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U. S. Nuclear Regulatory Commission
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Joseph M. Farley Nuclear Plant and
Vogtle Electric Generating Plant
Response to NRC Bulletin 2002-01
"Reactor Pressure Vessel Head Degradation
and Reactor Coolant Pressure Boundary Integrity"
60-Day Response
<u>Request for Additional Information</u>

Ladies and Gentlemen:

Pursuant to the requirements of Nuclear Regulatory Commission (NRC) Bulletin 2002-01, "Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity," 60-Day Response Request for Additional Information (RAI) issued to the Southern Nuclear Operating Company (SNC) Vogtle Project on November 14, 2002 and to the SNC Farley Project on November 15, 2002, SNC hereby submits the enclosed Attachment 1 and Attachment 2 which constitute the required 60-day responses to the RAI for Vogtle Electric Generating Plant (VEGP) Units 1 and 2 and Joseph M. Farley Nuclear Plant (FNP) Units 1 and 2, respectively.

The enclosed attachments contain the following three commitments:

1) FNP and VEGP commit to perform a best effort visual examination of the metal surface under the insulation of the RPV bottom head at each of their unit's next refueling outages.

2) FNP commits to follow the inspection program recommendations contained in MRP-75 when scheduling and performing its RPV upper head inspections, in addition to the existing ASME Section XI inspection requirements for the RPV upper head. VEGP previously committed to follow the inspection program recommendations contained in MRP-75 in the VEGP September 5, 2002 response to NRC Bulletin 2002-02 included in SNC letter LCV-1637-A.

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3) FNP and VEGP commit to perform a semi-annual sample and analysis of containment atmosphere for iron concentration as a measure to assist in the detection of low levels of RCS leakage. This measure is already in place at VEGP.

Mr. J. B. Beasley, Jr. states he is a Vice President of Southern Nuclear Operating Company, is authorized to execute this oath on behalf of Southern Nuclear Operating Company and to the best of his knowledge and belief, the facts set forth in this letter are true.

If there are any questions, please advise.

Respectfully submitted,

SOUTHERN NUCLEAR OPERATING COMPANY

J. B. Beasley, Jr.

Sworn to and subscribed before me this 17 *day of* January*, 2003.*

Jan C Edge
Notary Public

My commission expires: 7/27/05

JBB/DRG/sl

Attachments

cc: Southern Nuclear Operating Company
Mr. Jeffrey T. Gasser, Vice President – Plant Vogtle
Mr. G. R. Frederick, General Manager – Plant Vogtle
Mr. D. E. Grissette, General Manager – Plant Farley
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U. S. Nuclear Regulatory Commission, Washington, D. C.
Mr. F. Rinaldi, NRR Project Manager – Farley
Mr. F. Rinaldi, NRR Project Manager – Vogtle

U. S. Nuclear Regulatory Commission, Region II
Mr. L. A. Reyes, Regional Administrator
Mr. T. P. Johnson, Senior Resident Inspector – Farley
Mr. J. Zeiler, Senior Resident Inspector – Vogtle

ATTACHMENT 1

SNC Response to
NRC Bulletin 2002-01 Request for Additional Information
Regarding Boric Acid Corrosion Control Programs
for Vogtle Electric Generating Plant

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).**

2. **Provide the technical basis for determining whether or not insulation is removed to examine <u>all</u> 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.**

<u>Response to Items 1 & 2:</u>

Alloy 600 pressure boundary material locations and dissimilar metal Alloy 82/182 weld locations for the Vogtle reactor coolant pressure boundary (RCPB) are identified in document WCAP-12907, "Alloy 600 PWSCC Assessments of Vogtle 1 & 2 Primary Components". Please see a discussion below of each location. Also included in this document is a Technical Basis Summary Table listing each component's technical basis for inspection techniques used, personnel qualifications, extent of coverage, frequency of inspection, degree of insulation removal including insulation type and corrective action.

<u>RPV Top Head (Alloy 600 CRDM and Vent Line Nozzles)</u>

As previously noted in the Vogtle response to NRC Bulletin 2002-01 Item 1.A,

> "A visual inspection of the control rod drive mechanisms (CRDMs), canopy seal weld area, and the area inside the reactor vessel head shroud above the insulation is performed by a certified person during each refueling outage. The inspector looks for evidence of boric acid leaks or deposits. An inspection report is generated which records any leakage indication, including its suspected source and location. The

completed inspection report is provided to the engineering department for evaluation and resolution."

The inspectors that perform this inspection are VT-2 certified individuals. In addition to this visual inspection during each refueling outage, it is noted that complete 100% bare metal inspections of the Vogtle 1 & 2 RPV heads were completed during each unit's most recent refueling outage. These inspections were conducted by VT-2 certified personnel using remote automated or manual examination devices which eliminated the need to remove much of the vessel head insulation. For future inspections of the Vogtle RPV heads, as noted in response to NRC Bulletin 2002-02, SNC has committed to implement the MRP Inspection Plan.

RPV Flange Leakage Monitor Tube

This component is isolated from the RCS by the inner o-ring and therefore, has a low potential of causing boric acid corrosion. This component is included in the scope of the ASME Section XI Class 1 System Leakage Test. Also, leakage past the inner o-ring is monitored by temperature elements which alert operations personnel via annunciators in the control room if leakage is detected.

