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Kewaunee Nuclear Power Plant
Operated by Nuclear Management Company, LLC

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KEWAUNEE 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 KEWAUNEE NUCLEAR PLANT, REQUEST FOR ADDITIONAL INFORMATION (TAC NO. MB4552)

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 14, 2002, the NRC issued a request for additional information (RAI) concerning the 60 day response to BL 2002-01. The NRC requested that a response be provided within 60 days of receipt of the RAI. NMC is providing the attached RAI response for the Kewaunee 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.

Thomas Coutu

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Attachments

A095

ATTACHMENT 1

NUCLEAR MANAGEMENT COMPANY, LLC
KEWAUNEE NUCLEAR PLANT
DOCKET 50-305

JANUARY 20, 2003

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

18 Pages Follow

Requested Item

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

Response

The technical basis for the inspection techniques, scope and extent of coverage, frequency of inspections, personnel qualifications and degree of insulation removal are as required to fulfill the requirements of American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code, Section XI, 1989 Edition as modified by the Nuclear Regulatory Commission (NRC) commitments. Refer to Attachment 2, Table 1 for a summary of these details.

Kewaunee Nuclear Plant has 79 Alloy 600 (Inconel) penetrations all of which are contained within the reactor coolant system (RCS) on the reactor pressure vessel (RPV). The RCS includes two identical heat transfer loops connected in parallel to the RPV. Each loop contains one steam generator, one reactor coolant pump, flow and temperature instrumentation and connecting piping. A pressurizer is connected to one of the RPV outlets by means of a surge line.

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

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

Reactor Pressure Vessel Penetrations

The reactor pressure vessel (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 upper head flange are constructed of SA508-64 Class 2 carbon steel. The RPV upper head dome and bottom head are constructed of SA533 Grade B Class 1 carbon steel. The interior surfaces, which are in contact with the reactor coolant, are clad with a minimum 0.156 inches of 304 stainless steel or equivalent and Inconel.

The reactor pressure vessel (RPV) contains thirty-eight Alloy 600 penetrations, which are categorized as follows:

- Two 4-inch safe-end welds (which also contain SA-182 material on the nozzle ends) to the RPV shell.
- Thirty-six instrument tube penetrations to the RPV bottom head.

Reactor Pressure Vessel Upper Head Penetrations

The RPV upper head contains forty-one Alloy 600 penetrations, which are categorized as follows:

- Forty RPV closure head penetrations for control rod drive mechanisms and core exit thermocouples.
- One three-quarter-inch vent line to the RPV closure head.

A bare metal examination by VT-1 Level II and Level III personnel of the RPV closure head exterior was performed during the 2001 refueling outage with acceptable results.

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 visual inspection is performed by VT-2 Level II or Level III personnel without the removal of insulation.

In addition to the inspection requirements of ASME B&PV Code, Section XI, NMC has committed to perform a bare metal examination of the Kewaunee RPV closure head exterior during the 2003 refueling outage in response to NRC Bulletin 2002-02, "Reactor Pressure Vessel Head And Vessel Head Penetration Nozzle Inspection Programs."

The balance of safe-end welds connecting the reactor coolant piping to the RPV do not contain Alloy 600 material. The pressurizer and accompanying safe-end welds do not contain Alloy 600 material. The Alloy 600 safe-end welds for the replacement steam generators were fabricated with Alloy 690 cladding in contact with the reactor coolant fluid and are less susceptible to primary water stress corrosion cracking (PWSCC).

Pressurizer

The pressurizer maintains PCS operating pressure and compensates for changes in coolant volume during load changes. The pressurizer is constructed of SA-533, Grade A, Class 1 carbon steel. The internal surfaces are clad with austenitic stainless steel.

The pressurizer contains five safe-end penetrations, which are characterized as follows:

- One 6-inch pressure relief nozzle safe-end manufactured with a nozzle end of SA-182 TP 316 stainless steel.
- Two 6-inch pressurizer safety nozzle safe-ends manufactured with nozzle ends of SA-182 TP 316 stainless steel.
- One 4-inch pressurizer spray nozzle safe-end manufactured with a nozzle end of SA-182 TP 316 stainless steel.
- One 14-inch pressurizer surge nozzle safe-end manufactured with a nozzle end of SA-182 TP 316 stainless steel.

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 visual inspection is performed without the removal of insulation.

Steam Generators

Kewaunee Nuclear Plant has two steam generators (SGs) that were replaced in 2001. Each SG contains two nozzle to safe-end Alloy 600 welds with Alloy 690 cladding in contact with the reactor coolant fluid and are not susceptible to PWSCC. These are the loop A 31-inch ID outlet and 29-inch ID inlet, and the loop B 31-inch ID outlet and 29-inch ID inlet.

