CP&L
A Progress Energy Company

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SERIAL: HNP-02-063
10CFR50.54(f)

MAY 1 5 2002

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
60-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". In addition to information required within fifteen days the Bulletin directs addressees 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 Harris Nuclear Plant (HNP)] 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 your [HNP] plans, if any, for a review of your [HNP] programs."

Enclosure 1 to this letter provides Carolina Power & Light Company's (CP&L) response to this Bulletin for the Harris Nuclear Plant. The Harris Nuclear Plant has reviewed the facility boron corrosion control program and has determined that the program meets or exceeds the standards necessary to assure the integrity of the reactor coolant system pressure boundary, based on regular inspections and cleanliness requirements for any boric acid deposits due to system leakage. This response letter confirms with reasonable assurance that plant inspection and maintenance programs are adequate to prevent degradation due to boron corrosion as discussed in GL 88-05 and Bulletin 2002-01.

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

Sincerely,

James Scarola

RTG/rtg

Enclosure

A095

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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.

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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NRC Bulletin 2002-01 directs that within sixty days the Harris Nuclear Plant (HNP) will respond to the NRC. The NRC Bulletin wording under "Required Information" item 3 describing this requirement is as follows:

"Within 60 days of the date of this 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 your plans, if any, for a review of your programs."

The Harris Nuclear Plant provides the following information in accordance with the bulletin:

A. Program Definition and Responsibility

The HNP programs and procedures for inspecting the reactor coolant system pressure boundary are appropriate and provide assurance that degradation of the system components, including thinning, pitting, or other forms of degradation, will be identified and corrected.

Plant procedures and surveillances 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, "Boron Corrosion Program," provide a framework for the systematic monitoring of locations where borated water leakage could occur, and measures to prevent the degradation of the Reactor Coolant System (RCS) pressure boundary by boric acid corrosion.

The Harris Nuclear Plant 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. 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 boric acid corrosion. 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.

Boron Corrosion Program Description and Process

Operations Surveillance Test procedure, OST-1026, "Reactor Coolant System Leakage Evaluation, Computer Calculation, Daily Interval, Modes 1-2-3-4," provides for a daily Reactor Coolant System (RCS) evaluation for leakage (72 hour Technical Specification surveillance currently being performed daily). This procedure is intended to ensure detection of leakage volumes in excess of Technical Specification action limits, but has proven effective in detecting significantly smaller leakage rates through trending the results of daily evaluations.

In addition, plant functions are monitored by Health Physics and Operations personnel to detect leakage. Primary indications are provided by:

- Containment Air Particulate Monitor
- Containment Noble Gas Monitor
- Containment Sump Inleakage

Additional leak indications may be provided by Reactor Coolant Drain Tank (RCDT) level indications due to:

- Valve leak-offs
- Reactor Coolant Pump Seal Leakage
- Reactor flange leak-off

If significant leakage is suspected from trending, accessible areas are walked down to locate leakage.

Inspections to identify leaks, which result in boron deposits on plant Structures, Systems, and Components are conducted in accordance with the following plant procedures:

- Operations Periodic Test, OPT-1519, "Containment Visual Inspection for Boron and Evaluation of Containment Sump Inleakage Every Refueling Outage Shutdown, Mode 3"
- Engineering Surveillance Test, EST-227, "ASME Section XI Class 1 System Pressure Test"
- EST-201, "ASME System Pressure Tests"
- Corrective Maintenance-Mechanical Procedure, CM-M0070, "Reactor Vessel Mirror Insulation Disassembly/Reassembly Procedure"

These procedures may be invoked based on indications discovered during routine monitoring, as described above, and are also utilized during every plant refueling outage. Walkdowns may also be performed at the discretion of the Boron Corrosion Program Engineer. In addition, System Engineers responsible for RCS pressure boundary components periodically walk down their system components in accordance with plant Technical Support Management procedure, TMM-117, "System Walkdowns and Observations." During these system walkdown activities, if boric acid leaks are identified, actions are taken to correct the condition.

