SERIAL: HNP-02-052
10CFR50.54(f)

United States Nuclear Regulatory Commission
ATTENTION: Document Control Desk
Washington, DC 20555

SHEARON HARRIS NUCLEAR POWER PLANT
DOCKET NO. 50-400/LICENSE NO. NPF-63
15-DAY RESPONSE TO NRC BULLETIN 2002-01, REACTOR PRESSURE VESSEL HEAD DEGRADATION AND REACTOR COOLANT PRESSURE BOUNDARY INTEGRITY

Dear Sir or Madam:

By the letter dated March 18, 2002, the U. S. Nuclear Regulatory Commission (NRC) issued NRC Bulletin 2002-01, "Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity". The Bulletin directs addressees to submit: (1) information related to the integrity of the reactor coolant pressure boundary including the reactor pressure vessel head and the extent to which inspections have been undertaken to satisfy applicable regulatory requirements, and (2) the basis for concluding that plants satisfy applicable regulatory requirements related to the structural integrity of the reactor coolant pressure boundary and future inspections will ensure continued compliance with applicable regulatory requirements, and (3) a written response to the NRC in accordance with the provisions of Title 10, Section 50.54(f), of the Code of Federal Regulations (10 CFR 50.54(f)) if they are unable to provide the information or they cannot meet the requested completion dates.

Enclosure 1 to this letter provides Carolina Power & Light Company's (CP&L) response to this Bulletin for the Harris Nuclear Plant (HNP). The Harris Nuclear Plant response to the bulletin provides reasonable assurance that plant inspection and maintenance programs are adequate to prevent degradation as observed at the Davis-Besse Plant. Harris Nuclear Plant is considered to be in the NRC category of plants with low susceptibility (greater than 30 effective full power years of operation relative to Oconee 3). In addition, HNP has not previously identified either leakage from or cracking in Vessel Head Penetration (VHP) nozzles.

Please refer any questions regarding this submittal to Mr. John Caves at (919) 362-3137.

Sincerely,

RTG/rtg

Enclosure

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 Carolina Power & Light Company.

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Notary (Seal)

My commission expires:

c: Mr. J. B. Brady, NRC Sr. Resident Inspector
Mr. Mel Fry, Director, N.C. DENR
Mr. J. M. Goshen, NRC Project Manager
Mr. L. A. Reyes, NRC Regional Administrator

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The Harris Nuclear Plant, HNP, was licensed for commercial operation in 1987, at about the same time as the nuclear industry's awareness was heightened regarding the concerns addressed in NRC Generic Letter 88-05, "Boric Acid Corrosion of Carbon Steel Reactor Boundary Components in PWR Plants". HNP committed to the establishment of a formal program governing the control of boric acid corrosion in response to NRC Generic Letter 88-05 provided by CP&L, dated May 27, 1988, serial number NLS-88-110. Inspections of susceptible components are directed by a program, which is described in Plant Programs procedure PLP-600, "Boron Corrosion Program". PLP-600 states that the boron corrosion program has been implemented at HNP to "prevent boric acid attack of pressure boundary components and equipment important to safety." The program is based on identifying, evaluating, and repairing borated water leaks and the effects of these leaks from all sources that could result in a boric acid corrosion problem. This approach includes the use of high standards for cleanliness to leave the metal clean of corrosives including the cleaning of the affected components, and using the appropriate processes and qualified people to execute the program.

The Harris Nuclear Plant has been analyzed for susceptibility relative to Oconee 3 using the time-at-temperature model and plant-specific input data reported in EPRI's Material Reliability Project, MRP-2001-48. This evaluation showed that it would take HNP 115.5 effective full power years (EFPY) of additional operation from March 1, 2001, to reach the same time at temperature as Oconee 3 at the time that leaking nozzles were discovered in March 2001. Harris Nuclear Plant falls into the NRC category of plants with low susceptibility (greater than 30 effective full power years of operation relative to Oconee 3).

The following sections include the HNP responses to the specific items as required by NRC Bulletin 2002-01:

<u>NRC Item 1.A:</u>

Provide a summary of the reactor pressure vessel head inspection and maintenance programs that have been implemented at your plant.

<u>HNP Response for Item 1.A:</u>

The Harris Nuclear Plant has two principle elements to the overall boron corrosion program for inspection, documentation, and resolution of borated water leaks or boric acid build-up on the reactor pressure vessel, RPV, head. They are PLP-600 (described above) and Engineering Surveillance Test procedure EST-227 (ASME Section XI Class 1 System Pressure Test).

