CP&L
A Progress Energy Company

James Scarola
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Ref: 10 CFR 50.54(f)

HNP-02-164

JAN 2 4 2003

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555-0001

Subject: Harris Nuclear Plant – Request for Additional Information, Bulletin 2002-01, "Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity"

References: 1. NRC to Carolina Power & Light, (CP&L) letter, dated November 22, 2002 and received November 26, 2002, Bulletin 2002-01, "Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity, "60-Day Response for Harris Nuclear Plant Request for Additional Information (TAC No. MB4539)"

2. CP&L to NRC letter, HNP-02-052, dated April 02, 2002, "Harris Nuclear Plant – 15-Day Response to NRC Bulletin 2002-01, "Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity"

3. CP&L to NRC letter, HNP-02-063, dated May 15, 2002, Harris Nuclear Plant – 60-Day Response to Bulletin 2002-01, "Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity"

4. CP&L to NRC letter, HNP-02-118, dated September 12, 2002, "Harris Nuclear Plant – 30-Day Response to NRC Bulletin 2002-02, "Reactor Pressure Vessel Head and Vessel Head Penetration Nozzle Inspection Programs"

Dear Sir:

Reference 1 contains 9 questions regarding the Harris Nuclear Plant (HNP), 60-Day Response to Bulletin 2002-01, "Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity." Progress Energy Carolinas, Inc. (PEC) is providing the information requested in the Attachment to this letter.

The Attachment concludes that the HNP Boric Acid Corrosion, Inspection and Evaluation Program is in compliance with the applicable regulatory requirements discussed in GL 88-05 and NRC Bulletin 2002-01. Additionally, the program incorporates plant and industry operating experience. The program will continue being evaluated and enhanced, as needed, incorporating industry experience and best practices.

This letter establishes no new regulatory commitments.

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If you have any questions regarding this submittal, please contact Mr. John Caves, Supervisor, Licensing and Regulatory Programs at (919) 362-3137.

Sincerely,

James Scarola

RTG

Attachment: Response to Request for Additional Information, Items 1 Through 9 Regarding Harris Nuclear Plant, 60-Day Response for NRC Bulletin 2002-01, "Reactor Pressure Vessel Head Degradation And Reactor Coolant Pressure Boundary Integrity"

c:
Mr. J. B. Brady, NRC Sr. Resident Inspector
Ms. Beverly Hall, Section Chief, Radiation Protection Section, N.C. DENR
Mr. C. P. Patel, NRC Project Manager
Mr. L. A. Reyes, NRC Regional Administrator

James Scarola, having been first duly sworn, did depose and say that the information contained herein is true and correct to the best of his information, knowledge and belief, and the sources of his information are employees, contractors, and agents of Progress Energy Carolinas, Inc.

Darlene S. Yarborough
Notary (Seal)

My commission expires:
2-21-2005

DARLENE S. YARBOROUGH
NOTARY
***
PUBLIC
HARNETT COUNTY, NC

SERIAL: HNP-02-164

PROGRESS ENERGY CAROLINAS, INC.

(Alternately known as Carolina Power & Light)

HARRIS NUCLEAR PLANT

DOCKET NUMBER 50-400/LICENSE NUMBER NPF-63

ATTACHMENT

Response to Request for Additional Information, Items 1 through 9 Regarding Harris Nuclear Plant, 60-Day Response for NRC Bulletin 2002-01, "Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity"

Introduction:

The Westinghouse Owner's Group (WOG) and Electric Power Research Institute's (EPRI) Materials Reliability Program (MRP) have determined that the Harris Nuclear Plant (HNP) has a very low likelihood of experiencing Primary Water Stress Corrosion Cracking (PWSCC) in the Reactor Pressure Vessel (RPV) head, based on the plant's operating temperature and current time in service. These groups are currently analyzing Alloy 600 issues for other components, including inspection methods and frequencies.

