Refueling Outage	Insulation Not Removed / Nukon Blanket with Reflective Metal Cover	RT-71A Primary Coolant System, Class 1 System Leakage Test
Reactor Vessel Head Incore Instrument Penetrations ICI-1 thru ICI-8	Visual	VT-2	100% Bare Metal	2003 Refueling Outage	100% Removed / Nukon Blanket with Reflective Metal Cover	RT-71A
Reactor Vessel Head Control Rod Penetrations CRD-01 thru CRD-45	Visual	VT-2	Insulated VT-2	Each Refueling Outage	Insulation Not Removed / Nukon Blanket with Reflective Metal Cover	RT-71A
Reactor Vessel Head Control Rod Penetrations CRD-01 thru CRD-45	Visual	VT-2	100% Bare Metal	2003 Refueling Outage	100% Removed / Nukon Blanket with Reflective Metal Cover	RT-71A
Reactor Vessel Head Vent	Visual	VT-2	Insulated VT-2	Each Refueling Outage	Insulation Not Removed / Nukon Blanket with Reflective Metal Cover	RT-71A
Reactor Vessel Head Vent	Visual	VT-2	100% Bare Metal	2003 Refueling Outage	100% Removed / Nukon Blanket with Reflective Metal Cover	RT-71A
Reactor Vessel Outer O-Ring Leak Off	Visual	VT-2	VT-2	Each Refueling Outage	Non-Insulated	RT-71A
Reactor Vessel Inner O-Ring Leak Off	Visual	VT-2	VT-2	Each Refueling Outage	Non-Insulated	RT-71A
Pressurizer Upper Temperature Element TE-0101(repaired)	Visual	VT-2	Insulated VT-2	Each Refueling Outage	Insulation Not Removed / Nukon Blanket with Reflective Metal Cover	RT-71A
Pressurizer Temperature Element Nozzle TE-0102(repaired)	Visual	VT-2	Insulated VT-2	Each Refueling Outage	Insulation Not Removed / Nukon Blanket with Reflective Metal Cover	RT-71A
Pressurizer PORV Nozzle PCS-4-PRS-1P1 (Alloy 690)	Visual	VT-2	Insulated VT-2	Each Refueling Outage	Insulation Not Removed / Nukon Blanket with Reflective Metal Cover	RT-71A
Pressurizer PORV Nozzle PCS-4-PRS-1P1 (Alloy 690)	Volumetric and Surface	UT/PT	100%	Once every 10 years	100% Removed / Nukon Blanket with Reflective Metal Cover	EM-09-03 Inservice Inspection

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
Pressurizer Spray Nozzle PCS-4-PSS-1P1	Visual	VT-2	Insulated VT-2	Each Refueling Outage	Insulation Not Removed / Nukon Blanket with Reflective Metal Cover	RT-71A
Pressurizer Spray Nozzle PCS-4-PSS-1P1	Volumetric and Surface	UT/PT	100%	Every Other Refueling Outage	100% Removed / Nukon Blanket with Reflective Metal Cover	EM-09-03
Pressurizer Surge Line PCS-12-PSL-1H1	Visual	VT-2	Insulated VT-2	Each Refueling Outage	Insulation Not Removed / Nukon Blanket with Reflective Metal Cover	RT-71A
Pressurizer Surge Line PCS-12-PSL-1H1	Volumetric and Surface	UT/PT	100%	Every Other Refueling Outage	100% Removed / Nukon Blanket with Reflective Metal Cover	EM-09-03
Pressurizer Relief Valve RV-1039	Visual	VT-2	Insulated VT-2	Each Refueling Outage	Insulation Not Removed / Nukon Blanket with Reflective Metal Cover	RT-71A
Pressurizer Relief Valve RV-1039	Volumetric	UT or RT	100%	Once every 10 years	Insulation Not Removed / Nukon Blanket with Reflective Metal Cover	EM-09-03
Pressurizer Relief Valve RV-1040	Visual	VT-2	Insulated VT-2	Each Refueling Outage	Insulation Not Removed / Nukon Blanket with Reflective Metal Cover	RT-71A
