Progress Energy

10 CFR 50.54(f)

Serial: RNP-RA/03-0001

JAN 31 2003

United States Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555-0001

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2
DOCKET NO. 50-261/LICENSE NO. DPR-23

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION REGARDING 60-DAY RESPONSE TO NRC BULLETIN 2002-01, "REACTOR PRESSURE VESSEL HEAD DEGRADATION AND REACTOR COOLANT PRESSURE BOUNDARY INTEGRITY"

Ladies and Gentlemen:

On November 22, 2002, the NRC issued a Request for Additional Information (RAI) on the 60-day responses to NRC Bulletin 2002-01,"Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity," for H. B. Robinson Steam Electric Plant (HBRSEP), Unit No. 2. That RAI requested that Carolina Power and Light (CP&L) Company submit its response for HBRSEP, Unit No. 2, no later than 60 days from receipt of the letter. The purpose of this letter is to provide the information requested by the RAI.

Attachment I to this letter provides an Affirmation in accordance with 10 CFR 50.54(f).

CP&L provides its response to the RAIs in Attachment II.

If you have any questions regarding this submittal, please contact Mr. C. T. Baucom.

Sincerely,

B. L. Fletcher III
Manager – Support Services – Nuclear

JMG/jmg

A095

Progress Energy Carolinas, Inc.
3581 West Entrance Road
Hartsville, SC 29550

Attachments:

I. Affirmation

II. Response to Request for Additional Information Regarding 60-Day Response to NRC Bulletin 2002-01, "Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity"

c: Mr. L. A. Reyes, NRC, Region II
Mr. C. Patel, NRC, NRR
NRC Resident Inspectors

AFFIRMATION

The information contained in letter RNP-RA/03-0001 is true and correct to the best of my information, knowledge, and belief, and the sources of my information are officers, employees, contractors, and agents of Carolina Power and Light Company. I declare under penalty of perjury that the foregoing is true and correct.

Executed On: 31 JAN. 2003

J. W. Moyer

J. W. Moyer
Vice President, HBRSEP, Unit No. 2

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2

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Introduction:

The NRC staff's review of the licensees' responses to NRC Bulletin 2002-01 resulted in the following Request for Additional Information (RAI). The RAIs and Carolina Power and Light (CP&L) Company's responses for H. B. Robinson Steam Electric Plant (HBRSEP), Unit No. 2, are provided below.

NRC Question 1

Provide detailed information on, and the technical basis for, the inspection techniques, scope, extent of coverage, and frequency of inspections, personnel qualifications, and degree of insulation removal for examination of Alloy 600 pressure boundary material and dissimilar metal Alloy 82/182 welds and connections in the reactor coolant pressure boundary (RCPB). Include specific discussion of inspection of locations where reactor coolant leaks have the potential to come in contact with and degrade the subject material (e.g., reactor pressure vessel bottom head).

Response

Table A identifies the Alloy 600 pressure boundary components and Alloy 82/182 welds at HBRSEP, Unit No. 2. Also included in the table are the inspection technique, personnel qualifications, extent of coverage, inspection frequency, type of insulation, and the degree of insulation removal performed to facilitate the inspection.

The requirements of the Boric Acid Corrosion Control Program (BACCP) are described in Plant Program Procedure (PLP)-040, "Program For Prevention of Boric Acid Corrosion of RCS [Reactor Coolant System] Carbon Steel Bolting (Generic Letter [GL] 88-05)." PLP-040 states that if evidence of leakage is found, regardless of the source, sufficient insulation removal is required to positively identify the source of the leakage. Additional inspections are required to identify any potential drip or spray pathways and resulting targets of leakage, dripping, or spray. Leaks must be repaired or evaluated and documented to ensure pressure boundary integrity. Boric acid corrosion of the targets is evaluated in accordance with IWA-5000, "System Pressure Tests," and IWB-3000, "Acceptance Standards," of the American Society of Mechanical Engineers (ASME) Boiler & Pressure Vessel (B&PV) Code, Section XI, with regard to structural integrity. Repairs or replacements in accordance with IWA-4000, "Repair/Replacement Activities," are performed as necessary. Boric acid corrosion of other safety-related components are evaluated in accordance with the applicable codes and standards, and repaired or replaced as necessary. Plant procedures provide for a daily RCS evaluation for leakages and satisfy Technical Specification (TS) Surveillance Requirements for monitoring the RCS leakrate. These surveillance procedures have proven effective in detecting leakrates significantly smaller than TS limits.