The purpose of PLP-600 is to address the concerns identified in Generic Letter 88-05. This program is based on walkdown inspections during shutdown outages, inspection during maintenance activities, trending the daily reactor coolant system leakage evaluations, and monitoring for leakage during power operations.

Three plant procedures implement PLP-600 requirements. Operations Periodic Test procedure (OPT-1519) requires a visual inspection of the pressure boundary components inside containment building prior to cooldown for every Refueling Outage (RFO). Corrective Maintenance procedure (CM-M0070) requires inspection of the Control Rod Drive Mechanism (CRDM) area for any evidence of leakage. Operations Surveillance Test procedure (OST-1026) provides for a daily reactor coolant system evaluation for leakage. Corrective actions are taken to repair any identified borated water leakage in accordance with Work Package Planning procedure (WCM-002) and ASME Section XI requirements. Incidents of borated water leakage onto the reactor vessel head are documented in the Corrective Action Program.

In light of recent industry events, a bare metal visual examination was performed on the accessible portion of the Reactor Pressure Vessel (RPV) head and CRDM penetrations. Qualified Visual Testing (VT-2) examiners using approved plant procedures performed the examinations. The VT-2 examiners had been provided specific training regarding CRDM leakage. This training followed the training guideline, "Visual Examination for Leakage of Reactor Head Penetrations On Top Of Head" provided by Electric Power Research Institute (EPRI). The inspections of the "uphill" portions of the CRDM penetrations were limited to what could be seen from below and from the side. A circular area approximately 3 feet in diameter was not readily accessible for direct visual examination without the use of special tools or insulation removal. A 100% bare metal inspection has not been conducted at HNP. However, during RFO-10 approximately 85% of the reactor vessel head penetration nozzles were examined.

The purpose of EST-227 is to fulfill the pressure test requirements for Class 1 pressure retaining components (including RPV head penetrations) in accordance with the 1989 Edition of ASME Boiler and Pressure Vessel Code Section XI. The acceptance criteria and inspection requirements of ASME XI code are used to disposition any relevant indications. EST-227 is performed at the end of every refueling outage.

<u>NRC Item 1.B:</u>

Provide an evaluation of the ability of your inspection and maintenance programs to identify degradation of the reactor pressure vessel head including, thinning, pitting, or other forms of degradation such as the degradation of the reactor pressure vessel head observed at Davis-Besse.

<u>HNP Response for Item 1.B:</u>

The HNP programs and procedures for RPV head inspection and maintenance, as summarized in Item 1A above, are appropriate and provide assurance that degradation of the RPV head, including thinning, pitting, or other forms of degradation, will be identified and corrected.

Plant procedures and surveillances, summarized in 1A above, prescribe the actions necessary to both inspect and disposition borated water system leakage and any resultant corrosion of primary pressure boundary components. These procedures and surveillances, which include the programmatic implementation of NRC Generic Letter 88-05 via PLP-600, provide a framework for the systematic monitoring of locations where boric acid leakage could occur, and measures to prevent the degradation of the RCS pressure boundary by boric acid corrosion.

Certified VT-2 personnel using qualified plant procedures perform the visual inspections. Inspection of the RPV head, including the CRDMs and mechanical connections, are specific inspection items identified in EST-227 and CM-M0070. Any evidence of boric acid leakage (active or inactive) found during inspections, operator/system engineer walkdowns, or maintenance activities require evaluation. The evaluation consists of the following:

A. Identifying the source of leakage
B. Determining if leakage is active or inactive (with sensitivity to the leakage around CRDM penetrations)
C. Decontaminating (removing & cleaning) the boron.
D. Inspecting the component to identify degradation. If degradation is found, additional examinations (surface or volumetric) may be necessary to quantify the extent of damage.

The adequacy of these inspection and maintenance programs is evidenced by successful detection of evidence of leakage during RFO-08 (10/98-11/98) and RFO-10 (09/01-01/02) as described in Item 1C.

<u>NRC Item 1.C:</u>

Provide a description of any conditions identified (chemical deposits, head degradation) through the inspection and maintenance programs described in 1.A that could have led to degradation and the corrective actions taken to address such conditions.