If a leak is identified during a walkdown conducted under one of these procedures, PLP-600 requires that a Work Order be initiated to clean and, if necessary, repair damage. Work Control Manual procedure, WCM-002, "Work Package Planning", provides a flowchart to ensure that the leak and any resulting damage is addressed properly. This includes notification of the appropriate System Engineer and the Boric Acid Program Engineer. An Engineering Evaluation is performed if significant corrosion is found which would challenge the integrity of the reactor coolant pressure boundary. Nuclear Condition Reports are generated per PLP-600 for significant boric acid leaks.

HNP maintains high standards for the material condition of plant components and strives to prevent unplanned equipment failures. Personnel working inside the containment and reactor auxiliary buildings maintain these standards by identifying locations of boric acid leaks and taking actions to correct the conditions via the work control or corrective action programs.

B. Inspection Scope and Frequency

As provided in the 15-Day required response for NRC Bulletin 2002-01, 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 coolant pressure boundary. They are PLP-600 (described above) and Engineering Surveillance Test procedure EST-227. In addition, EST-201 is performed to examine ASME Section XI Class 2 systems.

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, inspections during maintenance activities, trending the daily reactor coolant system leakage, and evaluations.

Three plant procedures implement PLP-600 requirements. OPT-1519 requires a visual inspection of the pressure boundary components inside containment building prior to cooldown for every Refueling Outage (RFO). CM-M0070 requires inspection of the Control Rod Drive Mechanism (CRDM) area for any evidence of leakage. OST-1026 provides for a reactor coolant system evaluation for leakage in accordance with the HNP Technical Specification surveillance requirements for a once per 72 hours surveillance interval. OST-1026 is currently performed daily. Corrective actions are taken to repair any identified borated water leakage in accordance with WCM-002 and ASME Section XI requirements.

The purpose of EST-227 and EST-201 is to fulfill the pressure test requirements for Class 1 & 2 pressure retaining components in accordance with the 1989 Edition of ASME Boiler and Pressure Vessel Code Section XI. EST-227 includes a list of pressure retaining boundary bolted connections and Class 1 components subject to examination. The acceptance criteria and inspection requirements of ASME Section XI code are used to disposition any relevant indications. EST-227 is performed every refueling outage. EST-201 is performed once per Inservice Inspection Period.

Additionally, the ISI program was updated to meet the requirements of the 1989 Edition of ASME, Section XI. This edition of the code requires the insulation be removed from bolted connections and visually examined for evidence of leakage. During each refuel outage, certified VT-2 inspectors perform visual exams on all the bolted connections on the RCS. This exam is performed with the insulation removed.

The adequacy of these inspection and maintenance programs is evidenced by successful detection of evidence of leakage during RFO-08 and RFO-10 as described in section D.

Inspection of Containment Air Handling Units

The HNP containment building is cooled by four safety related containment air handling units and three non-safety related containment air-handling units. Each of the units contains service water cooling coils with two fans. Two safety related air handling units, one for safety train A and one for safety train B, are located on the operating deck of the containment building at elevation 286'. The other two safety related air handling units, one for safety train A and one for safety train B, are located on elevation 236' of the containment building. The three non-safety related air handling units are located on elevation 221' of the containment building.

Safety related containment air handling units air side coils and fins are visually inspected during refueling outages, in accordance with Maintenance Periodic Test procedure, MPT-M0091, "Heat Exchanger Opening/Closure for NRC Generic Letter 89-13 Inspections," and instructions provided in maintenance work orders. Only one safety related containment air-handling unit train is inspected during a refueling outage. The other train is inspected the following refueling outage. If the airside coils and fins require cleaning, they are cleaned in accordance with MPT-M0091. Inspection results are recorded in the completed procedure, or the associated work order. The condition of air handling units' airside coils and fins have also been visually inspected in accordance with Engineering Periodic Test procedure, EPT-163, "Generic Letter 89-13 Inspections (Raw Water Systems and Local Area Air Handler Inspection and Documentations)." These inspections have not identified any evidence of boron deposits on the cooling coils or fins. Future inspections of the airside coils and fins per EPT-163 will be performed as required by the Generic Letter 89-13 Program or the system engineer.