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 visual inspection is performed without the removal of insulation.

Reactor Coolant Pumps

Reactor coolant is circulated by two vertical, single suction, centrifugal type pumps. The pumps are constructed of ASTM A-351, GR CF8M.

The reactor coolant pumps have no Alloy 600 penetrations.

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

Reactor Coolant Piping

Each reactor coolant loop consists of the following: a 29-inch ID hot leg pipe from the RPV outlet to the SG inlet, a 31-inch ID intermediate leg pipe between the SG outlet and each reactor coolant pump suction nozzle, a 27.5-inch cold leg pipe between each reactor coolant pump discharge and the RPV inlets, and a 10-inch, schedule 140 surge line pipe between the pressurizer and one hot leg.

The reactor coolant piping has no Alloy 600 penetrations.

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

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

Personnel Qualifications

Kewaunee Nuclear 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 of ASME B&PV Code Section XI IWA-2300.
Visual (VT-1, VT-2, VT-3)	Nondestructive 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.

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 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 engineered safeguards systems (ESS), chemical and volume control system (CVCS) and the RCS, 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, 1989 Edition for system pressure tests. During regularly scheduled inservice inspection activities, insulation is removed as necessary to complete the specified inspection or examination technique.

These specified inspection or examination techniques are documented in a Relief Request for ASME B&PV Code, Section XI, Section IWA-5242 and Attachment 2, Table 1 for Kewaunee Generic Letter (GL) 88-05 Program.

In accordance with Kewaunee's corrective action process, an action request (AR) shall be initiated upon discovery of equipment malfunction, damage, or degradation that is sudden or unexpected. The following indications shall be considered recordable, and evaluated to acceptance criteria described below (evaluation may include removal of insulation to determine extent of condition):

1. All leakage or seepage.
2. Heavy boric acid (as determined by the VT-2 examiner) on component, including location of the leakage source and the areas of general corrosion, if any.
3. Light or moderate boric acid (as determined by the VT-2 examiner) on component that may mask (in the opinion of the VT-2 examiner) leakage or seepage, including location of the leakage source and the areas of general corrosion, if any.
4. Components with local areas of general corrosion that reduce the wall thickness by more than 10% (as determined by the VT-2 examiner). These areas shall be evaluated by Kewaunee Nuclear Plant personnel to determine whether the component may be acceptable for continued service or repair or replacement is required.

The results of the Class 1 (RPV, pressurizer, steam generators, heat exchangers, piping, pumps, and valves) system pressure test are acceptable if:

1. No reactor coolant pressure boundary leakage or seepage (other than normally controlled leakage or seepage) are detected.
2. Components with local areas of general corrosion that reduced the wall thickness by more than 10% have been repaired or replaced and accepted for continued use, or evaluated as being acceptable for continued use.

Recordable indications noted without corrective action may be accepted for continued use provided that the Kewaunee Nuclear Plant operations department accepts the recorded leakage or seepage as controllable and work orders have been initiated to correct leakage or seepage. As stated above, this would not apply reactor coolant pressure boundary leakage or seepage.

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 RCS. These limitations are considered when planning examinations for specific locations. Due to the proximity of each of the Alloy 600 locations to the RCS, 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 contained in ASME B&PV Code, Section XI. A leak test is performed on the primary system each refueling outage, as required by ASME B&PV Code, Section XI, Category B-P. This leak test is performed after reaching normal operating pressure during startup from the refueling outage.

The leak test procedure includes entry into the refueling cavity for a VT-2 examination of the primary pressure boundary for the upper area of the RPV, control rod drive mechanisms (CRDMs), head vent system and conoseals. Four inspection ports in the RPV upper shroud are removed for limited direct visual inspection of portions of the CRDMs. The leak test also includes entry into containment sump C for VT-2 examination of the RPV lower head and instrument penetrations. The mirror insulation on the lower head is not removable as insulation was installed prior to installing the thimble tubes.

The RPV hot and cold leg nozzles and the safety injection nozzles attached to the vertical shell of the RPV are below the cavity and are not readily accessible for direct visual examination. Limited access is available through the sand box covers, which are removed at a 10-year periodicity for surface examination of the nozzle welds. This surface examination was completed most recently for five nozzles in 1995 and for one nozzle in 1988. The vertical surfaces of the insulated RPV are examined from containment sump C in accordance with ASME B&PV Code, Section XI, IWA-5242.