RPV Nozzles (Inconel Safe-End Buttering and Safe-End Welds)

A VT-2 certified inspector visually checks the RPV nozzles and the condition of surrounding areas during each refueling outage. This examination is performed during the Class 1 System Leakage Test with insulation in place and is conducted either from within the annulus area or from vantage points outside the annulus. In addition, insulation is removed and a UT and PT examination is performed on each nozzle to safe-end weld once per inspection interval in accordance with Section XI Examination Category B-F requirements.

RPV Bottom Head Instrument Tubes

The reactor cavity sump room provides access to the RPV bottom head area. The surface of the RPV bottom head has not previously been observed directly during the Class 1 System Leakage Test due to the "boxed-in" metal reflective insulation surrounding it. However, the insulation is not form-fitted against the RPV bottom head and therefore leakage from the instrument tube penetrations would tend to accumulate boron on the insulation and not on the bottom head itself. Examinations have been completed in the area below each reactor vessel and no boric acid streaks or stains have been observed on the outer surface of the Unit 2 insulation. Some minor streaks are present on the outer surface of the Unit 1 insulation; i.e., on one side of the insulation below where previous leakage occurred from the nuclear instrumentation cover in the annulus region. SNC plans to perform a best effort visual examination of the metal surface under the insulation of the RPV bottom head at each unit's next refueling outage.

Pressurizer Surge Nozzle (Inconel Safe-End Buttering and Safe-End Weld)

The surge nozzle is located in the bottom head of the pressurizer. Blanket insulation is used that is form-fitted to the bottom head of the pressurizer. A VT-2 certified inspector visually checks the surge nozzle and the condition of the surrounding area during each refueling outage. This examination is performed at nominal operating pressure and temperature with insulation in place as part of the Class 1 System Leakage Test during the refueling outage. In addition, insulation is removed and an UT and PT examination is performed on the nozzle to safe-end weld once per inspection interval in accordance with Section XI Examination Category B-F requirements.

Pressurizer Safety and Relief Nozzles and Spray Nozzle (Inconel Safe-End Buttering and Safe-End Welds)

The upper head of the pressurizer contains three safety nozzles, a relief nozzle, and a spray nozzle as well as the pressurizer manway. The insulation is blanket insulation and is removed from the manway cover during each refueling outage to allow inspection of the manway bolting but is typically not removed from the nozzles. A VT-2 certified inspector visually checks the nozzles and the condition of the surrounding areas during each refueling outage. A VT-2 examination is performed at nominal operating pressure and temperature with insulation in place as part of the Class 1 System Leakage Test during the refueling outage. In addition, insulation is removed and an UT and PT examination is performed on each nozzle to safe-end weld once per inspection interval in accordance with Section XI Examination Category B-F requirements.

Steam Generator Channel Head Bottom Drain (Inconel Drain Tube and Welds)

The channel head bottom drain exits the steam generators at bottom dead center of the channel head. The insulation covering the bottom head of each steam generator is blanket type insulation. While insulation is removed from the primary manways each outage, the insulation covering the bottom dead center of the steam generators is not routinely removed. However, for several of the steam generators, the channel head drain tubes have been VT-1 inspected from inside the steam generators. Additionally, leakage from the channel head bottom drains would be observable during the Class 1 System Leakage Test from below the steam generators.

Steam Generator Tubing (Inconel Alloy 600)

The steam generator tubing is examined in accordance with Technical Specification requirements and EPRI Steam Generator Examination Guidelines (i.e., per NEI 97-06 Steam Generator Program requirements).

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 to Item 3:

A leakage inspection is performed in containment as early as practical during every refueling outage, and at the discretion of the Operations department, during any shutdown that lasts longer than 72 hours. A VT-2 leak inspection of the RCS pressure boundary is performed at the end of each refueling outage, after nominal operating pressure and temperature have been achieved, per ASME Section XI, Article IWB-5000 requirements. The ASME Section XI inspection includes a thorough visual inspection of the entire RCS (including all Class 1 components and piping) out to the second closed boundary valve. No portions of the RCS are considered inaccessible for the ASME Section XI inspection, however, a VT-2 inspection of the area under the reactor vessel has not consistently been performed during previous refueling outages. For future outages the area under the reactor vessel will receive a regularly scheduled VT-2 inspection during the Class 1 System Leakage Test. For the technical basis for the extent and frequency of walkdowns, refer to the Technical Basis Table which responds to Items 1 and 2.

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 to Item 4:

All RCS pressure boundary bolted connections are examined for leakage during each refueling outage. This includes pipe flange connections, valve body-to-bonnet connections, reactor coolant pump flange connections, manway flange connections, relief valve to pipe connections, etc. The examination of bolted connections occurs in the refueling outage during the Class 1 System Leakage Test at nominal operating pressure and temperature. For Class 1 bolted connections that are insulated, an additional examination of the connection is performed with the insulation removed. In accordance with Inservice Inspection (ISI) Program Relief Requests RR-26, the removal of the insulation and followup examination may occur at ambient pressure and temperature conditions instead of under normal operating pressure conditions.

