CP&L
A Progress Energy Company

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United States Nuclear Regulatory Commission
ATTENTION: Document Control Desk
Rockville, MD 20852

SHEARON HARRIS NUCLEAR POWER PLANT
DOCKET NO. 50-400/LICENSE NO. NPF-63
30-DAY RESPONSE TO NRC BULLETIN 2002-02, REACTOR PRESSURE VESSEL HEAD AND VESSEL HEAD PENETRATION NOZZLE INSPECTION PROGRAMS

Dear Sir or Madam:

By the letter dated August 9, 2002, the U. S. Nuclear Regulatory Commission (NRC) issued NRC Bulletin 2002-02, "Reactor Pressure Vessel Head And Vessel Head Penetration Nozzle Inspection Programs." The Bulletin directs addressees to submit within 30 days of the date of this bulletin either: (1) a summary discussion of the supplemental inspections to be implemented including Effective Degradation Years (EDY), methods, scope, coverage, frequencies, qualification requirements, and acceptance criteria if we are a Pressurized Water Reactor (PWR) planning to supplement our inspection program with non-visual Non-Destructive Examination (NDE) methods, or (2) if we do not plan to supplement our inspection program with non-visual examination methods, the justification for continued reliance on visual examinations as the primary method to detect degradation (i.e., cracking, leakage, or wastage). Included in the discussion should be statements that address the reliability and effectiveness of the inspections to ensure that all regulatory and technical specification requirements are met during the operating cycle, and that address the six concerns identified in the Discussion Section of this bulletin. Also included in the justification will be a discussion of our basis for concluding that unacceptable vessel head wastage will not occur between inspection cycles that rely on qualified visual inspections. We should include all applicable data to support our understanding of the wastage phenomenon and wastage rates.

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 meet all applicable regulatory and Technical Specification requirements and prevent degradation as observed in the industry. Included in this response are discussions of the reliability and effectiveness of visual examinations as they relate to the six concerns identified in the Discussion Section of this bulletin.

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Harris Nuclear Plant is in the NRC category of plants with very low Effective Degradation Years (EDY), having accumulated an EDY of 2.06. Hence, HNP is considered to have a very low likelihood of cracking of reactor pressure vessel head penetration nozzles. At this time, no significant safety issue has been identified for plants in this category. In addition, HNP performed a bare metal visual examination of a significant portion of the Vessel Head Penetrations (VHP) in the last outage, and did not identify either leakage from or cracking in Vessel Head Penetration (VHP) nozzles.

HNP plans to perform 100% bare metal inspections in Refueling Outage RFO-11 (currently planned for April 2003) per our commitment in the response to NRC Bulletin 2002-01. Future inspection plans will be developed using careful consideration of vendor capabilities and industry standards for such inspections, including activities and recommendations of the Electric Power Research Institute's (EPRI) Materials Reliability Program (MRP). In addition, NDE testing experience by high and medium susceptibility plants will provide useful information that HNP can use to evaluate when NDE testing should be scheduled. The information in Enclosure 1 provides the basis for the HNP plan for inspections in this area.

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

Sincerely,

James Scarola

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.

Darlene S. Yarborough
Notary (Seal)

DARLENE S. YARBOROUGH
NOTARY
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PUBLIC
HARNETT COUNTY, NC

My commission expires: 2-21-2005

c: Mr. J. B. Brady, NRC Sr. Resident Inspector
Ms. Beverly Hall, Section Chief, Radiation Protection Section, NCDENR
Mr. R. Subbaratnam, 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 Effective Degradation Years (EDY) using the time-at-temperature model and plant-specific input data reported in EPRI's Material Reliability Project document, MRP-2001-48. The current EDY for HNP is 2.06. The current rate of EDY accrual at the normal reactor pressure vessel head temperature of 558°F is 0.171 EDY/Effective Full Power Year (EFPY). 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 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 following sections include the HNP responses to the specific items as required by NRC Bulletin 2002-02.

HNP Response to the Six Concerns Identified In The "Discussion" Section of the Bulletin

Concern 1: Circumferential cracking of Control Rod Drive Mechanism (CRDM) nozzles was identified by the presence of relatively small amounts of boric acid deposits. This finding increases the need for more effective visual and non-visual NDE inspection methods to detect the presence of degradation in CRDM nozzles before nozzle integrity is compromised.