In accordance with Kewaunee Technical Specifications, the RCS is monitored by the weekly determination of RCS leakage per directions contained in site operating procedures. Site procedures use indicators, such as charging versus letdown and containment sump level change, to monitor for RCS leakage.

In accordance with Kewaunee Technical Specifications, visual examination, VT-2, is performed to detect leakage or evidence of leakage with the RCS at operating pressure and temperature each refueling outage.

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

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

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

The following indications (which apply to bolted connections) shall be considered recordable, and evaluated to acceptance criteria described below.

1. All leakage or seepage.
2. Heavy boric acid (as determined by the VT-2 examiner) on component, including location of the leakage source and the areas of general corrosion, if any.
3. Light or moderate boric acid (as determined by the VT-2 examiner) on component that may mask (in the opinion of the VT-2 examiner) leakage or seepage, including location of the leakage source and the areas of general corrosion, if any.
4. Components with local areas of general corrosion that reduce the wall thickness by more than 10% (as determined by the VT-2 examiner). These areas shall be evaluated by Kewaunee Nuclear Plant engineering personnel to determine whether the component may be acceptable for continued service or repair or replacement is required.

The results of the Class 1 (RPV, pressurizer, steam generators, heat exchangers, piping, pumps, and valves) system pressure test are acceptable if:

1. No reactor coolant pressure boundary leakage or seepage (other than normally controlled leakage or seepage) is detected.
2. Upon discovery of leakage from a mechanical joint, the first objective is to repair the component. In accordance with ASME Boiler and Pressure Vessel (B&PV) Code, Section XI, subparagraph IWA-5250(a)(2), the bolting shall be removed, and VT-3 visually examined for corrosion and evaluated in accordance with IWA-3100. This condition shall be dispositioned per the requirements of the Kewaunee Nuclear Power Plant 3rd Ten Year Inservice Inspection (ISI) Program 1994-2004 Relief Request No. RR-G-1 Rev. 1:
   a. Corrective measure shall be specified under Paragraph IWA-5250 of ASME B&PV Code, Section XI, 1980 Edition, up to and including Winter 1981 Addenda and including repair of the bolted connection at the next scheduled Refueling Outage if the joint can be shown to be acceptable for continued service for the next operation cycle.
   b. The following list includes those activities that may be performed to assess the integrity of the bolted connection without removal of the bolting to determine whether or not the joint can perform its function until the next scheduled refueling outage:
      - Ultrasonic Examination
      - Radiography
      - Observation and analysis of corrosion products
      - Assessment of affected area of joint
      - Analysis of number of fasteners needed to maintain closure
      - Consideration of corrosion resistance of bolting material
      - Tightening of joint to stop or reduce leakage
      - Inspection of other components in the immediate and surrounding vicinity to ensure no adverse conditions exist as a result of the leakage
      - Monitoring which shall include monitoring and trending as directed by the Inservice Inspection Program and NRC commitments.
3. Components with local areas of general corrosion that reduced the wall thickness by more than 10% have been repaired or replaced and accepted for continued use, or evaluated as being acceptable for continued use.

Recordable indications noted without corrective action may be accepted for continued use provided that the Kewaunee Nuclear Plant operations department accepts the recorded leakage or seepage as controllable, and that work orders have been initiated to correct leakage or seepage. As stated above, this would not apply reactor coolant pressure boundary leakage or seepage.

Requested Item

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

Response

Detection of low levels of 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 radioactivity and the containment sump level.

Leakage is monitored when the plant is at power and when it is shutdown. During normal plant operation, leakage is monitored in accordance with Technical Specification surveillance requirements. Containment airborne activity and humidity are monitored to detect leakage. During plant shutdown for refueling, inspections are performed per ASME B&PV Code, Section XI on the Class 1 pressure boundary to locate leakage and evaluate boric acid accumulation and corrosion.

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

a) Containment Atmosphere Relative Humidity – An increase in containment humidity is indicative of an external leak from the RCS. However, since this is less sensitive (2 gpm to 10 gpm), an increase in humidity due to a leak in the RCS should also show a significant increase in other plant monitors.

b) Containment Sump Level - Containment sump water level indication is provided in the main control room by one level indicator that can be used to detect primary coolant system leakage. The sump pump is started when the high level alarm is reached. The time the sump pump runs until the pump shutoff level is achieved is recorded in site procedures from which a volume and leak rate can be determined.

c) RPV Flange Leak Off – The reactor flange leakoff temperature is monitored. Temperatures higher than containment ambient indicate inner seal leakage. In the event that flange leakoff temperature remains high after the inner seal isolation valve is closed, the outer seal is leaking, and continued operation shall be evaluated.