As discussed in detail in the responses to the following questions, it has been determined that ASME Code Section XI inspection requirements, supplemented by additional measures implemented by HNP where judged to be prudent based on operating experience, constitute a comprehensive Boric Acid Corrosion Control Program. We will continue to review and update our programs in response to plant and industry experience, as well as the recommendations of industry groups investigating Alloy 600 issues.

The NRC staff's review of the licensees' responses to Bulletin 2002-01 resulted in the issuance of a Request for Additional Information (RAI). In accordance with NRC's request, Progress Energy Carolinas, Inc. (alternately known as Carolina Power & Light) is providing the following responses to the RAI for the HNP. The information provided below in conjunction with information previously provided constitute the basis for concluding that HNP's Boron Corrosion Program is providing reasonable assurance of compliance with the applicable regulatory requirements discussed in Generic Letter 88-05 and Bulletin 2002-01.

Question:

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. 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 bottom head).*

Response:

Table A identifies the Alloy 600 pressure boundary components and Alloy 82/182 welds that are currently in place at HNP. Also included in the table are the inspection technique, extent of coverage, inspection frequency, type of insulation, and the degree of insulation removal performed to facilitate the inspection. These items are scheduled with the remainder of the In-Service Inspection (ISI) exams and are performed by inspectors who are trained and qualified in accordance with ASME Code requirements.

The technical bases for the scheduled examination of Control Rod Drive Mechanism (CRDM) nozzles and J-groove welds are primarily provided in Electric Power Research Institute's (EPRI) Materials Reliability Program (MRP) document MRP-75 (*PWR Reactor Pressure Vessel (RPV) Upper Head Penetrations Inspection Plan*), supplemented by our responses to Bulletin 2002-01 (SERIAL: HNP-02-063) and 2002-02 (SERIAL: HNP-02-118). As shown in MRP-48, *PWR Materials Reliability*

Program Response to NRC Bulletin 2001-01, HNP's Effective Full Power Years (EFPY) is 11.6 as of February 1, 2001. Consequently, Harris Nuclear Plant is considered to be in the NRC category of plants with a very low likelihood of cracking of reactor pressure vessel (RPV) head penetration nozzles. At this time, no significant safety issue has been identified for plants in this category. In addition, HNP has not previously identified either leakage from, or cracking in, Vessel Head Penetration (VHP) nozzles.

The technical basis for the remainder of examinations is defined by the requirements of the American Society of Mechanical Engineers Boiler & Pressure Vessel (ASME B&PV) Code, Section XI, for visual, surface and volumetric examinations. Additionally, when performing surface and volumetric exams, evidence of leakage and boric acid residue are readily detectable.

HNP will perform a 100% bare metal visual inspection (BMV) of the top of the RPV closure head during the upcoming refueling outage (RFO), currently scheduled to begin in April 2003, as committed to in our response to NRC Bulletin 2002-01. This inspection exceeds ASME Code Section XI requirements. Additionally, HNP has recently received revised guidance from the MRP regarding baseline inspections of low-susceptibility plants' CRDM nozzles and J-groove welds. We plan to implement the MRP inspection recommendations, and are currently reviewing this guidance to determine the appropriate implementation schedule.

Reactor coolant system (RCS) piping and fittings are austenitic stainless steel and, thus, have a low susceptibility to boric acid corrosion. Locations where Alloy 600 material or Alloy 82/182 welds exist are inspected for evidence of leakage each refueling outage under the ASME Section XI Class I Pressure Test Program (Engineering Surveillance Test, EST-227, *ASME Section XI Class 1 System Pressure Test*).

There has been no operating experience which would indicate that leakage from through-wall cracking in the bottom reactor pressure vessel head incore instrumentation nozzles is an industry issue at this time. Therefore, the Code-required inspections have not been augmented. As indicated in Table A, these examinations are performed every refueling outage, in accordance with the ASME Code. The industry is evaluating Alloy 600 issues and we will incorporate their recommendations as appropriate for HNP.