Response: Since the initial discovery of circumferential cracks above the J-groove weld in 2001, visual inspection techniques and approaches employed have been dramatically improved and a heightened sense of awareness exists for the range in size and appearance of visual indications that must be further investigated. Non-visual techniques similarly have and continue to evolve to more effectively examine the penetration tube and associated welds for evidence of cracks. Nothing in the recent events at Davis-Besse has altered the fundamental inspection capability requirements previously established as necessary to identify the presence of Primary Water Stress Corrosion Cracking (PWSCC) and subsequent associated wastage. The effectiveness of inspection techniques continues to be evaluated and improved.

EPRI MRP has published detailed guidance for performing visual examinations of RPV heads (Ref. 3). A utility workshop was recently conducted to discuss this guidance and lessons learned from recent field experience (including Davis-Besse). RPV head bare metal visual inspections at Harris Nuclear Plant will be performed and documented in accordance with written procedures and acceptance criteria that comply with the guidance of the MRP Inspection Plan. Evaluations and corrective actions will be rigorous and thoroughly documented.

In order for outside diameter (OD) circumferential cracks above the J-groove weld to initiate and grow, a leak path must first be established to the CRDM annulus region from the inner wetted surface of the reactor vessel head (RVH). If primary water does not leak to the annulus the environment does not exist to cause circumferential OD cracking. Axial cracks in the CRDM nozzles or cracks in J-groove welds must first initiate and grow through wall. Experience has shown that through-wall axial cracks will result in observable leakage at the base of the penetration on the outer surface of the vessel, even with interference fits. Alloy 600 steam generator drain pipes at Harris Nuclear Plant (1988) and pressurizer instrument nozzles at Nogent 1 and Cattenom 2 (1989) were all roll expanded but still developed leaks during operation (Ref. 4). Plant specific top head gap analyses have been performed for a large number of plants, with nozzle initial interference fits ranging from 0 to 0.0034". These analyses have confirmed the presence of a physical leak path in essentially all nozzles under normal operating pressure and temperature conditions (Ref 4).

The probability of detecting small CRDM leaks by visual inspections alone is high. As the NRC staff has stated in reference 5, "Visual inspections of the reactor coolant system pressure boundary have been proven to be an effective method for identifying leakage from PWSCC cracks in Alloy 600 base metal and Alloy 82/182 weld metal. Specifically, visual inspections have detected leaks in reactor pressure vessel (RPV) head CRDM nozzles, RPV head thermocouple nozzles, pressurizer heater sleeves, pressurizer instrument nozzles, hot leg instrument nozzles, steam generator drain lines, a RPV hot leg nozzle weld, a power operated relief valve (PORV) safe end and a pressurizer manway diaphragm plate" (Ref. 5). To date, no leaking CRDM nozzles have been discovered by non-visual NDE examinations except for the three nozzles at Davis-Besse where leakage would have been detected visually had there been good access for visual inspections and the head cleaned of pre-existing boric acid deposits from other sources (Ref. 4).

Finally, as described under Concern 3 below, detailed probabilistic fracture mechanics (PFM) analyses have been performed to demonstrate the effectiveness of visual inspections in protecting the CRDM nozzles against failure due to circumferential cracking (Ref. 6). Even though the above discussion illustrates that visual inspections performed in accordance with MRP recommendations have a high probability of detecting through-wall leakage, a very low probability of detection was assumed in the PFM analyses. The PFM analyses assume only a 60% probability that leakage will be detected if a CRDM nozzle is leaking at the time a visual inspection is performed. Furthermore, if a nozzle has been inspected previously, and leakage was missed, subsequent visual inspections are assumed to have only a 12% probability of detecting the leak. Even with these conservative probabilities of detection assumptions, the PFM analyses show that visual inspection every outage reduces the probability of a nozzle ejection to an acceptable level for plants with 18 or more EDY. Visual inspections of plants with fewer than 18 EDY in accordance with the MRP Inspection Plan will maintain the probability of nozzle ejection for these plants more than an order of magnitude lower than that for plants with EDYs greater than 18. HNP has an EDY of 2.06.

In summary, the industry has responded to the need to detect small amounts of leakage by increased visual inspection sensitivity, increased inspection frequencies, and improved inspection capabilities. Small amounts of leakage can be detected visually and it has been shown that timely detection by visual examination will ensure the structural integrity of the RPV head penetrations with respect to circumferential cracking.

Concern 2: Cracking of 82/182 weld metal has been identified in CRDM nozzle J-groove welds for the first time and can precede cracking of the base metal. This finding raises concerns because examination of weld metal material is more difficult than base metal.