Pressurizer Relief Valve RV-1040	Volumetric	UT or RT	100%	Once every 10 years	Insulation Removed / Reflective Metal	EM-09-03
Pressurizer Relief Valve RV-1041	Visual	VT-2	Insulated VT-2	Each Refueling Outage	Insulation Not Removed / Reflective Metal	RT-71A
Pressurizer Relief Valve RV-1041	Volumetric	UT or RT	100%	Once every 10 years	Insulation Removed / Reflective Metal	EM-09-03
Pressurizer Level Instrument Nozzle LT-0103	Visual	VT-2	Insulated VT-2	Each Refueling Outage	Insulation Not Removed / Reflective Metal	RT-71A
Pressurizer Level Instrument Nozzle LT-0102	Visual	VT-2	Insulated VT-2	Each Refueling Outage	Insulation Not Removed / Reflective Metal	RT-71A
Pressurizer Level Instrument Nozzle LT-0101B	Visual	VT-2	Insulated VT-2	Each Refueling Outage	Insulation Not Removed / Reflective Metal	RT-71A
Pressurizer Level Instrument Nozzle LT-0101A	Visual	VT-2	Insulated VT-2	Each Refueling Outage	Insulation Not Removed / Reflective Metal	RT-71A
Pressurizer Level Instrument Nozzle LT-0103	Visual	VT-2	Insulated VT-2	Each Refueling Outage	Insulation Not Removed / Reflective Metal	RT-71A
Pressurizel Level Instrument Nozzle LT-0102	Visual	VT-2	Insulated VT-2	Each Refueling Outage	Insulation Not Removed / Reflective Metal	RT-71A

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
Pressurizer Level Instrument Nozzle LT-0101B	Visual	VT-2	Insulated VT-2	Each Refueling Outage	Insulation Not Removed / Reflective Metal	RT-71A
Pressurizel level Instrument Nozzle LT-0101A	Visual	VT-2	Insulated VT-2	Each Refueling Outage	Insulation Not Removed / Reflective Metal	RT-71A
120 Pressurizer Heaters	Visual	VT-2	Insulated VT-2	Each Refueling Outage	Insulation Not Removed / Removable Reflective Insulation	RT-71A
PCS Hot Leg Pressure Instrument Tap DPT-0112A	Visual	VT-2	Insulated VT-2	Each Refueling Outage	Insulation Not Removed / Removable Reflective Insulation	RT-71A
PCS Hot Leg Pressure Instrument Tap DPT-0112B	Visual	VT-2	Insulated VT-2	Each Refueling Outage	Insulation Not Removed / Removable Reflective Insulation	RT-71A
PCS Hot Leg Pressure Instrument Tap DPT-0112C	Visual	VT-2	Insulated VT-2	Each Refueling Outage	Insulation Not Removed / Removable Reflective Insulation	RT-71A
PCS Hot Leg Pressure Instrument Tap DPT-0112D	Visual	VT-2	Insulated VT-2	Each Refueling Outage	Insulation Not Removed / Removable Reflective Insulation	RT-71A
PCS Hot Leg Pressure Instrument Tap DPT-0122A	Visual	VT-2	Insulated VT-2	Each Refueling Outage	Insulation Not Removed / Removable Reflective Insulation	RT-71A
PCS Hot Leg Pressure Instrument Tap DPT-0122B	Visual	VT-2	Insulated VT-2	Each Refueling Outage	Insulation Not Removed / Removable Reflective Insulation	RT-71A
PCS Hot Leg Pressure Instrument Tap DPT-0122C	Visual	VT-2	Insulated VT-2	Each Refueling Outage	Insulation Not Removed / Removable Reflective Insulation	RT-71A
PCS Hot Leg Pressure Instrument Tap DPT-0122D	Visual	VT-2	Insulated VT-2	Each Refueling Outage	Insulation Not Removed / Removable Reflective Insulation	RT-71A