Technical guidance for the examination of Control Rod Drive Mechanism (CRDM) nozzles and J-groove welds performed during Refueling Outage – 21 (RO-21) was provided in Electric Power Research Institute (EPRI) document MRP-75, "PWR Reactor Pressure Vessel (RPV) Upper Head Penetrations Inspection Plan," supplemented by References 7 and 8. The technical basis for the remainder of the examinations is defined by the requirements of the ASME Code, Section XI, for visual, surface, and volumetric examinations.

Although HBRSEP, Unit No. 2, is ranked in the EPRI category of plants with a high susceptibility of cracking of RPV head penetration nozzles, recent inspections during RO-21 identified no leakage or cracking (Reference 2). Future inspections and their frequency will be considered based on these results and industry work on this issue.

NRC Question 2

Provide the technical basis for determining whether or not insulation is removed to examine locations where conditions exist that could cause high concentrations of boric acid on pressure boundary surfaces or locations that are susceptible to primary water stress corrosion cracking (Alloy 600 base metal and dissimilar metal Alloy 82/182 welds). Identify the type of insulation for each component examined, as well as any limitations to removal of insulation. Also include in your response actions involving removal of insulation required by your procedures to identify the source of leakage when relevant conditions (e.g., rust stains, boric acid stains, or boric acid deposits) are found.

Response

The HBRSEP, Unit No. 2, BACCP was developed in accordance with GL 88-05 and the ASME Code, Section XI, and does not explicitly require the removal of insulation for inspection purposes. This is based on the ability to detect leakage by insulation becoming wet or saturated with borated water at a low point in the system. When inspections reveal evidence of leakage, insulation is required to be removed, as necessary, to make an accurate determination of the source of the leakage and the condition of components in the leakage path. Table A identifies the types of insulation for the applicable inspection points. Past experience indicates that no significant limitations exist to the removal of insulation in support of inspections associated with Table A components. The reactor vessel head insulation has been replaced with blanket contoured insulation that provides for easier removal for RPV head inspection purposes.

NRC Question 3

Describe the technical basis for the extent and frequency of walk downs and the method for evaluating the potential for leakage in inaccessible areas. In addition, describe the degree of inaccessibility, and identify any leakage detection systems that are being used to detect potential leakage from components in inaccessible areas.

Response

The walk downs performed to detect evidence of leakage from borated systems are performed at the start and end of each refueling outage in accordance with Operations Surveillance Test (OST)-052, "RCS Leakage Test and Examination Prior to Startup Following an Opening of the Primary System (Refueling and/or Startup Interval)," and OST-053, "Inspection for Reactor Coolant System Leakage (Prior to and Following Cooldown) (Refueling Interval)." OST-052 is required by the ASME Code, Section XI. OST-053 is not required by the ASME Code, Section XI, but is performed in accordance with good engineering practice. These outage inspections are performed by VT-2 examiners who are briefed on the requirements of GL 88-05 and PLP-040 prior to the inspections. The walk downs cover the RCS pressure boundary, with specific components identified for inspection in the applicable plant procedures, as well as borated systems beyond the scope of Section XI requirements.

During each refueling outage, certified VT-2 examiners also perform visual examinations of connections on the RCS in accordance with Engineering Surveillance Test (EST)-083, "Inservice Inspection [ISI] Pressure Testing of Reactor Coolant System (Refueling Shutdown Interval)," as required by the ASME Code, Section XI. EST-083 includes both a VT-2 visual examination of pressure-retaining bolted connections and a VT-2 visual examination of the RPV head during inservice leakage testing with the RCS at a nominal operating pressure of 2235 psig. The acceptance criteria for EST-083 includes the requirement that "no through wall leakage exists on any piping system examined during the performance of this procedure." Personnel performing VT-2 visual examinations in accordance with EST-083 are certified and qualified to Level II or higher in accordance with non-destructive examination (NDE) procedures. Class 1, 2, and 3 systems are examined to the boundaries defined by Section XI, as modified by the following HBRSEP, Unit No. 2, Relief Requests for the Fourth Ten-Year ISI Interval:

RR-9 (Removal of bolting insulation following depressurization)
RR-10 (VT-1 examination and bolting evaluation in lieu of VT-3)
RR-11 (Class 3 system leakage test in lieu of hydrostatic test)
RR-12 (End of interval system leakage with normal RCS alignment

VT-2 examination techniques utilized for ASME Section XI pressure tests are conducted in accordance with IWA-5240, "Visual Examination," and are conducted by VT-2 Level II certified personnel with training to look for boric acid. The pressure test procedures have a step for a review of the boric acid program requirements contained in PLP-040.

During the course of the system leakage test, insulation is not required to be removed unless required to locate the source of leakage in accordance with IWA-5250, "Corrective Action," and IWB-3522, "Standards for Examination Category B-P, All Pressure Retaining Components." The ASME Code, Section XI, allows for examination of components at the component's lowest elevation or on the floor or equipment surfaces located beneath the component. For instances where portions of borated piping systems are inaccessible for direct VT-2 examination due to wall and floor penetrations, a VT-2 examination in accordance with the ASME Code, Section XI, is conducted on both sides of the wall penetration, and above and below the floor penetration. In the event that evidence of leakage is detected at the penetration, investigation would be performed in accordance with ASME Code requirements, and investigation of the leak/spray pattern would be performed in accordance with PLP-040.

During power operation, an RCS leakage surveillance is performed every 72 hours in accordance with OST-051, "Reactor Coolant System Leakage Evaluation (Every 72 hours during steady state operation and within 12 hours after reaching steady state operation)." Leak rates significantly lower than the TS limit can be identified and trended. Indication of RCS leakage is continuously available to the plant operators by use of the following monitors:

- Reactor coolant drain tank level (remote gage board)
- Pressurizer relief tank level
- Accumulator pressure and level indications
- Containment air particulate and noble gas monitors
- Containment sump level monitors
- The component cooling water radiation monitor (R-17)
- Increasing charging pump flow rate compared with reactor coolant system inventory changes
- Unscheduled increases in reactor makeup water usage

If significant leakage or an increasing trend is identified, both OST-051 and PLP-040 require that action be initiated to investigate the cause of the leakage, to identify the source, and take corrective action.

<u>NRC Question 4</u>

Describe the evaluations that would be conducted upon discovery of leakage from mechanical joints (e.g., bolted connections) to demonstrate that continued operation with the observed leakage is acceptable. Also describe the acceptance criteria that were established to make such a determination. Provide the technical basis used to establish the acceptance criteria. In addition,

a. if observed leakage is determined to be acceptable for continued operation, describe what inspection/monitoring actions are taken to trend/evaluate changes in leakage, or

b. if observed leakage is not determined to be acceptable, describe what corrective actions are taken to address the leakage.

Response

The goal of the BACCP is the identification and correction of RCS leaks. The process for identifying and dispositioning boric acid leaks is summarized as follows:

Leak Identification

If evidence of RCS leakage is discovered, PLP-040 requires the leak/spray path to be investigated, removing insulation as necessary, to determine the extent of any component degradation. Any borated system leakage in the vicinity of primary pressure boundary targets that could leak or spray on these targets is not acceptable. OST-053 and EST-083 require inspections to be documented. These procedures also require corrective action or evaluation of any identified leakage. Nuclear Condition Reports are generated in accordance with Nuclear Generation Group Common (NGGC) procedure, CAP-NGGC-0200, "Corrective Action Program [CAP]," for significant boric acid leaks.

Evaluation of Leaks

Boric acid corrosion of primary pressure boundary targets is addressed in accordance with PLP-040 and requires the performance of an engineering evaluation in accordance with the provisions of IWA-5000 and IWB-3000. Boric acid corrosion of other safety-related components is evaluated in accordance with applicable codes and standards, and repaired or replaced, as necessary.

Consistent with the guidance contained in EPRI Report Test Reference (TR)-1000975, "Boric Acid Corrosion Guidebook," the evaluations must consider:

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

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

These actions are intended to ensure that ASME Code and regulatory requirements are met, and that the integrity of the RCS pressure boundary is maintained and protected.