When active leakage and/or any significant buildup of boric acid residue is discovered at RCS Pressure Boundary bolted connections, an evaluation is conducted to determine the susceptibility of the bolting to corrosion and assess the potential for failure (assuming an immediate complete disassembly and inspection of the bolted connection is not warranted). The evaluation is performed in accordance with ISI Program Relief Request RR-25 and considers factors such as the material of the bolting, the corrosiveness of the leaking fluid, the leakage location, the leakage history of the connection, visual evidence of corrosion, the service age of the bolting, and the leakage path taken by any active leakage. As necessary, additional insulation is removed to inspect for corrosion damage of susceptible components in the leak flow path. Also, when excessive boron residue buildup is present, sufficient boron residue is removed to allow verification of corrosion damage to bolting. If the evaluation concludes that the connection can continue to perform its safety related function, then at the next component/system outage of sufficient duration, the bolt closest to the source of leakage is removed and a VT-1 visual examination of the bolt is performed by a certified examiner. If the initial evaluation concludes that the connection cannot conclusively perform its safety function, then the bolt closest to the source of leakage is promptly removed and VT-1 examined. Visual evidence during the initial evaluation that corrosion of a bolt has exceeded 5% of its cross-sectional area would be criteria for requiring prompt removal of the bolt (Ref: Section XI, paragraph IWB-3517.1). Followup monitoring activities for leaks which are not corrected in a prompt manner are determined on a case by case basis. Should it be determined that reinspection is prudent prior to the next refueling outage, then a plant action item is typically initiated to track performance of the inspection.

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 the bottom reactor pressure 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.**

<u>Response to Item 5:</u>

Refer to the response to Item 6 for a discussion of the measures that are employed to first detect and then identify the source of low levels of reactor coolant pressure boundary leakage. It is believed that those measures would identify the reactor vessel bottom head as the source of a leak. Should unexplained boron residue be found on the outer surface of the reflective insulation, sufficient insulation will be removed to identify the source and the potential impact on components located in the leak path.

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 to Item 6:

RCS operational leakage (identified, unidentified, and pressure isolation valve leakage) is monitored in accordance with technical specification requirements. Unidentified leakage is typically very low and is monitored for changes. Whenever a notable increase in unidentified leakage occurs, an investigation is initiated to determine the source of leakage. Steps that are typically taken to locate the source include: 1) reviewing recent trends of containment activity, moisture, and sump levels, 2) performing a walkdown of accessible areas of containment, 3) using a robot or other remote observation device to observe for leakage in the bioshield area of containment, and 4) performing a review and investigation of potential closed system leakage paths. The goal is to ensure that unidentified leakage is maintained sufficiently low to permit identification of new leaks at an early stage.

Other measures which have been implemented to assist in the detection of low levels of RCS leakage include: 1) The performance of a semiannual sample and analysis of containment atmosphere for iron concentration, and 2) The addition of procedure instructions to require that any significant amount of boron residue found on the containment coolers be investigated to determine the source.

If evidence of RCS pressure boundary leakage is identified, measures are in-place to identify the location, amount of leakage and/or boric acid residue, apparent source, observable corrosion damage, and apparent impacted components in the leak path. An evaluation is performed to verify the source, verify the extent of existing corrosion damage, assess the potential of further corrosion damage, identify susceptible components in the leak path, and to determine the need for monitoring and corrective actions. As necessary, insulation and/or boron residue is removed to complete the evaluation and assess the material condition of the affected component and any components in the leak path.

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.**

<u>**Response to Item 7:**</u>

Southern Nuclear Operating Company is a participant in the Electric Power Research Institute (EPRI) Material Reliability Project (MRP) and has applied the guidance provided by the MRP in reviewing the Vogtle boric acid inspection program. The MRP issued MRP-75, Revision 1, "PWR Reactor Vessel (RPV) Upper Head Penetration Inspection Plan" on September 6, 2002. The RPV head penetration nozzle inspection schedule presented in MRP-75 was based on the use of probabilistic fracture mechanics (PFM) analyses using the Monte-Carlo simulation algorithm to determine the probability of leakage and failure versus time for a set of input parameters, including head operating temperature, inspection types (visual or non-visual NDE) and inspection intervals. Input into this algorithm included an experience-based time to leakage correlation that used a Weibull model of plant inspections current at the time of the analysis, fracture mechanics analyses of various nozzle configurations containing axial and circumferential cracks and MRP developed statistical crack growth rate data for Alloy 600. The parameters used in the model were benchmarked against the most severe cracking found in the industry at the time the model was developed (B&W Plants) and produced results that were in agreement with experience at that time. This analysis assumed there exists an acceptable probability that primary leakage from a through-wall nozzle crack or J-groove weld crack would flow through the nozzle/head penetration interface to the top of the reactor pressure vessel head where it could be visibly identified.

The inspection schedule then employed plant categories defined by risk-informed susceptibility limits based on effective degradation years (EDY). EDY was defined as Effective Full Power Years (EFPY) @ 600°F (RPV head temperature). Low susceptibility plants were identified as having less than 10 EDY, without a leak or identified crack; moderate susceptibility plants were identified as having greater than or equal to 10 EDY and less than 18 EDY without a leak or identified through-wall crack; and high susceptibility plants were identified as having greater than or equal to 18 EDY or units that have identified leaks or through-wall cracks.

Per the MRP-75 criteria and the NRC criteria indicated in Bulletin 2002-02, Vogtle is considered a low susceptibility plant and has committed to follow the inspection program recommendations contained in MRP-75 when scheduling and performing its RPV upper head inspections, in addition to the existing ASME Section XI inspection requirements for the RPV upper head.

Otherwise, no other aspect of the Vogtle Boric Acid Corrosion Control (BACC) program makes use of susceptibility models or consequence models.

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.**

<u>**Response to Item 8:**</u>

In Westinghouse Owners Group (WOG) letter WOG-02-223 dated December 13, 2002, the WOG stated that it had reviewed databases and applicable communications to determine what recommendations Westinghouse had made to the owners of Westinghouse NSSSs on visual inspections of Alloy 600/82/182 materials in the reactor coolant pressure boundary. The detailed review of this information did not identify any generic recommendations by Westinghouse on visual inspections of Alloy 600/82/182 locations in Westinghouse NSSSs.