The non-safety related containment air handling units air side coils and fins have been visually inspected and/or cleaned during past refueling outages. These units have also been visually inspected in accordance with EPT-163. Inspection results are recorded in the completed EPT-163 test procedure. The past inspections have not identified any evidence of boron deposits on the cooling coils or fins. The airside coils and fins for all three air-handling units were cleaned during RFO10 in accordance with instructions provided in maintenance work orders. These work orders did not identify any evidence of boron deposits on the airside coils or fins. These air-handling units will continue to be inspected. The Generic Letter 89-13 Program or the system engineer will determine the frequency of future inspections.

The operation of the safety related and non-safety related air-handling units can be remotely monitored. An indication of the safety related air-handling unit's performance is provided by the cooling coils exit air temperature, which is recorded on the Auxiliary Equipment Panel, AEP-2. The effectiveness of the non-safety related air-handling units can be monitored by recording the cooling coils exit air temperature on a HVAC temperature recorder, which is also located on the Auxiliary Equipment Panel AEP-2.

Radiation Monitor Filter

The HNP Radiation Monitoring System includes a safety related Containment Leak Detection Monitor, which provides an indication to Operations Personnel of the particulate and gaseous radioactivity levels inside the containment building. Radioactivity in containment indicates the presence of fission products due to a reactor coolant pressure boundary leak, and as such this monitor is part of the Reactor Coolant Pressure Boundary Leakage Detection System required by Regulatory Guide 1.45. This radiation monitor is equipped with a paper roll-type of movable-filter media. Operations personnel can remotely advance the filter media to remove the filter media that has been in service, and replace it with clean filter media. This operation occurs approximately every two weeks. The flow across the filter media is monitored, and remote indications are received if low flow occurs. This would be indicative of filter clogging, and the cause of the clogging would be investigated. To date, HNP has not experienced filter clogging that has necessitated an increased frequency in the filter media advancement.

C. Training

Prior to performing the Mode 3 walkdown in accordance with procedure EST-227, the walkdown team is given instruction during the pre-job brief regarding actions to be taken if boric acid deposits are detected. The walkdown team is comprised of experienced, VT-2 qualified personnel and system engineers. The examiners are very familiar with the components to be examined and are sensitive to the issues regarding boric acid deposits. The Pressure Test Coordinator is the point of contact for the inspection team. Typically any leaks or evidence of leakage, including boric acid deposits, are reported to the Pressure Test Coordinator and Boron Corrosion Program Engineer for proper disposition in accordance with ASME Code Section XI and PLP-600.

Certified VT-2 personnel using qualified plant procedures perform the visual inspections of any leakage sites identified. Training for this certification describes types of leakage and methods used to identify and locate each type. Boric Acid deposits are specifically addressed. In addition, many HNP Operators are qualified as VT-2 examiners.

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).

HNP personnel were very sensitive to the hot leg nozzle cracking identified at the V.C. Summer Nuclear Plant. Dedicated involvement from Corporate Engineering with the MRP Alloy 600 ITG assured HNP had the most up to date information regarding the V.C. Summer issues. Active involvement by Corporate Engineering continues to keep HNP up to date with the latest information on Alloy 600 issues.

D. <u>Response to Leakage</u>

No boric acid deposits or head degradation have been found due to RPV head penetration CRDM nozzle leakage since the inspections began in Refueling Outage, RFO-05 (1994).

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 RFO9 (4/00-5/00) accessible portions of the reactor vessel head CRDM penetration area were visually inspected in accordance with procedure CM-M0070. A Qualified VT-2 inspector performed the inspection. No boric acid deposits were observed.

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 aid 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.