<u>CP&L Response for Item 1.C:</u>

No boric acid deposits or head degradation have been found due to RPV head penetration CRDM nozzle leakage at HNP. Borated water leakage involving the RPV head area has been identified at HNP in the following instances:

During RFO-08 (10/98-11/98), boric acid crystals indicative of a leaking CRDM lower canopy seal weld were discovered while performing CM-M0070. This was documented via the Corrective Action Program. Corrective actions were taken to remove the boric acid deposits from the reactor vessel head. Inspections indicated that there was no reactor vessel head degradation. A weld repair of the leaking CRDM seal weld was performed to prevent future leakage, which was effective as validated by subsequent refueling outage inspections. No evidence of leakage was detected on remaining canopy seal welds.

During RFO-10 (09/01-01/02), boric acid crystals indicative of a leaking thermocouple port column conoseal connection were discovered while performing CM-M0070. This boric acid crystal deposit was estimated to be less than 1 cubic inch. Corrective actions were taken to repair the leaking mechanical conoseal joint. The removal and cleaning of boric acid deposits, inspections and corrective actions to prevent future leakage were captured within the Corrective Action Program. No degradation of the head material was detected. Inspections performed during start-up from RFO-10 verified that the corrective actions were effective, and the conoseal joint was not leaking at operating temperature and pressure.

However, during the plant start-up from RFO-10, minor leakage (approximately 3 teaspoons in 5 hours) was detected at another conoseal connection. The plant was cooled down and depressurized to repair the leaking conoseal. The condition was documented via the Corrective Action Program. Corrective actions were taken to remove the boric acid deposits and to repair the conoseal connection. No reactor vessel head degradation was observed. Inspections performed at normal plant operating temperature and pressure verified that the corrective actions had stopped the leakage.

NRC Item 1.D:

Provide your schedule, plans, and basis for future inspections of the reactor pressure vessel head and penetration nozzles. This should include the inspection methods(s) scope, frequency, qualification requirements, and acceptance criteria.

CP&L Response for Item 1.D:

HNP will continue to meet ASME Section XI (EST-227) and Boron Corrosion Program (PLP-600) requirements for RPV head penetrations.

HNP's next refueling outage is scheduled for April 2003. HNP currently performs VT-2 examination of the accessible portion (bare metal) of reactor pressure vessel head and CRDM penetrations. This VT-2 examination will continue to be performed every refueling outage as prescribed by ASME XI code. ASME XI code will be the basis for examination/personnel qualification and acceptance criteria.

In light of recent industry events, HNP will perform a 100% bare metal inspection of its RPV head and CRDM penetrations during the next refueling outage (RFO-11). Future examinations (100% bare metal) may be scheduled based upon industry experience (root cause from Davis-Besse), improved remote examination methods, and Harris site-specific experience.

NRC Item 1.E:

Provide your conclusion regarding whether there is reasonable assurance that regulatory requirements are currently being met (see the Applicable Regulatory Requirements, above). This discussion should also explain your basis for concluding that the inspections discussed in response to Item 1.D will provide reasonable assurance that these regulatory requirements will continue to be met. Include the following specific information in this discussion:

(1) *If the evaluation does not support the conclusion that there is reasonable assurance that regulatory requirements are being met, discuss plans for plant shutdown and inspection.*

(2) *If the evaluation supports the conclusion that there is reasonable assurance that regulatory requirements are being met, provide your basis for concluding that all regulatory requirements discussed in the Applicable Regulatory Requirements section will continue to be met until the inspections are performed.*

HNP Response on Applicable Regulatory Requirements in NRC Item 1.E

The "Applicable Regulatory Requirements" identified within NRC Bulletin 2002-01 are as follows:

- 10 CFR 50, Appendix A, "General Design Criteria for Nuclear Power Plants," including the following:
    - GDC 14, "Reactor Coolant Pressure Boundary"
    - GDC 31, "Fracture Prevention of Reactor Coolant Boundary"
    - GDC 32, "Inspection of Reactor Pressure Coolant Pressure Boundary"
- 10 CFR 50.55a, "Codes and Standards"
- 10 CFR 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants,"
    - Criterion V, "Instructions, Procedures, and Drawings"
    - Criterion IX, "Control of Special Processes"
    - Criterion XVI, "Corrective Action"
- Technical Specifications

HNP has concluded there is reasonable assurance that regulatory requirements are currently being met. The following provides a description of how HNP satisfies these regulations and requirements, and how continued compliance will be maintained.