Through-wall cracking of the bottom RPV head instrument tube penetration nozzles would be detected by the visual observation of leakage through the mirror insulation. The insulation is examined as evidence of leakage would be visible with boric acid deposits or wetted areas. Additionally, a remote VT-3 with a mini-sub is performed on the nozzles every 10 years from inside the RPV when the core barrel is removed.

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

NMC routinely checks the leakage of the primary system each week via plant surveillance procedure when the reactor is at power or in the hot shutdown condition. Typical leakage at the Kewaunee Nuclear Plant during an operating cycle is 0.03 to 0.04 gallons per minute (gpm). An increase to 0.1 gpm would signify a possible problem.

Visual inspections for leakage or evidence of leakage affecting carbon steel and low-alloy steel components are conducted each refueling outage per the Kewaunee Generic Letter (GL) 88-05 Program (Refer to Attachment 2, Table 1) and during plant start up. (The Kewaunee GL 88-05 Program is in response to NRC GL 88-05, "Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components in PWR Plants".) Additionally, containment entries are made at routine intervals during plant operation.

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

Unidentified RCS leakage is maintained less than one gpm per Technical Specifications. The limitation of one gpm for an unidentified source of leakage is sufficiently above the minimum detectable leak rate to provide a reliable indication of leakage, and is well below the capacity of one charging pump (60 gpm). 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 radiation monitoring, additional trending, conducting leak rate calculations, generating a corrective action, and attempting to locate the source of RCS leakage.

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

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

Kewaunee Nuclear Plant has been analyzed for susceptibility to PWSCC of the CRDM nozzles relative to Oconee 3 using the time-at-temperature model and plant specific input date reported in MRP-48, "PWR Materials Reliability Program Response to NRC Bulletin 2001-01". (MRP-48 was submitted by NEI to the NRC on August 21, 2001). NRC Bulletin 2001-01, "Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles" states, in part, "[an] effective visual examination should not be compromised by the presence of insulation, existing deposits on the RPV head, or other factors that could interfere with the detection of leakage. On January 3, 2002, NMC provided the results of the Kewaunee Nuclear Plant 100% bare metal effective visual examination.

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

The Kewaunee Nuclear Plant RPV vendor, Westinghouse, has made no recommendations for inspections for vessels with Alloy 600/82/182 material.

Kewaunee Nuclear Plant 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 site's 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 plant Technical Specifications (TS) based on the above-noted compliance with ASME B&PV Code, Section XI. Kewaunee TS require that a number of programs shall be established, implemented, and maintained. Specifically included among these programs, as described in Kewaunee TS 4.2, is the Inservice Inspection and Testing Program, which provides controls for inservice testing of ASME Code Class 1, 2, and 3 components. Kewaunee 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 fulfills the requirements of Kewaunee TS. That is, there is not a TS RCPB BACC program separate from the Inservice Inspection and Testing Program of Kewaunee TS 4.2.

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

- Surveillance Procedure SP-36-267 "ASME Boiler and Pressure Vessel Code Class 1 System Pressure Test".
  This procedure defines the requirements, responsibilities and techniques for performance of VT-2 examinations on ASME Class 1 systems and components.

  Surveillance Procedure SP-36-082 "Reactor Coolant System Leak Rate Check".
  The procedure defines the requirements, responsibilities and techniques for performing a weekly mass balance leakrate calculation, and for investigating and evaluating any RCS leak paths.

- General Nuclear Procedure GNP-01.05.01 "Indication Evaluation for Inservice Inspection".
  This procedure provides requirements for evaluation, disposition, root cause determination, and ASME B&PV Code Section XI acceptance or repair/replacement of VT-1, VT-2 and VT-3 noted indications.

- Nuclear Engineering Procedure NEP No.15.5 "Visual Examination for Inservice Inspection".
  This procedure defines the requirements, responsibilities and techniques for performance of VT-1 and VT-3 examinations on ASME Class 1, Class 2 and Class 3 systems and components.

- Surveillance Procedure SP-55-085 "Ten Year Inservice Inspection Requirements".
  This procedure provides general requirements for performance of the ASME B&PV Code Section XI 1989 Edition and Technical Specification 4.2 Inservice Inspection Program for ASME Class 1, Class 2 and Class 3 systems and components.

- "Kewaunee Nuclear Power Plant Third 10-year Inservice Inspection (ISI) Program 1994-2004 Augmented Program."
  This program describes the boric acid leak inspection program. This program is designed to fulfill NMC's requirements to Generic Letter 88-05. Additionally, the boric acid leak inspection program compliments the ASME B&PV Code Section XI pressure testing program by providing suitable methods for dispositioning boric acid leaks discovered during pressure testing.