Evidence of leakage from through-wall cracking in the bottom reactor pressure vessel head incore instrumentation nozzles would be detected by visual examinations which are performed every refueling outage under EST-227. The examinations consist of visual examination of the insulation, the tubes outside the insulation, and the floor below the vessel for indications of leakage. Because these tubes are located on the downward face of the vessel, leakage that could result in significant corrosion is not likely to accumulate on the vessel itself. Also, evidence of leakage would be apparent on the insulation, tubes or floor. Therefore, it is concluded that these examinations are adequate to verify the possible existence of leakage from these locations, and any leakage would be identified before significant corrosion can occur. The need for future inspections will be determined based on careful consideration of industry standards (ASME Code, Section XI) and the recommendations of the EPRI Materials Reliability Program.

In addition, susceptible areas are inspected during scheduled refueling outages and selected forced outages in accordance with the HNP Boron Corrosion Program (Plant Program Procedure, PLP-600, *Boron Corrosion Program*) and Containment boric acid walkdown procedure (Operations Periodic Test, OPT-1519, *Containment Visual Inspection for Boron and Evaluation of Containment Sump*

Inleakage Every Refueling Outage Shutdown). Components susceptible to boric acid corrosion are included in the inspections required by these procedures. In addition, potential targets as well as specific components are inspected for evidence of leakage.

The walkdowns performed to detect evidence of leakage from borated systems are performed at the start and end of every refueling outage in accordance with ASME Section XI. The walkdowns cover the RCS pressure boundary, with specific components also identified for inspection in the applicable plant procedures, as well as borated systems beyond the scope of ASME Section XI. There are no inaccessible areas during refueling outages that inhibit the performance of visual examinations for the purpose of identifying leakage.

These outage walkdowns are performed by a team consisting, as a minimum, of the Boron Corrosion Program Engineer, VT-2 qualified inspectors, and a Radiation Control technician. The VT-2 qualified inspectors receive training in accordance with EPRI's "Visual Examination for Leakage of PWR Reactor Head Penetrations." This document provides additional guidance on performing effective visual examinations (VT-2) to detect and characterize boron deposits.

In addition to these outage walkdowns, certified VT-2 inspectors perform visual exams on the bolted connections in the RCS in accordance with EST-227, as required by the ASME Code, Section XI, during each refueling outage. These exams are performed with the insulation removed. The ASME Code-required RCS system leak test is performed during plant shutdown and startup (Mode 3) by certified visual inspectors. The scope of the inspection boundary includes the RCS. The HNP Corrective Action Program (CAP) is used to document, track, investigate, and correct adverse conditions identified during these exams, as well as all walkdowns.

In addition, walkdowns are performed during forced outages in accordance with OPT-1519 and PLP-600. These walkdowns are currently performed at the discretion of plant management and the Boron Corrosion Program Engineer. The decision to perform a walkdown during a forced outage is made based on such considerations as the time since the previous walkdown, indications of possible leakage, the duration of the forced outage, and recent operation experience. The extent of these walkdowns is dependent upon plant conditions. These walkdowns exceed ASME Code Section XI requirements.

The health physics personnel performing decontamination evolutions are aware of the effects of boric acid corrosion and, by procedure (WCM-002, *Work Control Manual Procedure*), are instructed in the Work Order to report any signs of degradation observed during their activities. Operators also perform walkdowns to determine leakage amounts and cleanliness per the applicable procedure (OPT-1519). System Engineers perform periodic walkdowns of their systems in accordance with Technical Support Management Manual, TMM-117, *System Walkdowns and Observations,* which instructs them to look for evidence of boric acid leaks.

These procedural requirements ensure that leaks that are smaller than the allowable Technical Specification limit are identified, and that leakage is identified before degradation which may challenge structural integrity occurs. Future inspection plans will be developed based on careful consideration of industry standards (ASME Code, Section XI) and the recommendations of the EPRI Materials Reliability Program.