General Design Criteria

The General Design Criteria (GDC) in existence at the time HNP was licensed for operation (January 1987) were contained in the Appendix A to 10 CFR 50, "General Design Criteria for Nuclear Power Plants," published in the Federal Register. HNP conformance with these GDC is described within Final Safety Analysis Report (FSAR) Section 3.1, "Conformance with NRC General Design Criteria." Applicability of these GDC to NRC Bulletin 2002-01 is discussed below.

The HNP design criteria meets the current GDC 14. This GDC states the following:

"The reactor coolant pressure boundary shall be designed, fabricated, erected and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture."

A discussion of HNP compliance with GDC 14 is provided within FSAR Section 3.1.10.

The reactor coolant pressure boundary (RCPB) is designed to accommodate the system pressures and temperatures attained under all expected modes of Unit operation, including all anticipated transients, and to maintain the stresses within applicable stress limits.

RCPB materials selection and fabrication techniques ensure a low probability of gross rupture or significant leakage.

In addition to the loads imposed on the system under normal operating conditions, consideration is also given to abnormal loading conditions, such as seismic and pipe rupture.

The system is protected from overpressure by means of pressure relieving devices, as required by applicable codes.

The RCPB has provisions for inspection, testing and surveillance of critical areas to assess the structural and leaktight integrity. For the reactor vessel, a material surveillance program conforming to applicable codes is provided.

Previous visual examinations of the HNP reactor vessel head have not identified VHP nozzle leakage. Based on the above, and industry experience to-date regarding the low levels of primary system leakage resulting from VHP nozzle leakage in plants in the low susceptibility category, HNP remains in compliance with the reactor coolant pressure boundary design criteria as set forth within GDC 14.

The HNP design criteria meets the current GDC 31. This GDC states the following:

"The reactor coolant pressure boundary shall be designed with sufficient margin to assure that when stressed under operating, maintenance, testing, and postulated accident conditions (1) the boundary behaves in a non-brittle manner and (2) the probability of rapidly propagating fracture is minimized. The design shall reflect consideration of service temperatures and other conditions of the boundary material under operating, maintenance, testing, and postulated accident conditions and the uncertainties in determining (1) material properties, (2) the effects of irradiation on material properties, (3) residual, steady state and transient stresses, and (4) size of flaws."

A discussion of HNP compliance with GDC 31 is provided within FSAR, Section 3.1.27.

Close control is maintained over material selection and fabrication for the RCS to assure that the boundary behaves in a nonbrittle manner. Materials for the RCS, which are exposed to the coolant, are corrosion-resistant stainless steel or Inconel. The reference temperature (RTNDT) of the reactor vessel structural steel is established by Charpy V-notch and drop-weight tests in accordance with 10CFR50, Appendix G.

As part of the reactor vessel specification, certain requirements, which are not specified by the applicable ASME Codes are performed as follows:

1. Ultrasonic Testing - Requirements for additional ultrasonic testing.

2. Radiation Surveillance Program - In the surveillance programs, the evaluation of the radiation damage is based on pre-irradiation testing of Charpy V-notch and tensile specimens and post-irradiation testing of Charpy V-notch and tensile 1/2 T (thickness) impact/tension fracture mechanics specimens. These programs are directed toward evaluation of the effect of radiation on the fracture toughness of reactor vessel steels based on the reference transition temperature approach and the fracture mechanics approach, and are in accordance with the American Society for Testing and Materials E 185-82, "Standard Practice for Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels E-706 (IF)", and the requirements of 10CFR50, Appendix H.

3. Reactor vessel core region material chemistry (copper, phosphorous and vanadium) is controlled to reduce sensitivity to embrittlement due to irradiation over the life of the plant.

The fabrication and quality control techniques used in the fabrication of the RCS are equivalent to those used for the reactor vessel. The inspections of reactor vessel, pressurizer, piping, pumps, and steam generator are governed by ASME Code requirements.

Allowable pressure/temperature relationships for plant heatup and cooldown rates are calculated using methods presented in the ASME Code, Section III Appendix G, "Protection Against Non-Ductile Failure." The approach specifies that allowed stress intensity factors for vessel level A and B service limits and hydrostatic tests shall not exceed the reference stress intensity factor (KIR) for the metal temperature at any time. Operating specifications include conservative margins for predicted changes in the material reference temperature (RTNDT) due to irradiation.

Previous visual examinations of the HNP reactor vessel head have not identified VHP nozzle leakage. Based on the above information and industry experience to-date regarding flaw development and propagation in VHP nozzles, HNP, remains in compliance with GDC 31 regarding rapidly propagating type failures of the reactor coolant pressure boundary.