PCS Hot Leg Sampling Tap SX-1023A	Visual	VT-2	Insulated VT-2	Each Refueling Outage	Insulation Not Removed / Removable Reflective Insulation	RT-71A
PCS Hot Leg Sampling Tap SX-1012	Visual	VT-2	Insulated VT-2	Each Refueling Outage	Insulation Not Removed / Removable Reflective Insulation	RT-71A
PCS Hot Leg Surge Nozzle PCS-12-PSL-1H1	Visual	VT-2	Insulated VT-2	Each Refueling Outage	Insulation Not Removed / Nukon Blanket with Reflective Metal Cover	RT-71A

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
PCS Hot Leg Surge Nozzle PCS-12-PSL-1H1	Volumetric and Surface	UT/PT	100%	Every Other Refueling Outage	100% Removed / Nukon Blanket with Reflective Metal Cover	EM-09-03
PCS Shutdown Cooling Outlet Nozzle PCS-12-SCS-2H1	Visual	VT-2	Insulated VT-2	Each Refueling Outage	Insulation Not Removed / Nukon Blanket with Reflective Metal Cover	RT-71A
PCS Shutdown Cooling Outlet Nozzle PCS-12-SCS-2H1	Volumetric and Surface	UT/PT	100%	Every Other Refueling Outage	100% Removed / Nukon Blanket with Reflective Metal Cover	EM-09-03
PCS Hot Leg Drain PCS-2-PSS-DRL-1H1	Visual	VT-2	Insulated VT-2	Each Refueling Outage	Insulation Not Removed / Nukon Blanket with Reflective Metal Cover	RT-71A
PCS Hot Leg RTD TE-0111H	Visual	VT-2	Insulated VT-2	Each Refueling Outage	Insulation Not Removed / Removable Donut Around Penetration	RT-71A
PCS Hot Leg RTD TE-0112HA	Visual	VT-2	Insulated VT-2	Each Refueling Outage	Insulation Not Removed / Removable Donut Around Penetration	RT-71A
PCS Hot Leg RTD TE-0112HB	Visual	VT-2	Insulated VT-2	Each Refueling Outage	Insulation Not Removed / Removable Donut Around Penetration	RT-71A
PCS Hot Leg RTD TE-0112HC	Visual	VT-2	Insulated VT-2	Each Refueling Outage	Insulation Not Removed / Removable Donut Around Penetration	RT-71A
PCS Hot Leg RTD TE-0112HD	Visual	VT-2	Insulated VT-2	Each Refueling Outage	Insulation Not Removed / Removable Donut Around Penetration	RT-71A
PCS Hot Leg RTD TE-0121H	Visual	VT-2	Insulated VT-2	Each Refueling Outage	Insulation Not Removed / Removable Donut Around Penetration	RT-71A
PCS Hot Leg RTD TE-0122HA	Visual	VT-2	Insulated VT-2	Each Refueling Outage	Insulation Not Removed / Removable Donut Around Penetration	RT-71A
PCS Hot Leg RTD TE-0122HB	Visual	VT-2	Insulated VT-2	Each Refueling Outage	Insulation Not Removed / Removable Donut Around Penetration	RT-71A
PCS Hot Leg RTD TE-0122HC	Visual	VT-2	Insulated VT-2	Each Refueling Outage	Insulation Not Removed / Removable Donut Around Penetration	RT-71A
PCS Hot Leg RTD TE-0122HD	Visual	VT-2	Insulated VT-2	Each Refueling Outage	Insulation Not Removed / Removable Donut Around Penetration	RT-71A

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
PCS Cold Leg Charging Nozzle CVC-2-CHL-1A1	Visual	VT-2	Insulated VT-2	Each Refueling Outage	Insulation Not Removed / Nukon Blanket with Metal Cover	RT-71A
PCS Cold Leg Charging Nozzle PCS-2-CHL-2A1	Visual	VT-2	Insulated VT-2	Each Refueling Outage	Insulation Not Removed / Nukon Blanket with Metal Cover	RT-71A