Question:

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.*

Response:

Alloy 600 components and Alloy 82/182 welds that are insulated are identified in Table A. The type of insulation is noted, as is the degree of insulation removal performed to facilitate the required examinations. In summary, insulation is removed to permit examination of reactor coolant pressure boundary welds and components where required by the ASME Code, Section XI. There are no limitations to the removal of insulation where insulation removal is required to perform inspections. As permitted by the ASME Code, insulation is not required to be removed in order to perform VT-2 inspections.

Where insulation is not required to be removed to facilitate inspection, specific guidance on inspection methodology is provided in plant procedures (PLP-600, EST-227, and PLP-652, *ASME Boiler and Pressure Vessel Code Section XI Pressure Test Program*) to ensure that any leakage is identified. This includes a description of specific characteristics to look for (e.g., stains or discoloration, deposits) as well as requiring that surrounding areas be examined for evidence of leakage. Once leakage has been identified, removal of insulation is required to identify the source, determine if degradation has occurred, and to evaluate the material condition of the affected systems, structures or components (SSC).

Concerning bolted connections, the In-Service Inspection (ISI) program was updated to meet the requirements of the 1989 Edition of the ASME Code, Section XI. This edition of the Code requires the insulation be removed from bolted connections in borated systems to permit visual examination for evidence of leakage. Therefore, during each refueling outage, certified VT-2 inspectors perform visual exams on all the bolted connections on the RCS. These exams are performed with the insulation removed.

Question:

3. *Describe the technical basis for the extent and frequency of walk downs 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 walkdowns performed to detect evidence of leakage from borated systems are performed at the start and end of every refueling outage in accordance with ASME Section XI. The walkdowns cover the RCS pressure boundary, with specific components also identified for inspection in the applicable plant procedures, as well as borated systems beyond the scope of ASME Section XI. There are no inaccessible areas during refueling outages that inhibit the performance of visual examinations for the purpose of identifying leakage.

These outage walkdowns are performed by a team consisting, as a minimum, of the Boron Corrosion Program Engineer, VT-2 qualified inspectors, and a Radiation Control technician. The VT-2 qualified inspectors receive training in accordance with EPRI's "Visual Examination for Leakage of PWR Reactor Head Penetrations." This document provides additional guidance on performing effective visual examinations (VT-2) to detect and characterize boron deposits.

In addition to these outage walkdowns, certified VT-2 inspectors perform visual exams on the bolted connections in the RCS in accordance with EST-227, as required by the ASME Code, Section XI, during each refueling outage. These exams are performed with the insulation removed. The ASME Code-required RCS system leak test is performed during plant shutdown and startup (Mode 3) by certified visual inspectors. The scope of the inspection boundary includes the RCS. The HNP Corrective Action Program (CAP) is used to document, track, investigate, and correct adverse conditions identified during these exams, as well as all walkdowns.

In addition, walkdowns are performed during forced outages in accordance with OPT-1519 and PLP-600. These walkdowns are currently performed at the discretion of plant management and the Boron Corrosion Program Engineer. The decision to perform a walkdown during a forced outage is made based on such considerations as the time since the previous walkdown, indications of possible leakage, the duration of the forced outage, and recent operation experience. The extent of these walkdowns is dependent upon plant conditions. These walkdowns exceed ASME Code Section XI requirements.

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RCS leakage is tracked by calculations performed under Operations Surveillance Test, OST-1026, *Reactor Coolant System Leakage Evaluation, Computer Calculation, Daily Interval, Modes 1-2-3-4.* Leak rates significantly lower than the Technical Specifications (TS) limit can be identified and trended. If significant leakage or an increasing trend is identified, both OST-1026 and PLP-600 require that action be initiated to investigate the cause of the leakage in order to identify the source and take corrective action.