Response: Cracks in the J-groove weld do not pose an increased risk regarding nozzle ejection as compared to penetration base metal cracks. J-groove weld cracks that initiate and grow through-wall will leak the same as cracks in the penetration base metal. Therefore, weld cracks pose a similar risk as cracks in the base material and are equally detectable by visual examination. Although higher crack growth rates have been observed in weld metal, the industry model of time-to-leakage includes plants that have had weld metal cracking as well as base metal cracking. The visual examination frequencies from the MRP Inspection Plan have been conservatively established based on the risk informed analyses considering leakage due to both weld metal and base metal cracking.

Concern 3: Through-wall circumferential cracking from the outside diameter of the CRDM nozzle has been identified for the first time. This raises concerns about the potential for failure of CRDM nozzles and control rod ejection, causing a Loss of Coolant Accident (LOCA).

Response: Probabilistic Fracture Mechanics (PFM) analyses using a Monte-Carlo simulation algorithm were performed to estimate the probability of nozzle failure and control rod ejection due to through wall circumferential cracking (Ref. 6). The PFM analyses conservatively assume that, once a leak path has extended to the annulus region, an OD circumferential crack develops instantaneously, with a length encompassing 30° of the nozzle circumference. Fracture mechanics crack growth calculations are then performed for this initially assumed crack, using material crack growth rate data form EPRI Report MRP-55 (Ref. 7). The parameters used in the PFM model were benchmarked against the most severe cracking found to date in the industry (B&W Plants) and produced results that are in agreement with experience to date. The analyses were used to determine probability of nozzle failure versus EFPY for various head operating temperatures. Analyses were then performed to estimate the effect of visual and non-visual (NDE) inspections of the plants in the most critical inspection category, using the conservative assumption discussed above (see Concern #1 response) for probability of leakage detection by visual inspection. These analyses demonstrate that performing visual inspections significantly reduces the probability of nozzle ejection, and that performing such examinations on a regular basis (in accordance with the inspection schedule prescribed in the MRP Inspection Plan) effectively maintains the probability of nozzle ejection at an acceptably low level indefinitely.

In the extremely unlikely event that nozzle failure and rod ejection were to occur due to an undetected circumferential crack, an acceptable margin of safety to the public would still be maintained (Ref. 8). The consequences of such an event are similar to that of a small-break LOCA, which is a design-basis event. The probability of core damage given a nozzle failure (assuming that failure leads to ejection of the nozzle from the head) has been estimated to be 1×10^{-3}. The PFM analyses demonstrate that periodic visual inspections are capable of maintaining the probability of nozzle failure due to circumferential cracking well below 1×10^{-3}. Therefore, the PFM analyses demonstrate that the resulting incremental change in core damage frequency due to CRDM nozzle cracking can be maintained at less than 1×10^{-6} (i.e., 1×10^{-3} times 1×10^{-3} equals 1×10^{-6}) per plant year, through a program of periodic visual examinations performed in accordance with the MRP inspection plan. This result is consistent with NRC Regulatory Guide 1.174 that defines an acceptable change in core damage frequency (1×10^{-6} per plant year) for changes in plant design parameters, technical specifications, etc.

Concern 4: The environment in the CRDM housing/RPV head annulus will likely be more aggressive after any through-wall leakage because potentially highly concentrated borated primary water may become oxygenated. This raises concerns about the technical basis for current crack growth rate models.

Response: The MRP panel of international experts on Stress Corrosion Cracking (SCC) (including representatives from ANL/NRC Research), prior to the Davis-Besse incident, gave extensive consideration to the likely environment in the annulus between a leaking CRDM nozzle and the RPV head and revisited this issue subsequently (Ref. 7). When revisited, the relevant arguments remain valid for leak rates in the less than 1 liter/hour or 0.004 gallon per minute, which plant experience has shown to be the usual case. The conclusions were

1. An oxygenated crevice environment is highly unlikely because:
   - Back diffusion of oxygen is too low compared to counter flow of escaping steam (two independent assessments based on molecular diffusion models were examined).
   - Oxygen consumption by the metal walls would further reduce its concentration.
   - Presence of hydrogen from leaking water and diffusion through the upper head results in a reducing environment.
   - Even if the concentration of hydrogen was depleted by local boiling, coupling between low alloy steel and Alloy 600 would keep the electrochemical potential low.
   - Corrosion potential will be close to the Ni/NiO equilibrium, resulting in PWSCC susceptibility similar to normal primary water.