Even though there were no generic industry recommendations identified on visual inspections of Alloy 600/82/182 locations in Westinghouse NSSSs, Vogtle did receive plant specific visual examination recommendations through WCAP-12907 "Alloy 600 PWSCC Assessment of Vogtle 1 & 2 Primary Components" in May 1991. The WCAP was initiated by Vogtle to address Information Notice (IN) 90-10 "Primary Water Stress Corrosion Cracking (PWSCC) of Inconel 600" that was issued by the NRC on February 23, 1990. The IN recommended that licensees of all PWRs review their inconel 600 applications in the primary coolant pressure boundary, and when necessary, to implement an augmented inspection program. WCAP-12907 recommended visual inspections of the steam generator partition plate on five steam generators (SG 1881 – SG 1884, and SG 1982) and the channel drain tube on four steam generators (SG 1881 – SG 1884). Vogtle included the steam generator partition plate and channel drain tube visual examinations as augmented examinations in the Inservice Inspection Plans. The WCAP did not recommend augmented visual inspections on the Vogtle RPV or pressurizer nozzles containing Alloy 600/82/182 material.

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.**

<u>**Response to Item 9:**</u>

The RCS leakage detection systems, which are required by technical specifications, afford the ability to detect low levels of RCS leakage through a variety of independent means. In addition , the FSAR Chapter 15 accident analysis describes the design basis of Vogtle to address and mitigate the effects of RCS leakage. Finally the Vogtle boric acid inspection program that was developed in response to GL 88-05 is formalized by procedures that include a procedure (11864-C) for performing a containment general inspection to identify leaks or boric acid accumulations, procedures (85060-C, 14910-1 and 14910-2) for performing ASME Section XI leak inspections of the reactor coolant system (RCS), and guidance procedures for performing visual inspections. Also included in the program is a procedure (83201-C "Corrosion Assessment") that establishes the responsibilities and methodology for performing an engineering evaluation of boric acid leaks to assess the effects of corrosion on components/material exposed to the leakage flow path. Subparagraph 4.3.5 of the procedure specifically addresses ASME Section XI, paragraph IWA-5250 (b) by stating

> "When boric acid residues are discovered on ferritic steel components, the location of the leakage source and the areas of general corrosion, if any, must be determined. As necessary, insulation should be removed to complete this determination. General corrosion is an approximate uniform wastage of a surface of the component, through chemical or electrochemical reaction, free of deep pits or cracks. Components with local areas of general corrosion that reduce the wall thickness by more than 10% shall be evaluated to determine whether the component may be acceptable for continued service, or whether repair or replacement is required".

In summary, the combination of inspection plans, technical specification surveillance requirements, and design basis analysis makes up the BACC program and provides assurance that the technical specification requirements and the regulatory requirements are met. However, in light of the recent reactor vessel head corrosion identified at Davis-Besse, SNC is reviewing its BACC program to ensure that the lessons learned and operating experiences will be appropriately addressed in its inspection program.

SNC Response to BL 2002-01 RAI Items 1 and 2 Technical Basis Summary Table for Vogtle Electric Generating Plant						
Components with Alloy 600/82/182	**Inspection Techniques**	**Personnel Qualifications**	**Extent of Coverage**	**Frequency**	**Degree of Insulation Removal/Insulation Type**	**Corrective Action**
RPV Top Head (Alloy 600 CRDM nozzles and Vent Pipe)	Inspection techniques specified in Note 1 and MRP-75	See Note 2	Requirements per ASME Section XI, Table IWB-2500-1 and MRP-75	Requirements per ASME Section XI, Table IWB-2500-1 and MRP-75	Requirements in Note 6 and MRP-75 / Stepped reflective metal Insulation	Requirements contained in Note 4 and MRP-75
RPV Flange Leakage Monitor Tube	See Note 1	See Note 2	Per ASME Section XI, Table IWB-2500-1	Per ASME Section XI, Table IWB-2500-1	See Note 6 NA	See Note 4
RPV Nozzles [Inconel Safe-End (SE) Buttering and SE Welds]	Inspection techniques specified in Note 1 and Note 5	Requirements of Note 2 and Note 7	Per ASME Section XI, Table IWB-2500-1	Per ASME Section XI, Table IWB-2500-1	See Note 6 / Blanket Insulation	See Note 4
RPV Bottom Head Instrument Tubes	See Note s 1, 8 & 9	See Note 2	Per ASME Section XI, Table IWB-2500-1	Per ASME Section XI, Table IWB-2500-1	See Note 6 / Boxed-in metal reflective insulation	See Note 4
Pressurizer Surge Nozzle (Inconel SE Buttering and SE Weld)	Inspection techniques specified in Note 1 and Note 5 and WCAP-12907, Table 6-1	Requirements of Note 2 and Note 7	Per ASME Section XI, Table IWB-2500-1 and WCAP-12907, Table 6-1.	Per ASME Section XI, Table IWB-2500-1 and WCAP-12907, Figure 6-1.	See Note 6 / Blanket insulation form fitted to bottom head	See Note 4
Pressurizer Safety and Relief Nozzles and Spray Nozzle (Inconel SE Buttering and SE Welds)	Inspection techniques specified in Note 1 and Note 5 and WCAP-12907, Table 6-1	Requirements of Note 2 and Note 7	Per ASME Section XI, Table IWB-2500-1 and WCAP-12907, Table 6-1.	Per ASME Section XI, Table IWB-2500-1 and WCAP-12907, Figure 6-1.	See Note 6 / Blanket Insulation	See Note 4
Steam Generator (SG) Channel Head Bottom Drain (Inconel Drain Tube and Welds)	See Note 3	See Note 2	Per ASME Section XI, Table IWB-2500-1 and WCAP-12907, Table 6-1.	Per ASME Section XI, Table IWB-2500-1 and WCAP-12907, Figure 6-1.	See Note 6 / Blanket Insulation	See Note 4
SG Tubing (Inconel Alloy 600)	Eddy Current Exams per Tech Spec and EPRI SG Exam Guide	Certified	Per Tech Spec and EPRI SG Exam Guide	Per Tech Spec and EPRI SG Exam Guide	NA	Per Tech Spec and EPRI SG Exam Guide