As described in the HNP Final Safety Analysis Report (FSAR), the following leak detection parameters, mechanisms or systems provide the ability to continuously monitor the Reactor Coolant Pressure Boundary (RCPB) leakage from the HNP main control room by the observation of variations from normal conditions:

- Reactor coolant drain tank level
- Pressurizer relief tank level
- Accumulator pressure and level indications
- Air particulate and noble gas monitors
- Containment sump level monitors
- The Component Cooling Water Radioactivity Monitoring System
- Increasing charging pump flow rate compared with reactor coolant system inventory changes
- Unscheduled increases in reactor makeup water usage
- Containment airborne particulate monitors
- Gaseous radioactivity monitors.

Question:

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 were 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:

The goal of the Boron Corrosion Program is to have no leaks left in service. As identified in the recent assessment of the HNP Boron Corrosion Program, HNP has historically demonstrated a low tolerance for boric acid leaks. Such leaks are corrected at the first available opportunity. The process for identifying and dispositioning boric acid leaks is described below.

<u>Leak Identification</u>
Visual exams of the Class 1 pressure boundary are conducted under EST-227 during scheduled refueling outages. Outage boric acid walkdowns are conducted under OPT-1519 and PLP-600. The Boron Corrosion Program is controlled under PLP-600, which provides requirements for identifying, evaluating, and dispositioning boric acid leaks however they are identified.

If a leak is identified, PLP-600 requires that a Work Order be initiated to clean and inspect the component. Work Control Manual, WCM-002, provides a flowchart to ensure that the leak and any resulting damage are addressed properly. This includes notification of the appropriate System Engineer and the Boron Corrosion Program Engineer. Nuclear Condition Reports are generated per CAP-NGGC-0200, *Corrective Action Program*, for significant boric acid leaks that could potentially cause or have caused degradation of Reactor Coolant Pressure Boundary (RCPB) components.

Evaluation of Leaks

Bolted connections found to be leaking are addressed in accordance with PLP-652 and EST-227. This procedure requires that corrective actions be taken in accordance with IWA-5250, as modified by Relief Request 2RG-009. Consistent with the guidance contained in EPRI Report TR 1000975, *Boric Acid Corrosion Guidebook*, the evaluations must consider:

- Location of leakage
- History of leakage
- Fastener material
- Evidence of corrosion with component assembled
- Corrosiveness of the process fluid
- Other components within the vicinity that may be degraded due to the leakage.

If the evaluation determines that the leaking condition has not degraded the fasteners, then no further action is necessary. Reasonable attempts to stop the leakage are then taken. Also, any leakage that may affect system operability will be quantified, and dispositioned per AP-618, *Operability Determinations*.

If the evaluation indicates the need for further investigation, or no evaluation is performed, then a bolt closest to the source of leakage shall be removed. The bolt receives a VT-1 examination and is evaluated for corrosion in accordance with IWA-3100 (a) and dispositioned in accordance with IWB-3140. When the removed bolt shows evidence of rejectable degradation, all remaining bolts are removed and receive a VT-1 examination and evaluation in accordance with IWB-3140. If the leakage is identified when the bolted connection is in service, the removal of the bolt for VT-1 examination may be deferred to the next refueling outage if justified by evaluation.

This evaluation may impose inspection/monitoring requirements as appropriate to ensure that the corrective action taken is adequate. The imposition of such requirements is an engineering decision based on consideration of the factors listed above. Long-term corrective action may include replacement of a component with a more corrosion resistant material.

If observed leakage is determined to be unacceptable, the degraded component is repaired or replaced in accordance with HNP's ASME Repair and Replacement Program (PLP-605), "ASME Boiler and Pressure Vessel Code Section XI Repair and Replacement Program."

These requirements ensure that all ASME Code and regulatory requirements are met and the integrity of the Reactor Coolant Pressure Boundary is protected.

Question:

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 have the potential for causing boric acid corrosion. The NRC has had a concern with the 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:

There has been no operating experience which would indicate that leakage from through-wall cracking in the bottom reactor pressure vessel head incore instrumentation nozzles is an industry issue at this time. Therefore, the Code-required inspections have not been augmented. As indicated in Table A, these examinations are performed every refueling outage, in accordance with the ASME Code. The industry is evaluating Alloy 600 issues and we will incorporate their recommendations as appropriate for HNP.