2. The most likely crevice environments are either hydrogenated steam or PWR primary water within normal specifications and both would result in similar, i.e. non-accelerated, susceptibility of the Alloy 600 penetration material to PWSCC.

3. If the boiling interface happens to be close to the topside of the J-weld, itself a low probability occurrence, concentration of PWR primary water solutes, lithium hydroxide and boric acid, can in principle occur. Of most concern here would be the accelerating effect of elevated pH on SCC, but calculations and experiments show that any changes are expected to be small, in part because of the buffering effects of precipitates. A factor of 2x on the crack growth rate (CGR) conservatively covers possible acceleration of PWSCC, even up to a high-temperature pH of around 9.

For larger leakage rates, which could lead to local cooling of the head, concentration of boric acid and development of a sizeable wastage cavity adjacent to the penetration, the above arguments no longer directly apply. However, limited data (Berge et al., 1997) on SCC in concentrated boric acid solutions indicate that:

- Alloy 600 is very resistant to transgranular SCC (material design basis).
- High levels of oxygen <u>and</u> chloride are necessary for intergranular cracking to occur at all.
- The effects are then worse at intermediate temperatures, suggesting that the mechanism is different from PWSCC.

The above considerations show that there is no basis for assuming that any post-leakage, crevice environment in the CRDM housing/RPV head annulus would be significantly more aggressive with regard to SCC of the Alloy 600 penetration material than normal PWR primary water, irrespective of the assumed leakage rate and/or annulus geometry. The current industry model (Ref. 7), which includes a factor of 2x on CGR to cover residual uncertainty in the composition of the annulus environment, remains valid.

Concern 5: The presence of boron deposits or residue on the RPV head, due to leakage from mechanical joints, could mask pressure boundary leakage. This raises concerns that a through-wall crack may go undetected for years.

Response: The experience at Davis-Besse has clearly demonstrated that effective visual inspection for leakage from CRDM nozzle and weld PWSCC requires unobstructed inspection access and that the head surface be free of pre-existing boric acid deposits. Accumulations of debris and boric acid deposits from other sources can interfere with a determination as to the presence or absence of boric acid deposits extruding from the tube-to-head annulus. Therefore, to effectively perform a visual examination of the RPV head outer surface for penetration leakage, such deposits and debris accumulations must be carefully inspected, removed, and the area re-inspected. Evaluation may show that it is necessary to perform a non-visual examination to establish the source of the leakage.

Accordingly, each inspection at HNP is conducted with a questioning attitude and any boric acid deposit on the vessel head is evaluated to determine its source in accordance with existing industry guidance, supplemented by the most recent industry experience at the time of the inspection. These requirements are incorporated in the visual inspection guidance contained in the MRP Inspection Plan. Implementation of these requirements precludes the cited condition of a through-wall crack remaining undetected for years.

At HNP, 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, "Reactor Vessel Mirror Insulation Disassembly/Reassembly Procedure." 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-09 (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 all of the cases discussed above the boric acid deposits discovered were removed thereby precluding the possibility that they would mask future leaks.

Concern 6: The causative conditions surrounding the degradation of the RPV head at Davis-Besse have not been definitively determined. The staff is unaware of any data applicable to the geometries of interest that support accurate predictions of corrosion mechanisms and rates.

Response: The causes of the Davis-Besse degradation are sufficiently well known to enable other facilities to avoid similar conditions that would result in significant wastage. The root cause evaluation performed by the utility (Ref. 9) clearly identifies the root cause as PWSCC of CRDM nozzles followed by boric acid corrosion. The large extent of degradation has been attributed to failure of the utility to address evidence that had been accumulating over a five-year period of time (Figure 26 of Ref. 9).

The industry has provided utilities with guidance for vessel top head visual inspections to ensure that conditions such as those discovered at Davis-Besse will not occur. Visual inspection guidelines have been provided (Ref. 3), and a workshop was conducted to thoroughly review industry experience, regulatory requirements, leakage detection, and analytical work performed to understand the causes of high wastage rates (Ref. 10).

Subsequent to significant wastage being discovered on the Davis-Besse RPV head, the industry has performed analysis work to determine how a small leak such as seen at several plants can progress to the significant amounts of wastage discovered at Davis-Besse. This work is referenced in the basis for the MRP Inspection Plan (Ref. 11) and was previously presented to the NRC (Ref. 12).