EPRI Procedure 1006296, "Visual Examination for Leakage of PWR Reactor Head Penetrations."
The purpose of this procedure is to provide guidance on performing effective Visual Examination (VT-2) of RPV head penetrations to detect and characterize boron deposits due to leakage of RCS. Specific guidance is provided to enable personnel to distinguish deposits stemming from CRDM penetration leaks from deposits that may be present from other leakage sources.

- General Nuclear Procedure GNP-08.02.01 "Work Request/Work Order Processing".
  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 order.

- NMC Fleet Procedure NMC-FP-PA-ARP-01 "Action Request 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
KEWAUNEE NUCLEAR PLANT
DOCKET 50-305

JANUARY 20, 2003

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

9 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						
Alloy 600 RPV (40) CRDMs: 1 thru 35 and 37 thru 41 ¾ inch Vent Line	Visual	VT-3	100% In Place	Each Refueling Outage	100% : 4" Thick Panel Insulation	GNP-01.05.01, "Indication Evaluation for Inservice Inspection"
Alloy 600 RPV Instrument Tube Penetrations (36) RV-P1 through RV-P36	Visual	VT-2	100% in Place with Insulation Covering the Welds on The RPV Bottom Head	Each Refueling Outage	Not Removed. Mirror Insulation	GNP-01 05 01
Alloy 600 RPV Safe-End Weld with Nozzle End SA182 SI-W54DM	Visual	VT-2	100% as Accessible	Each Refueling Outage	Not Removed. Mirror Insulation:	GNP-01 05 01
Alloy 600 RPV Safe-End Weld with Nozzle End SA182 SI-W54DM	Surface and Volumetric	Liquid Penetrant and Ultrasonic	100%	Once Every 10 Years	100%: Mirror Insulation Removed	GNP-01 05 01
Alloy 600 RPV Safe-End Weld with Nozzle End SA182 SI-W112DM	Visual	VT-2	100% as Accessible	Each Refueling Outage	Not Removed: Mirror Insulation:	GNP-01.05 01
Alloy 600 RPV Safe-End Weld with Nozzle End SA182 SI-W112DM	Surface and Volumetric	Liquid Penetrant and Ultrasonic	100%	Once Every 10 Years	100%: Mirror Insulation Removed	GNP-01.05 01
Alloy 600 with Alloy 690 Cladding Steam Generator A Safe-End Weld RC-76DM	Visual	VT-2	100% as Accessible	Each Refueling Outage	Not Removed: Mirror Insulation:	GNP-01 05 01
Alloy 600 with Alloy 690 Cladding Steam Generator A Safe-End Weld RC-76DM	Surface and Volumetric	Liquid Penetrant and Ultrasonic	100%	Once Every 10 Years	100%: Mirror Insulation Removed	GNP-01 05 01
Alloy 600 with Alloy 690 Cladding Steam Generator A Safe-End Weld RC-77DM	Visual	VT-2	100% as Accessible	Each Refueling Outage	Not Removed: Mirror Insulation:	GNP-01 05 01
Alloy 600 with Alloy 690 Cladding Steam Generator A Safe-End Weld RC-77DM	Surface and Volumetric	Liquid Penetrant and Ultrasonic	100%	Once Every 10 Years	100%: Mirror Insulation Removed	GNP-01.05.01