Evidence of leakage from through-wall cracking in the bottom reactor pressure vessel head incore instrumentation nozzles would be detected by visual examinations which are performed every refueling outage under EST-227. The examinations consist of visual examination of the insulation, the tubes outside the insulation, and the floor below the vessel for indications of leakage. Because these tubes are located on the downward face of the vessel, leakage that could result in significant corrosion is not likely to accumulate on the vessel itself. Also, evidence of leakage would be apparent on the insulation, tubes or floor. Therefore, it is concluded that these examinations are adequate to verify the possible existence of leakage from these locations, and any leakage would be identified before significant corrosion can occur. The need for future inspections will be determined based on careful consideration of industry standards (ASME Code, Section XI) and the recommendations of the EPRI Materials Reliability Program.

As discussed in the response to Question 3 above, RCS leakage is tracked by calculations performed under OST-1026 and is continuously monitored via the observation of key parameters, mechanisms, and systems. Leak rates significantly lower than the TS limit are identified and trended. If significant (i.e.; greater than 1 gallon per minute unidentified) leakage, or an increasing trend is identified, both OST-1026 and PLP-600 require that action be initiated to investigate the cause of the leakage in order to identify the source and take corrective action. In addition, RCPB leakage is continuously monitored in the control room. Therefore, low levels of leakage can be identified.

The Acceptance Standard provided within the 1989 Edition of the ASME B&PV Code for the referenced VT-2 visual examinations is identified as IWB-3522, which requires correction of pressure boundary leakage prior to continued service. HNP maintains procedures and programs to implement these requirements (PLP-600 and PLP-652). The acceptance criterion for these procedures is that no through-wall leakage exists. In the event that leakage is identified, corrective actions are taken in accordance with plant procedures and the ASME Code prior to continued plant operation. Any leakage that may affect system operability will be quantified, and dispositioned per AP-618, *Operability Determinations*. Plant procedures require that all evaluations of leakage consider the effect on components in the leak path.

Question

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 have 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:
As shown in Table A, no other small diameter nozzles consisting of Alloy 600 material in the RCS exist at HNP. However, other small diameter nozzles of other alloys are specifically included within the scope of boric acid walkdowns and pressure tests. As discussed in the responses to the above questions, the HNP Boron Corrosion Program provides detailed guidance for performing inspections and walkdowns to ensure that leakage is identified. The program is, therefore, capable of identifying leakage from these sources, thereby ensuring that leaks that are smaller than the allowable Technical Specification limit are identified before degradation that may challenge structural integrity occurs.

In addition, Reactor Coolant Pressure Boundary leakage is continuously monitored in the control room. If significant leakage or an increasing trend is identified, both OST-1026 and PLP-600 require that action be initiated to investigate the cause of the leakage in order to identify the source and take corrective action.

The Acceptance Standard provided within the 1989 Edition of the ASME B&PV Code for the referenced VT-2 visual examinations is identified as IWB-3522, which requires correction of pressure boundary leakage prior to continued service. HNP maintains procedures and programs to implement these requirements (PLP-600 and PLP-652). The acceptance criterion for these procedures is that no through-wall leakage exists. In the event that leakage is identified, corrective actions are taken in accordance with plant procedures and the ASME B&PV Code prior to continued plant operation. Any leakage that may affect system operability will be quantified, and dispositioned per AP-618, *Operability Determinations*. Plant procedures require that all evaluations of leakage consider the effect on components in the leak path.

Question:

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:
HNP has used the susceptibility model as described in MRP-44 and corresponding recommendations contained in MRP-75 for guidance in scheduling the inspection of the Reactor Vessel head. However,

the Westinghouse Owners Group Materials Committee has not created susceptibility models or performed consequence reviews for other Alloy 600 components. Therefore, all other examinations are performed in accordance with ASME Section XI Code requirements.