Note 1 VT-2 exams per ASME Section XI, IWA-5000

Note 2 VT-2 examinations are performed by VT-2 certified personnel. Other leakage inspections or walkdowns may be performed by plant operators or other personnel who are qualified in their job but are not VT-2 certified

Note 3. VT-2 Exams per ASME Section XI, IWB-5000 and VT-1 Exams per WCAP-12907, Table 6-1

Note 4 Per ASME Section XI, IWA-5250 (b) and Vogtle Plant Procedure 83201-C, Subparagraph 4 3 5

Note 5. UT and PT exams per ASME Section XI, Table IWB-2500-1

Note 6. Per ASME Section XI, IWA 5242 and Vogtle Plant Procedure 83201-C, Subparagraph 4 3 5

Note 7 UT and PT exams are performed by certified personnel For future exams personnel will also be PDI certified as required by 10 CFR 50 55(a)

Note 8 In addition to the VT-2 exams per ASME Section XI, a best-effort visual examination of the metal surface under the insulation will be performed at each unit's next refueling outage

Note 9. While a VT-2 inspection of the area under the reactor vessel has not consistently been performed during previous refueling outages, it is intended that this be a regularly scheduled VT-2 inspection for future refueling outages

ATTACHMENT 2

SNC Response to
NRC Bulletin 2002-01 Request for Additional Information
Regarding Boric Acid Corrosion Control Programs
for Joseph M. Farley Nuclear Plant

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).**

2. **Provide the technical basis for determining whether or not insulation is removed to examine <u>all</u> 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.**

<u>**Response to Items 1 & 2:**</u>

Alloy 600 pressure boundary material locations and dissimilar metal Alloy 82/182 weld locations for the Farley Nuclear Plant (FNP) reactor coolant pressure boundary (RCPB) are discussed below. Also included in this document is a Technical Basis Summary Table listing each component's technical basis for inspection techniques used, personnel qualifications, extent of coverage, frequency of inspection, degree of insulation removal including insulation type and corrective action.

<u>RPV Top Head (Alloy 600 CRDM and Vent Line Nozzles)</u>

As previously noted in the SNC response to NRC Bulletin 2002-01 Item 1.A for FNP,

> "At every refueling outage prior to cooldown (i.e., at approximately normal operating temperature and pressure) a general inspection for evidence of reactor coolant system (RCS) leakage is performed. This inspection includes examination of components external to the reflective insulation on the reactor pressure vessel (RPV) head. A more detailed

inspection of the head area (but still external to the insulation) is performed after cooldown during head disassembly. Finally, a return-to-service inspection is performed at the end of the outage at approximately normal operating pressure. The startup inspection is performed in accordance with ASME Section XI requirements and specifically includes a visual examination of accessible RPV head components for any leakage or boron buildup. Any evidence of leakage is evaluated to identify and document the leakage path(s), the effects of the leakage and appropriate corrective actions."

In addition to this visual inspection, performed by VT-2 personnel during each refueling outage, it is noted that complete 100% bare metal inspections of the FNP Unit 1 & 2 RPV heads were completed during each unit's most recent refueling outage. These inspections were conducted by VT-2 certified personnel using remote automated or manual inspection devices which eliminated the need to remove much of the vessel head insulation. Moreover, volumetric NDE of all RPV head nozzles was performed on FNP Unit 2 during the most recent (Fall 2002) outage and similar inspection of Unit 1 is planned for the Spring 2003 outage. SNC plans to work with the EPRI MRP and NRC to determine the nature and frequency of future supplemental inspections for both FNP units. SNC commits to implementing the MRP Inspection Plan on the existing FNP RPV heads but notes that inspection plans will be influenced by SNC's intention to replace the RPV heads at both FNP units within the next three years. The planned new heads will use Alloy 690 for the CRDM and vent line nozzles.

<u>RPV Flange Leakage Monitor Tube</u>

This component is isolated from the RCS by the inner o-ring and therefore has a low potential of causing boric acid corrosion. This component is included in the scope of the ASME Section XI Class 1 System Leakage Test. Also, leakage past the inner o-ring is monitored by temperature elements which alert operations personnel via annunciators in the control room if leakage is detected.

<u>RPV Nozzles (Inconel Safe-End Buttering and Safe-End Welds)</u>

A VT-2 certified inspector visually checks the RPV nozzles and the condition of surrounding areas during each refueling outage. This examination is performed during the Class 1 System Leakage Test with insulation in place and is conducted either from within the annulus area or from vantage points outside the annulus. In addition, insulation is removed and a UT and PT examination is performed on each nozzle to safe-end weld once per inspection interval in accordance with Section XI Examination Category B-F requirements.