The HNP design criteria meets the current GDC 32. This GDC states the following:

“Components which are part of the reactor coolant pressure boundary shall be designed to permit (1) periodic inspection and testing of important areas and features to assess their structural and leaktight integrity, and (2) an appropriate material surveillance program for the reactor pressure vessel.”

A discussion of HNP compliance with GDC 32 is provided within FSAR, Section 3.1.28.

The design of the RCPB provides the capability for accessibility during service life to the entire internal surfaces of the reactor vessel, certain external zones of the vessel including the top and bottom heads, and external surfaces of the reactor coolant piping except for the area of pipe within the primary shielding concrete. The inspection capability complements the Leak Detection System in assessing the RCPB components' integrity. The RCPB, as defined by 10CFR50.2(v) and 10CFR50.55a footnote 2, will be periodically inspected under the provisions of the ASME Code, Section XI for Operations Quality Group A requirements.

Monitoring of changes in the fracture toughness properties of the reactor vessel core region plates, forgings, weldments and associated heat-affected zones are performed in accordance with 10CFR50, Appendix H, "Reactor Vessel Material Surveillance Program Requirements," and E 185-82, "Standard Practice for Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels E-706(IF)." Samples of reactor vessel plate materials are retained and catalogued in the event future engineering development shows the need for further testing.

The material properties surveillance program includes not only the conventional tensile and impact tests, but also fracture mechanics specimens. The observed shifts in RTNDT of the core region materials with irradiation will be used to confirm the allowable limits calculated for all operational transients.

In addition to the design elements discussed above, the visual examination of the accessible portion of the HNP reactor vessel head during RFO-10 (09/01-01/02) provides an additional measure of assurance regarding VHP nozzle integrity until the next scheduled visual examinations are performed during RFO-11. It is reasonable to expect that leakage into the annulus area above the J-groove weld would have resulted in boric acid deposition on the reactor vessel head.

10 CFR 50.55a, Codes and Standards

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 RCS pressure boundary.

The HNP 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 applicable to VHP nozzles 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. Since the reactor vessel head is insulated and the VHP nozzles do not represent a bolted connection, Article IWA-5000, “System Pressure Tests,” subsection IWA-5242, “Insulated Components,” permits these inspections to be performed without removal of insulation.

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.

As described under Item 1.D above, HNP has and maintains procedures and programs to implement ASME Code requirements relative to VHP nozzles. The acceptance criterion for these procedures is that no through-wall leakage exists. No VHP nozzle leakage has been identified during previous reactor vessel head examinations. In the event that VHP nozzle leakage is identified during future examinations, corrective actions will be taken in accordance with plant procedures and the ASME Code prior to continued plant operation.

As previously noted, a visual examination of the reactor vessel head was performed during RFO-10 (9/01-1/02). A 100% bare metal visual examination is planned for RFO-11 (April 2003).

10 CFR 50, Appendix B

NRC Bulletin 2002-01 identified the following Criteria of 10 CFR 50, Appendix B, as being applicable to VHP nozzle degradation and leakage:

- Criterion V, "Instructions, Procedures, and Drawings"
- Criterion IX, "Control of Special Processes"
- Criterion XVI, "Corrective Action"

HNP, has and maintains the required instructions, procedures, and drawings for special processes and activities affecting quality to satisfy the requirements of 10 CFR 50, Appendix B, Criterion V and IX. As an additional action to assure the integrity of VHP nozzles, HNP intends to perform a 100% bare metal examination of the reactor vessel head during RFO-11. The scope of this examination will include each of the VHP nozzles. Examinations or special processes performed during RFO-11 will be implemented using appropriate instructions, procedures, or drawings in accordance with Criterion V and IX.

10 CFR 50, Appendix B, Criterion XVI, requires that measures be established to assure that conditions adverse to quality are promptly identified and corrected. Additionally, significant conditions adverse to quality will have the cause determined and corrective actions taken to preclude repetition. HNP has and maintains programs and procedures to satisfy the requirements of Criterion XVI. As described under Item 1.C above, previous inspection activity has not identified VHP nozzle leakage. As further noted under Item 1.D above, HNP will perform a visual examination of the reactor vessel head during RFO-11 (April 2003). Additionally, HNP will monitor the results of VHP inspections performed by other utilities, and the results of industry-sponsored efforts to better understand the contributors to and potential effects of primary water stress corrosion cracking of VHP nozzles. Industry efforts will also be monitored relative to the development and demonstration of reliable NDE techniques for examination of VHP nozzle penetrations. Plans for future reactor vessel head inspections may be modified, where appropriate, to incorporate "lessons learned" from other utilities and to assure that proposed inspection techniques will produce accurate and reliable results. These actions are consistent with 10 CFR 50, Appendix B, Criterion XVI, and with the discussion of Criterion XVI provided within NRC Bulletin 2002-01.