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 with Alloy 690 Cladding Steam Generator B Safe-End Weld RC-78DM	Visual	VT-2	100% as Accessible	Each Refueling Outage	Not Removed: Mirror Insulation:	GNP-01.05 01
Alloy 600 with Alloy 690 Cladding Steam Generator B Safe-End Weld RC-78DM	Surface and Volumetric	Liquid Penetrant and Ultrasonic	100%	Once Every 10 Years	100%: Mirror Insulation Removed	GNP-01.05 01
Alloy 600 with Alloy 690 Cladding Steam Generator B Safe-End Weld RC-79DM	Visual	VT-2	100% as Accessible	Each Refueling Outage	Not Removed: Mirror Insulation.	GNP-01.05 01
Alloy 600 with Alloy 690 Cladding Steam Generator B Safe-End Weld RC-79DM	Surface and Volumetric	Liquid Penetrant and Ultrasonic	100%	Once Every 10 Years	100%: Mirror Insulation Removed	GNP-01.05 01
SA-182 Pressurizer Relief Safe-End Weld PR-W1DM	Visual	VT-2	100% in Place with the Insulation Covering the Weld	Each Refueling Outage	Not Removed: Mirror Insulation	GNP-01.05.01
SA-182 Pressurizer Relief Safe-End Weld PR-W1DM	Surface and Volumetric	Liquid Penetrant and Ultrasonic	100%	Once Every 10 Years	100%: Mirror Insulation Removed	GNP-01.05 01
SA-182 Pressurizer Safety Safe-End Weld PR-W16DM	Visual	VT-2	100% in Place with the Insulation Covering the Weld	Each Refueling Outage	Not Removed: Mirror Insulation	GNP-01 05 01
SA-182 Pressurizer Safety Safe-End Weld PR-W16DM	Surface and Volumetric	Liquid Penetrant and Ultrasonic	100%	Once Every 10 Years	100%: Mirror Insulation Removed	GNP-01.05 01
SA-182 Pressurizer Safety Safe-End Weld PR-W26DM	Visual	VT-2	100% in Place with the Insulation Covering the Weld	Each Refueling Outage	Not Removed: Mirror Insulation	GNP-01.05.01
SA-182 Pressurizer Safety Safe-End Weld PR-W26DM	Surface and Volumetric	Liquid Penetrant and Ultrasonic	100%	Once Every 10 Years	100%: Mirror Insulation Removed	GNP-01.05 01
SA-182 Pressurizer Spray Safe-End Weld PS-W61DM	Visual	VT-2	100% in Place with the Insulation Covering the Weld	Each Refueling Outage	Not Removed: Mirror Insulation	GNP-01.05 01

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
SA-182 Pressurizer Spray Safe-End Weld PS-W61DM	Surface and Volumetric	Liquid Penetrant and Ultrasonic	100%	Once Every 10 Years	100%: Mirror Insulation Removed	GNP-01.05 01
SA-182 Pressurizer Surge Safe-End Weld RC-W67DM	Visual	VT-2	100% in Place with the Insulation Covering the Weld	Each Refueling Outage	Not Removed. Mirror Insulation	GNP-01 05 01
SA-182 Pressurizer Spray Safe-End Weld RC-W67DM	Surface and Volumetric	Liquid Penetrant and Ultrasonic	100%	Once Every 10 Years	100%: Mirror Insulation Removed	GNP-01.05 01

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						
Safety Injection From Containment Pen.28N To Accumulators And Cold Leg Loops Valve SI-13A	Visual	VT-3	100% Bolts in Place	Each Refueling Outage	100%: Mirror Insulation	GNP-01.05 01
Safety Injection From Containment Pen 28N To Accumulators and Cold Leg Loops Valve SI-13B	Visual	VT-3	100% Bolts in Place	Each Refueling Outage	100% Mirror Insulation	GNP-01.05 01
Safety Injection-From Accumulator 1A To Loop A Cold Leg Valve SI-21A	Visual	VT-3	100% Bolts in Place	Each Refueling Outage	Not Insulated	GNP-01 05.01
Safety Injection-From Containment Pen 10 To Reactor From Accumulator 1B To Loop B Cold Leg Valve SI-21B	Visual	VT-3	100% Bolts in Place	Each Refueling Outage	Not Insulated	GNP-01 05 01
Safety Injection Pumps Suction Piping Valve SI-3	Visual	VT-3	100% Bolts in Place	Each Refueling Outage	100%: Mirror Insulation	GNP-01.05 01
Reactor Coolant From Pressurizer To Pressurizer Relief Tank Valve PR-2A	Visual	VT-3	100% Bolts in Place	Each Refueling Outage	Not Insulated	GNP-01.05 01
Reactor Coolant From Pressurizer To Pressurizer Relief Tank Valve PR-2B	Visual	VT-3	100% Bolts in Place	Each Refueling Outage	Not Insulated	GNP-01.05 01
Safety Injection-From Accumulator 1A To Loop A Cold Leg Valve SI-22A	Visual	VT-3	100% Bolts in Place	Each Refueling Outage	100%: Mirror Insulation	GNP-01.05 01
Safety Injection-From Containment Pen.10 To Reactor From Accumulator 1B To Loop B Cold Leg Valve SI-22B	Visual	VT-3	100% Bolts in Place	Each Refueling Outage	100%: Mirror Insulation	GNP-01.05 01
Safety Injection From Containment Pen. 10 To Reactor From Accumulator 1B To Loop B Cold Leg Valve SI-303A	Visual	VT-3	100% Bolts in Place	Each Refueling Outage	Not Insulated	GNP-01 05 01