<u>RPV Bottom Head Instrument Tubes</u>

The reactor cavity sump room provides access to the RPV bottom head area. The surface of the RPV bottom head has not previously been observed directly during the Class 1

System Leakage Test due to the "boxed-in" metal reflective insulation surrounding it. However, the insulation is not form-fitted against the RPV bottom head and therefore leakage from the instrument tube penetrations would tend to accumulate boron on the insulation and not on the bottom head itself. SNC plans to perform a best-effort bare metal examination of the bottom head during the next refueling outage at each FNP unit by means of remotely manipulated visual inspection equipment inserted through the insulation.

<u>Pressurizer Surge Nozzle (Inconel Safe-End Buttering and Safe-End Weld)</u>

The surge nozzle as well as the heater penetrations are located in the bottom head of the pressurizer. The reflective metal insulation used is form-fitted to the bottom head of the pressurizer but contains holes to accommodate the heater penetrations. A VT-2 certified inspector visually checks the surge nozzle and the condition of the surrounding area during each refueling outage. This examination is performed at nominal operating pressure and temperature with insulation in place as part of the Class 1 System Leakage Test during the refueling outage. In addition, insulation is removed and a UT and PT examination is performed on the nozzle to safe-end weld once per inspection interval in accordance with Section XI Examination Category B-F requirements.

<u>Pressurizer Safety and Relief Nozzles and Spray Nozzle (Inconel Safe-End Buttering and Safe-End Welds)</u>

The upper head of the pressurizer contains three safety nozzles, a relief nozzle, and a spray nozzle as well as the pressurizer manway. The upper head is encased in reflective metal insulation. Insulation is removed from the manway cover during each refueling outage to allow inspection of the manway bolting but is typically not removed from the nozzles. A VT-2 certified inspector visually checks the nozzles and the condition of the surrounding areas during each refueling outage. A VT-2 examination is performed at nominal operating pressure and temperature with insulation in place as part of the Class 1 System Leakage Test during the refueling outage. In addition, insulation is removed and a UT and PT examination is performed on each nozzle to safe-end weld once per inspection interval in accordance with Section XI Examination Category B-F requirements.

<u>Steam Generators</u>

Steam Generators (SGs) on both FNP units have recently been replaced and the new SGs contain no Alloy 600/82/182 materials.

3. **Describe the technical basis for the extent and frequency of walkdowns and the method for evaluating the potential for leakage in <u>inaccessible areas</u>. 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 to Item 3:

A leakage inspection is performed in containment as early as practical during every refueling outage, and at the discretion of Operations department management during other shutdowns. A VT-2 leak inspection of the RCS pressure boundary is performed at the end of each refueling outage, after nominal operating pressure and temperature have been achieved, per ASME Section XI, Article IWB-5000 requirements. The ASME Section XI inspection includes a thorough visual inspection of the entire RCS (including all Class 1 components and piping) out to the second closed boundary valve. No portions of the RCS are considered inaccessible for the ASME Section XI inspection, however, a VT-2 inspection of the area under the reactor vessel has not consistently been performed during previous refueling outages. For future outages the area under the reactor vessel will receive a regularly scheduled VT-2 inspection during the ASME Section XI Class 1 System Leakage Test. For the technical basis for the extent and frequency of walkdowns, refer to the Technical Basis Table which responds to Items 1 and 2.

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 to Item 4:

All RCS pressure boundary bolted connections are examined for leakage during each refueling outage, except as noted below. This includes pipe flange connections, valve body-to-bonnet connections, reactor coolant pump flange connections, manway flange connections, relief valve to pipe connections, etc. The examination of bolted connections occurs in the refueling outage during the Class 1 system leakage test at nominal operating pressure and temperature. For bolted connections that are insulated, an additional examination of the connection is performed with the insulation removed. In accordance with Inservice Inspection (ISI) Program Relief Request RR-27, the removal of the insulation and followup examination may occur at ambient pressure and temperature conditions instead of under normal operating pressure conditions. The insulation is removed every refueling outage to satisfy the Code except for approximately thirty-two bolted connections per unit consisting of corrosion-resistant materials. Those bolted connections are covered by Relief Request RR-42 which allows the insulation to remain in place every refueling outage except once in the ten-year ISI interval. In order that these bolted connections be examined on an ongoing basis, approximately one-third of these bolted connections are uncovered each 40-month ISI period for a direct visual examination.

When active leakage and/or any significant buildup of boric acid residue is discovered at RCS pressure boundary bolted connections, an evaluation is conducted to determine the susceptibility of the bolting to corrosion and assess the potential for failure (assuming an immediate complete disassembly and inspection of the bolted connection is not warranted). The evaluation is performed in accordance with ISI Program Relief Request RR-41 and considers factors such as the material of the bolting, the corrosiveness of the leaking fluid, the leakage location, the leakage history of the connection, visual evidence of corrosion, the service age of the bolting, and the leakage path taken by any active leakage. As necessary, additional insulation is removed to inspect for corrosion damage of susceptible components in the leak flow path. Also, when excessive boron residue buildup is present, sufficient boron residue is removed to allow verification of corrosion damage to bolting. If the initial evaluation concludes that the connection cannot conclusively perform its safety function, then the bolt closest to the source of leakage is promptly removed and VT-1 examined. Visual evidence during the initial evaluation that corrosion of a bolt has exceeded 5% of its cross-sectional area would be criteria for requiring prompt removal of the bolt (Ref: Section XI, paragraph IWB-3517.1). Followup monitoring activities for leaks which are not corrected in a prompt manner are determined on a case by case basis. Should it be determined that reinspection is prudent prior to the next refueling outage, then a plant action item is typically initiated to track performance of the inspection.