NRC Question 5

Explain the capabilities of your program to detect the low levels of reactor coolant pressure boundary leakage that may result from through-wall cracking in the bottom reactor pressure vessel head incore instrumentation nozzles. Low levels of leakage may call into question reliance on visual detection techniques or installed leakage detection instrumentation, but have the potential for causing boric acid corrosion. The NRC has had a concern with the bottom reactor pressure vessel head incore instrumentation nozzles because of the high consequences associated with loss of integrity of the bottom head nozzles. Describe how your program would evaluate evidence of possible leakage in this instance. In addition, explain how your program addresses leakage that may impact components that are in the leak path.

Response

The primary means for identifying RCS leakage trends is OST-051. This procedure is performed every 72 hours, at a minimum, during steady-state plant operation, in accordance with TS Surveillance Requirement (SR) 3.4.13.1. The test results are maintained and trended by plant personnel and reviewed by plant management.

Another method used to monitor RCS leakage is the trending of Radiation Monitor R-11, CV and Plant Vent Air Particulate Monitor. R-11 trends are monitored in accordance with the Chemistry shift log, and the results are trended in accordance with Chemistry Procedure (CP)-116, "Chemistry Data Tracking And Trending Program." R-11 and RCS leakage trends are correlated so that the data may be compared to provide further confirmation of leakage. R-11 trends are shared with plant organizations, along with the measured leakage results, such that plant personnel are aware of current trends. These trending and monitoring practices have been successful during prior operating cycles in identifying and correcting the sources of low levels of RCS leakage.

In accordance with OST-051, if RCS unidentified leakage exceeds 0.20 gpm or RCS identified leakage is greater than 0.30 gpm, an investigation is commenced, plant staff is informed, and daily leak rate calculations are performed. The investigation required by OST-051 due to increasing parameter trends typically includes a walk down of accessible areas of the reactor coolant pressure boundary by Operations staff. The walk down is sometimes accompanied by Engineering personnel with the purpose of identifying external leakage and internal leakage across system boundaries, such as in-leakage into the pressurizer relief tank.

Areas that are inaccessible for visual inspection during the plant operating cycle, such as the bottom reactor pressure vessel head incore instrumentation nozzles, are visually inspected as part of the ISI program at the beginning of a refueling outage or during plant startup. When performing visual inspections, if either the suspect component or adjacent components and those that are in the leak path are wetted or suspected of being wetted, they are evaluated in accordance with PLP-040. Further details of this element of the program are provided in the response to NRC Question 2 and in References 7 and 8.

NRC Question 6

Explain the capabilities of your program to detect the low levels of reactor coolant pressure boundary leakage that may result from through-wall cracking in certain components and configurations for other small diameter nozzles. Low levels of leakage may call into question reliance on visual detection techniques or installed leakage detection instrumentation, but have the potential for causing boric acid corrosion. Describe how your program would evaluate evidence of possible leakage in this instance. In addition, explain how your program addresses leakage that may impact components that are in the leak path.

Response

Leakage from these components is treated the same as that addressed in Response 5 above. Potential RCS pressure boundary leakage is treated with equally high levels of importance. Plant procedures require that evaluations of leakage consider the effects of components in the leak path.

During RO-21, post-shutdown pressure boundary component inspections in accordance with EST-083 and OST-053 identified no through-wall leaks, and led to the repair of a number of small boric acid leaks consisting of packing and body/bonnet connections. As indicated in the response to NRC Question 5, these small leaks contributed to increasing trends of RCS unidentified leakage and increasing trends in R-11 values. Suspect areas were identified, evaluated, and repaired in accordance with the previously described procedures. These actions reduced the average unidentified RCS leakage to a post-outage value of approximately 0.03 gpm.

NRC Question 7

Explain how any aspects of your program (e.g., insulation removal, inaccessible areas, low levels of leakage, evaluation of relevant conditions) make use of susceptibility models or consequence models.