Question:

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:
At the request of the WOG, Westinghouse reviewed its databases and applicable communications to determine what recommendations Westinghouse had made to the owners of Westinghouse Nuclear Steam Supply Systems (NSSS) regarding visual inspections of Alloy 600/82/182 materials in the reactor coolant pressure boundary. This detailed review did not identify any Westinghouse recommendations on visual inspections of Alloy 600/82/182 locations in Westinghouse NSSS.

Question:

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, 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:

HNP has concluded that the inspections and evaluations described above comply with all applicable regulatory, ASME Code and TS requirements. The following discussion provides a description of how HNP satisfies these regulations and requirements.

<u>Compliance with 10 CFR 50.55a, "Codes and Standards"</u>

10 CFR 50.55a, "Codes and Standards," requires that inservice inspection and testing be performed in accordance with the requirements of the ASME B&PV Code, Section XI, "Inservice Inspection of Nuclear Plant Components." Section XI contains applicable rules for examination, evaluation, and repair of code class components, including the RCPB.

The HNP Second Ten-Year Inservice Inspection (ISI) Interval, which commenced on February 2, 1998, has been implemented in accordance with the ASME B&PV Code, 1989 Edition with no Addenda. Examination requirements are contained within Table IWB-2500-1, Examination Category B-E, "Pressure Retaining Partial Penetration Welds in Vessels," and B-P, "All Pressure Retaining Components." The required extent and frequency (once every 10 years) of Examination Category B-E is a VT-2 visual examination of 25% of the vessel nozzles from the external surface. The required extent and frequency (every refueling outage) of examination for Examination Category B-P is also a VT-2 visual examination of reactor vessel pressure retaining boundary.

The Acceptance Standard provided within the 1989 Edition of the Code for the referenced VT-2 visual examinations is identified as IWB-3522, which requires correction of pressure boundary leakage prior to continued service. HNP maintains procedures and programs to implement these requirements (PLP-600 and PLP-652). The acceptance criterion for these procedures is that no through-wall leakage exists. In the event that leakage is identified, corrective actions are taken in accordance with plant procedures and the ASME Code prior to continued plant operation.

HNP has performed inspections of the RCPB during previous refueling outages using volumetric, surface, and visual examination techniques. The visual examinations, as required by plant procedures, include both direct and indirect observation for leakage. Direct examinations are performed on bolted connections in the RCPB. Indirect inspection is performed through the observation of evidence of leakage; i.e., signs of boric acid accumulation. These visual inspections meet the requirements of Section XI Table IWB-2500-1. The visual inspections also meet the requirements of NRC Generic Letter 88-05, "Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components in PWR Plants."

If the VT-2 examinations detect the conditions described in IWB-3522.1, then corrective actions required would be taken in accordance with IWA-5250(b) ("Corrective Measures") and the HNP CAP in accordance with CAP-NGGC-0200. PLP-600 and PLP-652 require that corrective action be taken to repair boric acid leaks or evaluated to confirm that leaks left in service will not challenge the integrity of the RCPB. PLP-600 also requires that consideration be given to corrective actions that will prevent leak recurrence.

Compliance with Technical Specifications

10 CFR 50.36, "Technical Specifications," provides requirements for Technical Specifications (TS) for licenses associated with production and utilization facilities. 10 CFR 50.36(c)(2) provides requirements specific to "Limiting Conditions for Operation," and 10 CFR 50.36(c)(3) provides requirements relative to "Surveillance Requirements." The HNP Operating Licensing and TS were developed and approved in accordance with these requirements and provide Limiting Conditions for Operation (LCO), Action Statements, and Surveillance Requirements (SR) regarding the RCPB. The current HNP TS requirements, e.g., LCOs and SRs, are consistent with the requirements of 10 CFR 50.36 and specify actions to maintain plant operations within analysis and design limits.