In accordance with the requirements of the Boron Corrosion Program, any evidence of boric acid leakage (active or inactive) found during inspections, operator/system engineer walkdowns, maintenance, or any other activity requires evaluation. Whenever boric acid residue is found, it must be reported to Engineering or a Work Order initiated, which will be reviewed by the System Engineer. The Boron Corrosion Program Engineer coordinates inspections during outages and reviews inspection results.

Inspection findings are documented, including:

- Identification of the source of the leakage
- Determination of whether the leakage is active or inactive
- If active, reporting of actions taken to stop the leakage

Where boric acid leakage is determined to be significant and has caused corrosion damage of reactor coolant pressure boundary components, a Condition Report is initiated. A formal Engineering Evaluation (Disposition) is performed to evaluate any damage and to justify continued operation, where appropriate. In these cases, procedural guidance is provided for collecting information required to perform the evaluation (including leak rate, system conditions and leak path), and obtaining and preserving evidence. As part of the investigation to disposition the Condition Report, corrective actions to prevent recurrence are identified.

E. <u>Review of Program Effectiveness</u>

As documented in this response, the HNP Boron Corrosion Program provides reasonable assurance that HNP complies with the applicable regulatory requirements discussed in Generic Letter 88-05 and NRC Bulletin 2002-01. The program ensures that leaks in the reactor coolant pressure boundary are identified at levels below Technical Specification limits and that boric acid residue is promptly removed. As documented by the historical record, the program has ensured that no significant wastage has occurred as a result of boric acid corrosion.

The program is periodically reviewed and is updated in response to plant and industry experience. For example, in light of recent industry events, a 100% bare metal inspection (VT-2) of the RPV head and CRDM penetrations will be performed at HNP during the next refueling outage (RFO-11). Future examinations will incorporate industry (i.e., the root cause evaluation of the Davis-Besse event) and site-specific experience.

F. Applicable Regulatory Requirements

The "Applicable Regulatory Requirements" identified within NRC Bulletin 2002-01 and GL 88-05 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 30, "Quality of 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 GDC 14 as described in the HNP FSAR. 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 leak tight 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 GDC 30 as described in the HNP FSAR. This GDC states the following:

"Components which are part of the reactor coolant pressure boundary shall be designed, fabricated, erected and tested to the highest quality standards practical. Means shall be provided for detecting and, to the extent practical, identifying the location of the source of reactor coolant leakage."

A discussion of HNP compliance with GDC 30 is provided within FSAR, Section 3.1.26.

By using conservative design practices and detailed quality control procedures, the pressure-retaining components of the RCPB are designed and fabricated to retain their integrity during normal and postulated accident conditions. Components for the RCPB are designed, fabricated, inspected and tested in conformance with ASME B&PV Code, Section III. All components are classified according to ANSI N18.2-1973 and are accorded the quality measures appropriate to the classification. The design bases and evaluations of RCPB components are discussed in Chapter 5. Because the subject matter of this criterion deals with aspects of the RCPB, further discussion on this subject is treated in the response to Criterion 14, "Reactor Coolant Pressure Boundary."

Means are provided for detecting reactor coolant leakage. The Leak Detection System consists of sensors and instruments to detect and annunciate potentially hazardous leaks before predetermined limits are exceeded. Small leaks are detected by sump level and flow monitoring, airborne particulate radioactivity monitoring and airborne gaseous radioactivity monitoring. In addition to these means of detection, large leaks are detected by changes in flowrates in process lines, and changes in pressurizer level. The allowable leak rates were based on the predicted and experimentally determined behavior of cracks in pipes, the ability to make up coolant system leakage, the normally expected background leakage due to equipment design, and the detection capability of the various sensors and instruments. While the Leak Detection System provides protection from small leaks, the ECCS network provides protection for the complete range of discharges from ruptured pipes. Thus, protection is provided for the full spectrum of possible discharges.

The RCPB and the Leak Detection System are designed to meet the requirements of Criterion 30.

The HNP design criteria meets the GDC 31 as described in the HNP FSAR. 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 (Reference Temperature for Nil-Ductility Transition, 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 GDC 32 as described in the HNP FSAR. 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 leak tight 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.

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.

References

1. 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.