The analytical work shows that the corrosion rate is a strong function of the leakage rate. Finite element thermal analyses show that leak rates must reach approximately 0.1 gpm for there to be sufficient cooling of the RPV top head surface to support concentrated liquid boric acid that will produce high corrosion rates. The leak rate is in turn a strong function of the crack length. The effect of crack length above the J-groove weld on crack opening displacement and area has been confirmed by finite element modeling of nozzles including the effects of welding residual stresses and axial cracks. Leak rates have been calculated using crack opening displacements and areas determined by the finite element analyses, and leak rate models based on PWSCC cracks in steam generator tubes.

Cracks that just reach the annulus through the base metal or weld metal will result in small leaks such as those that produced small volumes of boric acid deposits on several vessel heads at locations where the CRDM nozzles penetrate the RPV head outside surface. These leaks are typically on the order of 10^{-6} to 10^{-4} gpm. There is no report of any of these leaks resulting in significant corrosion. A leak rate of 10^{-3} gpm will result in a release of about 500 in^3 of boric acid deposits in an 18-month operating cycle, which will be detectable by visual inspections.

The time for a crack to grow from a length that will produce a leak rate of 10^{-3} gpm to a leak rate of 0.1 gpm has been estimated by deterministic analyses based on the MRP crack growth models to be 1.7 years for plants with 602°F head temperatures. Probabilistic analyses show that there is less than a $1x10^{-3}$ probability that corrosion will proceed to the point that the inside surface cladding of the head would be uncovered over a significant area before the wastage would be detected by supplemental visual inspections as required under the MRP Inspection Plant. During the transition from leak rates of 10^{-3} gpm to 0.1 gpm, loss of material will be by relatively slow processes (Ref. 11).

The ability to detect leakage prior to the risk of structural failure is illustrated by Figure 26 of the Davis-Besse root cause analysis report. There was visual evidence of boric acid deposits on the vessel head for five years prior to the degradation being detected. Guidance provided in the MRP Inspection Plan would not permit these conditions to exist without determining the source of the leak, including nondestructive examinations if necessary.

Therefore, while the exact timing of the event progression at Davis-Besse cannot be definitively established, the probable durations can be predicted with sufficient certainty to conclude that a visual inspection regimen can ensure continued structural integrity of the RCS pressure boundary.

Discussion of Regulatory Commitments and Technical Specification Requirements

10 CFR 50, Appendix B

NRC Bulletin 2002-02 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. Previous inspection activity has not identified VHP nozzle leakage. 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. EPRI Letter MRP 2002-086, "Transmittal of PWR Reactor Pressure Vessel (RPV) Upper Head Penetrations Inspection Plan, Revision 1, August 6, 2002," from Leslie Hartz, MRP Senior Representative Committee Chairman, August 15, 2002.
2. EPRI Document MRP-75, "PWR Reactor Pressure Vessel (RPV) Upper Head Penetrations Inspection Plan, Revision 1," August 2002
3. EPRI Technical Report 1006899, "Visual Examination for Leakage of PWR Reactor Head Penetrations on Top of the RPV Head: Revision 1," March 2002
4. Appendix B of EPRI Document MRP-75, "Probability of Detecting Leaks in RPV Top Head Nozzles," August 2002
5. EPRI TR-103696, "PWSCC of Alloy 600 Materials in PWR Primary System Penetrations," July 1994.
6. Appendix A of EPRI Document MRP-75, "Technical Basis for CRDM Top Head Penetration Inspection Plan," August 2002.
7. EPRI Document MRP-55, "Crack Growth Rates for Evaluating Primary Water Stress Corrosion Cracking (PWSCC) of Thick-Wall Alloy 600 Material," July 2002.
8. Walton Jensen, NRC, Reactor Systems Branch, Division of Systems Safety and Analysis (DSSA), Sensitivity Study of PWR Reactor Vessel Breaks, memo to Gary Holahan, NRC, DSSA, May 10, 2002.
9. Davis-Besse Nuclear Power Station report CR2002-0891, Root Cause Analysis Report – Significant Degradation of the Reactor Pressure Vessel Head, April 2002.
10. Proceedings: EPRI Boric Acid Corrosion Workshop, July 25–26, 2002, Baltimore, Maryland, to be published by EPRI.
11. Appendix C of EPRI Document MRP-75, "Supplemental Visual Inspection Intervals to Ensure RPV Closure Head Structural Integrity," August 2002.
12. Glenn White, Chuck Marks and Steve Hunt, Technical Assessment of Davis-Besse Degradation, Presentation to NRC Technical Staff, May 22, 2002.