HNP TS 3.4.6, "Reactor Coolant System Operational Leakage," provides criteria and limits regarding primary system leakage, including LCO 3.4.6.2, which prohibits RCS pressure boundary leakage. Verification that RCS operational leakage is within limits by performance of an RCS water inventory balance is performed at least once per 72 hours in accordance with SR 4.4.6.2.1.d. Should pressure boundary leakage exist, Condition "a" would be entered which requires the unit to be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

As noted in the paragraph above and in the response to Question 3, the RCS leakage detection systems provide the means to detect small levels of RCS leakage. An RCS leak of sufficient magnitude to be detected by on-line leak detection systems would be evaluated in accordance with TS requirements and the appropriate actions taken to ensure that further degradation of the RCPB does not ensue.

Visual inspections conducted during refueling outages provide the opportunity to access areas/components within the plant that are normally not accessible during plant operations. As discussed in the responses to Questions 1, 2 and 4, above, these inspections are conducted in a manner, which ensures that leakage is identified. Once identified, plant procedures ensure that conditions are properly evaluated and appropriate corrective actions are taken.

Table A: HNP Alloy 600 Components and Alloy 82/182 Welds

Component (Alloy 600 pressure boundary material and Alloy 82/182 welds)	Quantity	Inspection Techniques	Extent of Coverage	Frequency	Degree of Insulation Removal	Insulation Type
PZR Spray Nozzle Safe End Weld	1	UT/PT [1]	100%	Once per 10 years	100%	Fiberglass
PZR Surge Nozzle Safe End Weld	1	UT/PT [1]	100%	Once per 10 years	100%	Reflective
PZR Safety And Relief Nozzle Safe End Welds	4	UT/PT [1]	100%	Once per 10 years	100%	Fiberglass
RPV Nozzle to Reactor Coolant Piping Welds	6	UT/PT [1]	100%	Once per 10 years	100%	Reflective
RV CRDM Head Penetrations	65	VT-2	100%	Every refuel outage	Accessible under insulation	Reflective
RV CRDM Nozzle To Head J-Groove Weld	65	VT-2 [2]	100%	Every refuel outage	Accessible under insulation	Reflective
RV CRDM Welds	65	PT [1]	100%	10% of outer periphery every 10 yrs.	Not insulated	None
RV Core Support Pads (Lower)	4	VT-3	100%	Once per 10 years	Not insulated	None
RV Instrumentation Tubes	50	VT-3 (ID)	100%	Once per 10 years	ID not insulated	None (ID)
		VT-2 (OD)	100%	Every refuel outage	OD not removed	Reflective (OD)
RV Instrumentation Tube J Welds [4]	50	VT-3 (ID)	100%	Once per 10 years	ID not insulated	None (ID)
		VT-2 (OD)	100%	Every refuel outage	OD not removed	Reflective (OD)
RV Instrumentation Tube Weld Overlay (OD)[3]	50	VT-2	100%	Every refuel outage	Not removed	Reflective
RV Vent Pipe	1	VT-2	100%	Every refuel outage	Accessible under insulation	Reflective
RV Vent Pipe Weld	1	VT-2	100%	Every refuel outage	Accessible under insulation	Reflective

[1] In addition to the inspection shown, a VT-2 is also performed every refueling outage. Insulation is not required to be removed to perform VT-2 inspections.
[2] This VT-2 is performed on the outer surface of the head.
[3] The weld overlay is used to reinforce the reactor vessel bottom head.
[4] There are 0.01" gaps between the tubes and the bottom RV head, and the tubes and the instrumentation tube weld overlay, which enable visual inspection for leakage of the instrumentation tube J-groove welds to be performed.

Abbreviations:
PZR = Pressurizer
RV = Reactor Vessel
VT = Visual Testing
RPV = Reactor Pressure Vessel
CRDM = Control Rod Drive Mechanism
ID = Inside Vessel
UT = Ultrasonic Testing
PT = Penetrant Testing
OD = Outside Vessel