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 the bottom reactor pressure 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.**

<u>**Response to Item 5:**</u>

Refer to the response to Item 6 for a discussion of the measures employed to detect, identify and then evaluate the source and effects of low levels of reactor coolant pressure boundary leakage. It is believed that those measures would identify the reactor vessel bottom head as the source of a leak. Should unexplained boron residue be found on the outer surface of the reflective insulation, sufficient insulation will be removed to identify the source and the potential impact on components located in the leak path.

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 to Item 6:

RCS operational leakage (identified, unidentified, and pressure isolation valve leakage) is monitored in accordance with technical specification requirements. Unidentified leakage is typically very low and is monitored for changes. Whenever a notable increase in unidentified leakage occurs, an investigation is initiated to determine the source of leakage. Steps that are typically taken to locate the source include: 1) reviewing recent trends of containment activity, moisture, and sump levels, 2) performing a walkdown of accessible areas of containment, and 3) performing a review and investigation of potential closed system leakage paths. The goal is to ensure that unidentified leakage is maintained sufficiently low to permit identification of new leaks at an early stage.

Other measures which have been implemented to assist in the detection of low levels of RCS leakage include: 1) weekly checks of the containment atmosphere radiation monitors for filter paper discoloration, and 2) inspection of the containment coolers for chemical deposits during each refueling outage. Any such discoloration or chemical deposits (e.g. boric acid) are required to be analyzed and a plan developed to determine their source. Another measure which will be implemented is performance of a semi-annual sample and analysis of containment atmosphere for iron concentration.

If evidence of RCS leakage is discovered, measures are in place to identify the location, amount of leakage and/or boric acid residue, apparent source, observable corrosion damage, and apparent impacted components in the leak path. An evaluation is performed to verify the source, verify the extent of existing corrosion damage, assess the potential of further corrosion damage, identify susceptible components in the leak path, and to determine the need for monitoring and corrective actions. As necessary, insulation and/or boron residue is removed to complete the evaluation and assess the material condition of the affected component and any components in the leak path.

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 to Item 7:

Southern Nuclear Operating Company is a participant in the Electric Power Research Institute (EPRI) Material Reliability Project (MRP) and has applied the guidance provided by the MRP in reviewing the FNP boric acid inspection program. The MRP issued MRP-75, Revision 1, "PWR Reactor Vessel (RPV) Upper Head Penetration Inspection Plan" on September 6, 2002. The RPV upper head penetration nozzle inspection schedule presented in MRP-75 was based on the use of probabilistic fracture mechanics (PFM) analyses using the Monte-Carlo simulation algorithm to determine the probability of leakage and failure versus time for a set of input parameters, including head operating temperature, inspection types (visual or non-visual NDE) and inspection intervals. Input into this algorithm included an experience-based time to leakage correlation that used a Weibull model of plant inspections current at the time of the analysis, fracture mechanics analyses of various nozzle configurations containing axial and circumferential cracks and MRP developed statistical crack growth rate data for Alloy 600. The parameters used in the model were benchmarked against the most severe cracking found in the industry at the time the model was developed (B&W Plants) and produced results that were in agreement with experience at that time. This analysis assumed there exists an acceptable probability that primary leakage from a through-wall nozzle crack or J-groove weld crack would flow through the nozzle/head penetration interface to the top of the reactor pressure vessel head where it could be visibly identified.

The inspection schedule then employed plant categories defined by risk-informed susceptibility limits based on effective degradation years (EDY). EDY was defined as Effective Full Power Years (EFPY) @ 600°F (RPV head temperature). Low susceptibility plants were identified as having less than 10 EDY, without a leak or identified crack; moderate susceptibility plants were identified as having greater than or equal to 10 EDY and less than 18 EDY without a leak or identified through-wall crack; and high susceptibility plants were identified as having greater than or equal to 18 EDY or units that have identified leaks or through-wall cracks.

Per the MRP-75 criteria FNP is considered a moderate susceptibility plant. FNP is committed to follow the inspection program recommendations contained in MRP-75 when scheduling and performing its RPV upper head inspections, in addition to the existing ASME Section XI inspection requirements for the RPV upper head. This program will be influenced by SNC's intention to replace the RPV heads at both FNP units within the next three years. The planned new heads will use Alloy 690 for the CRDM and vent line nozzles.

Otherwise, no other aspect of the FNP Boric Acid Corrosion Control (BACC) program makes use of susceptibility models or consequence models.

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 to Item 8:

In Westinghouse Owners Group (WOG) letter WOG-02-223 dated December 13, 2002, the WOG stated that it had reviewed databases and applicable communications to determine what recommendations Westinghouse had made to the owners of Westinghouse NSSSs on visual inspections of Alloy 600/82/182 materials in the reactor coolant pressure boundary. The detailed review of this information did not identify any generic recommendations by Westinghouse on visual inspections of Alloy 600/82/182 locations in Westinghouse NSSSs. While Westinghouse made some plant specific recommendations (e.g. WCAP-12907, "Alloy 600 PWSCC Assessment of Vogtle 1 & 2 Primary Components," May 1991), Farley Nuclear Plant (FNP) did not receive a plant specific WCAP.