PCS Cold Leg Spray Nozzle PCS-3-PSS-1B1	Visual	VT-2	Insulated VT-2	Each Refueling Outage	Insulation Not Removed / Nukon Blanket with Metal Cover	RT-71A
PCS Cold Leg Spray Nozzle PCS-3-PSS-2A1	Visual	VT-2	Insulated VT-2	Each Refueling Outage	Insulation Not Removed / Nukon Blanket with Metal Cover	RT-71A
PCS Cold Leg Pressure Instrument Tap DPT-0112A/C Pump 1A Suction	Visual	VT-2	Insulated VT-2	Each Refueling Outage	Insulation Not Removed / Removable Reflective Metal Insulation	RT-71A
PCS Cold Leg Pressure Instrument Tap DPT-0112A/C Pump 1B Suction	Visual	VT-2	Insulated VT-2	Each Refueling Outage	Insulation Not Removed / Removable Reflective Metal Insulation	RT-71A
PCS Cold Leg Pressure Instrument Tap DPT-0112B/D Pump 1A Suction	Visual	VT-2	Insulated VT-2	Each Refueling Outage	Insulation Not Removed / Removable Reflective Metal Insulation	RT-71A
PCS Cold Leg Pressure Instrument Tap DPT-0112B/D Pump 1B Suction	Visual	VT-2	Insulated VT-2	Each Refueling Outage	Insulation Not Removed / Removable Reflective Metal Insulation	RT-71A
PCS Cold Leg Pressure Instrument Tap DPT-0122A/C Pump 2A Suction	Visual	VT-2	Insulated VT-2	Each Refueling Outage	Insulation Not Removed / Removable Reflective Metal Insulation	RT-71A
PCS Cold Leg Pressure Instrument Tap DPT-0122A/C Pump 2B Suction	Visual	VT-2	Insulated VT-2	Each Refueling Outage	Insulation Not Removed / Removable Reflective Metal Insulation	RT-71A
PCS Cold Leg Pressure Instrument Tap DPT-0122B/D Pump 2A Suction	Visual	VT-2	Insulated VT-2	Each Refueling Outage	Insulation Not Removed / Removable Reflective Metal Insulation	RT-71A
PCS Cold Leg Pressure Instrument Tap DPT-0122B/D Pump 2B Suction	Visual	VT-2	Insulated VT-2	Each Refueling Outage	Insulation Not Removed / Removable Reflective Metal Insulation	RT-71A
PCS Loop Drain Nozzle PCS-2-DRL-1A1 Pump 1A Suction	Visual	VT-2	Insulated VT-2	Each Refueling Outage	Insulation Not Removed / Removable Reflective Metal Insulation	RT-71A

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
PCS Loop Drain Nozzle PCS-2-DRL-1B1Pump 1B Suction	Visual	VT-2	Insulated VT-2	Each Refueling Outage	Insulation Not Removed / Removable Reflective Metal Insulation	RT-71A
PCS Loop Drain Nozzle PCS-2-DRL-2A1Pump 2A Suction	Visual	VT-2	Insulated VT-2	Each Refueling Outage	Insulation Not Removed / Removable Reflective Metal Insulation	RT-71A
PCS Loop Drain Nozzle PCS-2-DRL-2B1Pump 2B Suction	Visual	VT-2	Insulated VT-2	Each Refueling Outage	Insulation Not Removed / Removable Reflective Metal Insulation	RT-71A
PCS Safety Injection Nozzle PCS-12-SIS-1A1 Pump 1A Discharge	Visual	VT-2	Insulated VT-2	Each Refueling Outage	Insulation Not Removed / Nukon Blanket with Metal Cover	RT-71A
PCS Safety Injection Nozzle PCS-12-SIS-1B1 Pump 1B Discharge	Visual	VT-2	Insulated VT-2	Each Refueling Outage	Insulation Not Removed / Nukon Blanket with Metal Cover	RT-71A
PCS Safety Injection Nozzle PCS-12-SIS-2A1Pump 2A Discharge	Visual	VT-2	Insulated VT-2	Each Refueling Outage	Insulation Not Removed / Nukon Blanket with Metal Cover	RT-71A