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Component	Inspection Techniques	Personnel Qualifications	Extent of Coverage	Frequency	Degree of Insulation Removal/Insulation Type	Corrective Action
Safety Injection From Containment Pen. 48 To Reactor Valve SI-303B	Visual	VT-3	100% Bolts in Place	Each Refueling Outage	Not Insulated	GNP-01 05 01
Safety Injection From Containment Pen. 10 To Reactor From Accumulator 1B To Loop B Cold Leg Valve SI-304A	Visual	VT-3	100% Bolts in Place	Each Refueling Outage	100%: Mirror Insulation	GNP-01 05.01
Safety Injection From Containment Pen. 48 To Reactor Valve SI-304B	Visual	VT-3	100% Bolts in Place	Each Refueling Outage	100%: Mirror Insulation	GNP-01 05 01
Residual Heat Removal From Reactor Coolant Loops A and B Hot Legs To Containment Pen.9 and To Containment Sump B Valve RHR-2A	Visual	VT-3	100% Bolts in Place	Each Refueling Outage	Not Insulated	GNP-01 05 01
Residual Heat Removal From Reactor Coolant Loops A and B Hot Legs To Containment Pen 9 and To Containment Sump B Valve RHR-2B	Visual	VT-3	100% Bolts in Place	Each Refueling Outage	Not Insulated	GNP-01.05.01
Safety Injection Pumps Suction Piping Valve SI-2A	Visual	VT-3	100% Bolts in Place	Each Refueling Outage	100%: Mirror Insulation	GNP-01 05 01
Safety Injection Pumps Suction Piping Valve SI-2B	Visual	VT-3	100% Bolts in Place	Each Refueling Outage	100%: Mirror Insulation	GNP-01.05 01
Reactor Coolant From Pressurizer To Pressurizer Relief Tank Valve PR-3A	Visual	VT-3	100% Bolts in Place	Each Refueling Outage	Not Insulated	GNP-01.05 01
Reactor Coolant From Pressurizer To Pressurizer Relief Tank Valve PR-3B	Visual	VT-3	100% Bolts in Place	Each Refueling Outage	Not Insulated	GNP-01.05 01
Safety Injection Pumps Suction Piping Valve SI-4A	Visual	VT-3	100% Bolts in Place	Each Refueling Outage	Not Insulated	GNP-01 05 01
Safety Injection Pumps Suction Piping Valve SI-4B	Visual	VT-3	100% Bolts in Place	Each Refueling Outage	Not Insulated	GNP-01 05 01

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Component	Inspection Techniques	Personnel Qualifications	Extent of Coverage	Frequency	Degree of Insulation Removal/Insulation Type	Corrective Action
Residual Heat Removal From Reactor Coolant Loops A and B Hot Legs To Containment Pen.9 and To Containment Sump Valve RHR-1A	Visual	VT-3	100% Bolts in Place	Each Refueling Outage	100%: Mirror Insulation	GNP-01.05.01
Residual Heat Removal From Reactor Coolant Loops A and B Hot Legs To Containment Pen.9 and To Containment Sump B Valve RHR-1B	Visual	VT-3	100% Bolts in Place	Each Refueling Outage	100%: Mirror Insulation	GNP-01.05 01
Safety Injection-From Containment Pen.10 To Reactor From Accumulator 1B To Loop B Cold Leg Valve RHR-11	Visual	VT-3	100% Bolts in Place	Each Refueling Outage	Not Insulated	GNP-01 05 01
Chemical Volume Control To Pressurizer Reactor Coolant Valve CVC-15	Visual	VT-3	100% Bolts in Place	Each Refueling Outage	Not Insulated	GNP-01.05 01
Reactor Coolant From Pressurizer To Pressurizer Relief Tank Valve PR-1A	Visual	VT-3	100% Bolts in Place	Each Refueling Outage	Not Insulated	GNP-01.05.01
Reactor Coolant From Pressurizer To Pressurizer Relief Tank Valve PR-1B	Visual	VT-3	100% Bolts in Place	Each Refueling Outage	Not Insulated	GNP-01 05 01
Reactor Coolant RTD Line From Reactor Coolant Loop A Valve RC-103A	Visual	VT-3	100% Bolts in Place	Each Refueling Outage	100%: Mirror Insulation	GNP-01 05 01
Reactor Coolant RTD Line From Reactor Coolant Loop B Valve RC-103B	Visual	VT-3	100% Bolts in Place	Each Refueling Outage	100%: Mirror Insulation	GNP-01.05.01
Reactor Coolant To Pressurizer Valve PS-1A	Visual	VT-3	100% Bolts in Place	Each Refueling Outage	100%: Mirror Insulation	GNP-01.05.01
Reactor Coolant To Pressurizer Valve PS-1B	Visual	VT-3	100% Bolts in Place	Each Refueling Outage	100%: Mirror Insulation	GNP-01 05.01
Chemical Volume Control From Discharge Line Of Regenerative Heat Exchanger Anchor Point On Line To Reactor Coolant System Cold Leg Loop B Valve CVC-11	Visual	VT-3	100% Bolts in Place	Each Refueling Outage	Not Insulated	GNP-01.05.01