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 to Item 9:

The RCS leakage detection systems, which are required by technical specifications, afford the ability to detect low levels of RCS leakage through a variety of independent means. In addition , the FSAR Chapter 15 accident analysis describes the plant design basis to address and mitigate the effects of RCS leakage. Finally the boric acid inspection program that was developed in response to GL 88-05 is formalized by procedures that include procedures (FNP-1/2-UOP-2.2 and -STP-34.0 and 34.1) for performing a containment general inspection to identify leaks or boric acid accumulations, procedures (FNP-1/2-STP-156.0, 156.1, 156.2 and 157.0) for performing ASME Section XI leak inspections of the reactor coolant system (RCS), and guidance procedures for performing visual inspections. Also included in the program is a procedure (FNP-0-M-101) that establishes the responsibilities and methodology for performing an engineering evaluation of boric acid leaks to assess the effects of corrosion on components/material exposed to the leakage flow path. Paragraph 8.5 of the procedure specifically addresses ASME Section XI, paragraph IWA-5250 (b) by stating

> "When boric acid residues are discovered on ferritic steel components, the location of the leakage source and the areas of general corrosion, if any, must be determined. General corrosion is an approximate uniform wastage of a surface of the component, through chemical or electrochemical reaction, free of deep pits or cracks. Components with local areas of general corrosion that reduce the wall thickness by more than 10% shall be evaluated to determine whether the component may be acceptable for continued service, or whether repair or replacement is required".

Also, subparagraph 6.2.3 states,

> "Discoloration or residue on surfaces examined shall be given particular attention to detect evidence of boric acid accumulations from borated water leakage. Insulation shall be removed to facilitate inspection for corrosion damage when there is evidence of boric acid leakage."

In summary, the combination of inspection plans, technical specification surveillance requirements, and design basis analysis makes up the BACC program and provides assurance that the technical specification requirements and the regulatory requirements are met. However, in light of the recent reactor vessel head corrosion identified at Davis-Besse, SNC is reviewing its BACC program to ensure that the lessons learned and operating experiences will be appropriately addressed in its inspection program.

SNC Response to BL 2002-01 RAI
Items 1 and 2 Technical Basis Summary Table for Farley Nuclear Plant

Components with Alloy 600/82/182	Inspection Techniques	Personnel Qualifications	Extent of Coverage	Frequency	Insulation Type / Degree of Removal	Corrective Action
RPV Top Head (Alloy 600 CRDM and Vent Line Nozzles)	Notes 1 & 3, MRP-75	Note 2	Per ASME Section XI, Table IWB-2500-1, MRP-75 and Note 3	Per ASME Section XI, Table IWB-2500-1, MRP-75 and Note 3	Reflective Metal Insulation (RMI); MRP-75, Notes 3 & 6	Note 4 and MRP-75
RPV Flange Leakage Monitor Tube	Note 1	Note 2	Per ASME Section XI, Table IWB-2500-1	Per ASME Section XI, Table IWB-2500-1	RMI; Note 6	Note 4
RPV RCS Nozzles [Inconel Safe End (SE) Buttering and SE Welds]	Notes 1 & 5	Notes 2 & 7	Per ASME Section XI, Table IWB-2500-1	Per ASME Section XI, Table IWB-2500-1	RMI; Note 6	Note 4
RPV Bottom Head Instrument Tubes	Notes 1, 8 & 9	Note 2	Per ASME Section XI, Table IWB-2500-1	Per ASME Section XI, Table IWB-2500-1	RMI; Note 6	Note 4
Pressurizer Surge Nozzle (Inconel SE Buttering and SE Weld)	Notes 1 & 5	Notes 2 & 7	Per ASME Section XI, Table IWB-2500-1	Per ASME Section XI, Table IWB-2500-1	RMI; Note 6	Note 4
Pressurizer Safety and Relief Nozzles and Spray Nozzle (Inconel SE Buttering and SE Welds)	Notes 1 & 5	Notes 2 & 7	Per ASME Section XI, Table IWB-2500-1	Per ASME Section XI, Table IWB-2500-1	RMI; Note 6	Note 4
Steam Generators (SGs)	N/A – SGs on both FNP units have recently been replaced; the new SGs contain no Alloy 600/82/182 materials.					

Note 1: VT-2 exams per ASME Section XI, IWA-5000.

Note 2: VT-2 exams are performed by VT-2 certified personnel. Other leakage inspections or walkdowns may be performed by plant operators or other personnel who are qualified in their jobs but not VT-2 certified.

Note 3: A 100% bare metal visual top of head inspection and volumetric NDE of all RPV top head nozzles was performed on FNP Unit 2 during the Fall 2002 outage. Top of head inspection was performed by lifting selected RMI panels for insertion of remote video equipment. Nozzle NDE was performed by remote tooling from beneath the RPV head. Similar inspection of FNP Unit 1 is planned for the Spring 2003 outage.

Note 4: Per ASME Section XI, IWA-5250 (b) and plant procedures (FNP-1/2-SOP-1.4)

Note 5: UT and PT exams per ASME Section XI, Table IWB-2500-1.

Note 6: Per ASME Section XI, IWA 5242 and plant procedure FNP-0-M-101 subparagraph 6.2.3.

Note 7: UT and PT exams are performed by certified personnel. For future exams they will also be PDI certified as required by 10 CFR 50.55(a).

Note 8: In addition to the VT-2 exams per ASME Section XI, a best-effort visual examination of the metal surface under the insulation will be performed at each unit's next refueling outage.

Note 9: While a VT-2 inspection of the area under the reactor vessel has not consistently been performed during previous refueling outages, it is intended that this be a regularly scheduled VT-2 inspection for future refueling outages.