12" Shutdown Cooling Line PCS-12-SIS-2B1 Pump 2B Discharge	Visual	VT-2	Insulated VT-2	Each Refueling Outage	Insulation Not Removed / Nukon Blanket with Metal Cover	RT-71A
PSC Cold Leg RTD TE-0111A Pump 1A Discharge	Visual	VT-2	Insulated VT-2	Each Refueling Outage	Insulation Not Removed / Donut Insulation Around Penetration	RT-71A
PSC Cold Leg RTD TE-0111B Pump 1B Discharge	Visual	VT-2	Insulated VT-2	Each Refueling Outage	Insulation Not Removed / Donut Insulation Around Penetration	RT-71A
PSC Cold Leg RTD TE-0112CA Pump 1A Discharge	Visual	VT-2	Insulated VT-2	Each Refueling Outage	Insulation Not Removed / Donut Insulation Around Penetration	RT-71A
PSC Cold Leg RTD TE-0112CB Pump 1B Discharge	Visual	VT-2	Insulated VT-2	Each Refueling Outage	Insulation Not Removed / Donut Insulation Around Penetration	RT-71A
PSC Cold Leg RTD TE-0112CC Pump 1A Discharge	Visual	VT-2	Insulated VT-2	Each Refueling Outage	Insulation Not Removed / Donut Insulation Around Penetration	RT-71A
PSC Cold Leg RTD TE-0112CD Pump 1B Discharge	Visual	VT-2	Insulated VT-2	Each Refueling Outage	Insulation Not Removed / Donut Insulation Around Penetration	RT-71A

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
PSC Cold Leg RTD TE-0121A Pump 2A Discharge	Visual	VT-2	Insulated VT-2	Each Refueling Outage	Insulation Not Removed / Donut Insulation Around Penetration	RT-71A
PSC Cold Leg RTD TE-0121B Pump 2B Discharge	Visual	VT-2	Insulated VT-2	Each Refueling Outage	Insulation Not Removed / Donut Insulation Around Penetration	RT-71A
PSC Cold Leg RTD TE-0122CA Pump 2A Discharge	Visual	VT-2	Insulated VT-2	Each Refueling Outage	Insulation Not Removed / Donut Insulation Around Penetration	RT-71A
PSC Cold Leg RTD TE-0122CB Pump 2B Discharge	Visual	VT-2	Insulated VT-2	Each Refueling Outage	Insulation Not Removed / Donut Insulation Around Penetration	RT-71A
PSC Cold Leg RTD TE-0122CC Pump 2A Discharge	Visual	VT-2	Insulated VT-2	Each Refueling Outage	Insulation Not Removed / Donut Insulation Around Penetration	RT-71A
PSC Cold Leg RTD TE-0122CD Pump 2B Discharge	Visual	VT-2	Insulated VT-2	Each Refueling Outage	Insulation Not Removed / Donut Insulation Around Penetration	RT-71A
"A" Steam Generator Bowl Plug	Visual	VT-2	Insulated VT-2	Each Refueling Outage	Insulation not Removed/ Nukon Blanker with Reflective Metal CoverMirror Insulation	RT-71A
"A" Steam Generator Bowl Plug	Visual	VT-2	Insulated VT-2	Each Refueling Outage	Insulation not Removed/ Nukon Blanker with Reflective Metal CoverMirror Insulation	RT-71A
"B" Steam Generator Bowl Plug	Visual	VT-2	Insulated VT-2	Each Refueling Outage	Insulation not Removed/ Nukon Blanker with Reflective Metal CoverMirror Insulation	RT-71A
"B" Steam Generator Bowl Plug	Visual	VT-2	Insulated VT-2	Each Refueling Outage	Insulation not Removed/ Nukon Blanker with Reflective Metal CoverMirror Insulation	RT-71A

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
Bolted Connection Inspections IWA-5242						