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Component	Inspection Techniques	Personnel Qualifications	Extent of Coverage	Frequency	Degree of Insulation Removal/Insulation Type	Corrective Action
Chemical Volume Control From Loop B of Pump Suction To Regenerative Heat Exchanger Valve LD-2	Visual	VT-3	100% Bolts in Place	Each Refueling Outage	100%: Mirror Insulation	GNP-01.05.01
Chemical Volume Control From Loop B of Pump Suction To Regenerative Heat Exchanger Valve LD-3	Visual	VT-3	100% Bolts in Place	Each Refueling Outage	100%: Mirror Insulation	GNP-01 05 01
Chemical Volume Control From Regenerative Heat Exchanger 1A and Point Near Valve LD-60 To Containment Pen. 11 Valve LD-4A	Visual	VT-3	100% Bolts in Place	Each Refueling Outage	100%: Mirror Insulation	GNP-01.05 01
Chemical Volume Control From Regenerative Heat Exchanger 1A and Point Near Valve LD-60 To Containment Pen. 11 Valve LD-4B	Visual	VT-3	100% Bolts in Place	Each Refueling Outage	100%: Mirror Insulation	GNP-01 05 01
Chemical Volume Control From Regenerative Heat Exchanger 1A and Point Near Valve LD-60 To Containment Pen. 11 Valve LD-4C	Visual	VT-3	100% Bolts in Place	Each Refueling Outage	100%: Mirror Insulation	GNP-01 05 01
Reactor Coolant RTD Line From Reactor Coolant Loop A Flange FE-458	Visual	VT-3	100% Bolts in Place	Each Refueling Outage	100%: Mirror Insulation	GNP-01.05 01
Reactor Coolant RTD Line From Reactor Coolant Loop B Flange FE-459	Visual	VT-3	100% Bolts in Place	Each Refueling Outage	100%: Mirror Insulation	GNP-01.05 01
Reactor Coolant From Pressurizer To Pressurizer Relief Tank Flange PR-F1	Visual	VT-3	100% Bolts in Place	Each Refueling Outage	Not Insulated	GNP-01.05.01
Reactor Coolant From Pressurizer To Pressurizer Relief Tank Flange PR-F2	Visual	VT-3	100% Bolts in Place	Each Refueling Outage	Not Insulated	GNP-01.05 01
RPV Closure Head Conoseal Bolting RV-CD34	Visual	VT-3	100% Bolts Disassembled	Each Refueling Outage	Not Insulated	GNP-01 05 01
RPV Closure Head Conoseal Bolting RV-CD35	Visual	VT-3	100% Bolts Disassembled	Each Refueling Outage	Not Insulated	GNP-01 05 01

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Component	Inspection Techniques	Personnel Qualifications	Extent of Coverage	Frequency	Degree of Insulation Removal/Insulation Type	Corrective Action
RPV Closure Head Conoseal Bolting RV-CD37	Visual	VT-3	100% Bolts Disassembled	Each Refueling Outage	Not Insulated	GNP-01.05.01
Steam Generator-1A (2) Manways	Visual	VT-2 or VT-3	100% Bolts Disassembled or in Place	Each Refueling Outage	100%: Panel Insulation	GNP-01 05 01
Steam Generator 1B (2) Manways	Visual	VT-2 or VT-3	100% Bolts Disassembled or in Place	Each Refueling Outage	100%: Panel Insulation	GNP-01 05 01
Pressurizer Manway	Visual	VT-3	100% Bolts in Place	Each Refueling Outage	100%: Panel Insulation	GNP-01.05 01

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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	100%	Each Refueling Outage	Not Removed: Mirror Insulation	GNP-01.05 01
Pressurizer	Visual	VT-2	100%	Each Refueling Outage	Not Removed: Mirror Insulation	GNP-01 05 01
Steam Generator 1A	Visual	VT-2	100%	Each Refueling Outage	Not Removed: Mirror Insulation	GNP-01.05 01
Steam Generator 1B	Visual	VT-2	100%	Each Refueling Outage	Not Removed: Mirror Insulation	GNP-01.05 01