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 RCS pressure boundary.

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. 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. 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.

As noted above under the General Design Criteria discussion, and as indicated within the HNP TS Bases for LCO 3.4.6, the RCS leakage detection systems provide the means to detect RCS leakage to the extent practical. Industry experience from VHP nozzle leakage has shown that the associated primary system leakage can be well below TS limits and the sensitivity of on-line leakage detection systems. 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. 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.

<u>CP&L Response Summary for Item 1.E:</u>

HNP was not required by the NRC Bulletin 2001-01 to perform any examinations of the vessel head penetrations due to the Unit’s relative time at temperature. However, during RFO-10, which was competed on 01/03/02, HNP performed a visual inspection (VT-2 of bare metal) of the accessible portions of the reactor pressure vessel head and CRDM penetrations.

Item 1.E (1):

For Item 1.E (1) there is reasonable assurance that the regulatory requirements are met at HNP so this item is not applicable to HNP.

Item 1.E (2)

There is a high confidence level of no external corrosion for the following reasons:

A. There was no evidence of vessel head penetration nozzle leakage from the inspection performed during RFO-10 where a significant portion of the head was inspected specifically for indications of leakage. Harris Nuclear Plant is considered to be in the NRC category of plants with low susceptibility to CRDM penetration nozzle cracking (greater than 30 effective full power years of operation relative to Oconee 3) as reported in MRP 2001-48.

B. Previous boron deposits from canopy seal weld leaks and conoseal leaks were cleaned up at the time of discovery, and the surrounding area examined for residual boron and wastage. Boron was removed from the RPV head and no wastage was observed. All past leakage sites were inspected in RFO-10.

C. During start-up from RFO-10, Quality Control (QC) personnel performed a VT-2 inspection of mechanical seals above the reactor vessel head at normal operation pressure and temperature (after 4 hour hold time) to verify that no RCS leakage was present.

D. HNP's Boric Acid Program requires that any Boric Acid detected must be cleaned up.

E. Qualified VT-2 examiners, using approved plant procedures, performed the examinations. The VT-2 examiners were provided specific training regarding CRDM leakage. This training followed the training guideline "Visual Examination for Leakage of Reactor Head Penetrations On Top Of Head" provided by EPRI.

<u>NRC Item 2:</u>

Within 30 days after plant restart following the next inspection of the reactor pressure vessel head to identify any degradation, all PWR addressees are required to submit to the NRC the following information:

A. The inspection scope (if different than that provided in response to Item 1.D) and results, including the location, size, and nature of any degradation detected.

B. The corrective actions taken and the root cause of the degradation.

<u>CP&L Response for Item 2:</u>

HNP will provide the requested information within 30 days following restart from the next scheduled refueling outage, i.e., RFO-11, which is currently scheduled to begin in April 2003.

<u>NRC Item 3:</u>

Within 60 days of the date of the bulletin, all PWR addressees are required to submit to the NRC the following information related to the remainder of the reactor coolant pressure boundary:

A. The basis for concluding that the boric acid inspection program is providing reasonable assurance of compliance with the applicable regulatory requirements discussed in Generic Letter 88-05 and this bulletin. If a documented basis does not exist, provide plans, if any, for a review of these programs.

CP&L Response for Item 3:

HNP will provide the requested information within 60 days of the date of the bulletin.

References

1. *PWR Materials Reliability Program Response to NRC Bulletin 2001-01 (MRP-48),* EPRI, Palo Alto, CA: August 2001. TP-1006284.

2. *PWR Materials Reliability Project Interim Alloy 600 Safety Assessments for US PWR Plants (MRP-44),* EPRI, Palo Alto, CA: May 2001. TP-1001491.

3. CP&L Letter to NRC dated July 29, 1997, Serial HNP-97-152, Response to NRC Generic Letter 97-01, "Degradation of Control Rod Drive Mechanism Nozzle and Other Vessel Closure Head Penetrations".

4. CP&L Letter to NRC dated May 27, 1988, Serial NLS-88-110, Response to NRC Generic Letter 88-05, "Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components in PWR Plants.