CV-1059 Pressurizer Spray Valve from Loop 2A	Visual	VT-2	100%	Each Refueling Outage	Insulation Removed/Blanket Insulation	RT-71A-1 Primary Coolant System, Class 1 Examination of Bolted Connections
CK-ES3131 PCS Loop Check Valve (Loop 2A)	Visual	VT-2	100%	Each Refueling Outage	Insulation Removed/Blanket Insulation	RT-71A-1
CK-ES3132 Safety Injection Tank T-82C Check Valve	Visual	VT-2	100%	Each Refueling Outage	Insulation Removed/Blanket Insulation	RT-71A-1
CV-2001 Regenerative Heat Exchanger E-56 Letdown Stop Valve	Visual	VT-2	100%	Each Refueling Outage	Insulation Removed/Blanket Insulation	RT-71A-1
CK-ES3146 PCS Loop Check Valve (Loop 2B)	Visual	VT-2	100%	Each Refueling Outage	Insulation Removed/Blanket Insulation	RT-71A-1
CK-ES3147 Safety Injection Tank T-82D Check Valve	Visual	VT-2	100%	Each Refueling Outage	Insulation Removed/Blanket Insulation	RT-71A-1
CV-2002 Letdown Orifice Bypass Valve	Visual	VT-2	100%	Each Refueling Outage	Insulation Removed/Blanket Insulation	RT-71A-1
CK-ES3101 PCS Loop Check Valve (Loop 1A)	Visual	VT-2	100%	Each Refueling Outage	Insulation Removed/Blanket Insulation	RT-71A-1
CK-ES3102 Safety Injection Tank T-82A Check Valve	Visual	VT-2	100%	Each Refueling Outage	Insulation Removed/Blanket Insulation	RT-71A-1
CK-ES3116 PCS Loop Check Valve (Loop 1B)	Visual	VT-2	100%	Each Refueling Outage	Insulation Removed/Blanket Insulation	RT-71A-1
CK-ES3117 Safety Injection Tank T-82B Check Valve	Visual	VT-2	100%	Each Refueling Outage	Insulation Removed/Blanket Insulation	RT-71A-1
Primary Coolant Pumps P-50A, P-50B, P-50C, P-50D Casing Flanges (4)	Visual	VT-2	100%	Each Refueling Outage	Insulation Removed/Nukon Blanket with Metal Cover	RT-71A-1
Steam Generator E-50A and E-50B Primary Manways (2)	Visual	VT-2	100%	Each Refueling Outage	Insulation Removed/Nukon Blanket with Metal Cover	RT-71A-1
Pressurizer T-72 Manway	Visual	VT-2	100%	Each Refueling Outage	Insulation Removed/Nukon Blanket with Metal Cover	RT-71A-1
Reactor Vessel N-50 Flange	Visual	VT-2	100%	Each Refueling Outage	Insulation Removed/Reflective Metal	RT-71A-1
Pressurizer Relief Valves (3)	Visual	VT-2	100%	Each Refueling Outage	Insulation Removed/Blanket Insulation	RT-71A-1
Reactor Head Vent Restricting Orifice RO-0101	Visual	VT-2	100%	Each Refueling Outage	Insulation Removed	RT-71A-1
Reactor Head Vent Spool Piece RSP-0101	Visual	VT-2	100%	Each Refueling Outage	Insulation Removed	RT-71A-1

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
Inner Seal Leakoff Test Connection	Visual	VT-2	100%	Each Refueling Outage	Insulation Removed	RT-71A-1

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	RT-71A-2 Primary Coolant System, Class 1 Reactor Vessel Visual Examination
Pressurizer	Visual	VT-2	Insulated VT-2	Each Refueling Outage	Insulation not Removed/Mirror Insulation	RT-71A
Steam Generator A	Visual	VT-2	Insulated VT-2	Each Refueling Outage	Insulation not Removed/Nukon Blanket with Metal Cover	RT-71A
Steam Generator B	Visual	VT-2	Insulated VT-2	Each Refueling Outage	Insulation not Removed/Nukon Blanket with Metal Cover	RT-71A