Response

HBRSEP, Unit No. 2, used the susceptibility model contained within the Electric Power Research Institute (EPRI) Report TP-1006284, "PWR Materials Reliability Program Response to NRC Bulletin 2001-01 (MRP-48)," dated August 2001 for guidance in scheduling the inspection of the RPV head. However, the Westinghouse Owners Group (WOG) Materials Committee has not created susceptibility models or performed consequence reviews for other Alloy 600 components. PLP-040 and the surveillances for leakage which make up the BACCP do not use susceptibility modeling for the targeting of inspections, the scheduling of inspections, or the determination of coverage. Any leakage found is assessed using visual examination, followed by NDE, if warranted. Evaluations of leakage-induced degradation are performed in accordance with the provisions of the ASME Code with regard to structural integrity. If corrosion rates are used, conservative values, such as those referenced in the EPRI Report TR-1000975, are utilized.

NRC Question 8

Provide a summary of recommendations made by your reactor vendor on visual inspections of nozzles with Alloy 600/82/182 material, actions you have taken or plan to take regarding vendor recommendations, and the basis for any recommendations that are not followed.

Response

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

Question 9

Provide the basis for concluding that the inspections and evaluations described in your responses to the above questions comply with your plant Technical Specifications and Title 10 of the Code of Federal Regulations, Section 50.55(a), which incorporates Section XI of the American Society of Mechanical Engineers (ASME) Code by reference. Specifically, address how your boric acid corrosion control program complies with ASME Section XI, paragraph IWA-5250(b) on corrective actions. Include a description of the procedures used to implement the corrective actions.

Response

HBRSEP, Unit No. 2, has concluded that the inspections and evaluations described above comply with applicable regulatory, ASME Code, and TS requirements. The following discussion provides a description of how HBRSEP, Unit No. 2, satisfies these regulations and requirements.

Compliance with 10 CFR 50.55a, "Codes and Standards"

10 CFR 50.55a, "Codes and Standards," requires that ISI and testing be performed in accordance with the requirements of the ASME 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 reactor coolant pressure boundary.

ASME Code, Section XI, paragraph IWA-5250(b), states "If boric acid residues are detected on components, the leakage source and the areas of general corrosion shall be located. Components with local areas of general corrosion that reduce the wall thickness by more than 10% shall be evaluated to determine whether the component may be acceptable for continued service, or whether repair/replacement activities will be performed."

The HBRSEP, Unit No. 2, Third Ten-Year ISI Interval, which commenced on February 19, 1992, was implemented in accordance with the ASME Code, 1986 Edition with no Addenda. The Fourth Ten-Year ISI Interval for HBRSEP, Unit No. 2, commenced on February 19, 2002, and has been developed in accordance with the ASME Code, 1995 Edition with 1996 Addenda.

ASME Code requirements for the code class pressure boundary are implemented through the performance of surveillance tests EST-083 and OST-052. Where leakage is found on adjacent carbon steel material, evaluation and corrective action is conducted in accordance with IWA-5250 by identifying the source of the leakage and areas of general corrosion. Leakage detected at bolted connections is evaluated in accordance with Fourth Ten-Year ISI Interval Relief Request RR-10. These procedures prescribe corrective measures that satisfy Code requirements for the resolution of reactor coolant pressure boundary leakage and boric acid deposition.

The Acceptance Standards provided within both the 1986 and 1995 Editions of the ASME Code for the referenced VT-2 visual examinations are identified as IWB-3522, which requires correction of pressure boundary leakage prior to continued service. Additionally, both the 1986 and 1995 Editions of the ASME Code contain within IWB-3140, "Inservice Visual Examinations," Acceptance Standards for visual examinations required by IWB-2500, "Examination and Pressure Test Requirements." Specifically, IWB-3142, "Acceptance," prescribes Acceptance Standards regarding the acceptability for continued service of components whose visual examination detects relevant conditions.

Guidance for implementing corrective actions for detecting boric acid leakage is described in PLP-040 and by reference to Section XI of the ASME code. Additionally, responsibility for initiating corrective measures for leakage detected during Section XI pressure testing is described in Technical Management Procedure (TMM)-020, "Inservice Pressure Testing Program," and by reference to ASME Section XI. Therefore, the process for managing boric acid corrective measures at HBRSEP, Unit No. 2, meets the requirements of the ASME Code, Section XI, IWA-5250(b) and IWA-5250(a)(2), as modified by Relief Request RR-10.

Compliance with Technical Specifications

10 CFR 50.36, "Technical Specifications," provides requirements for 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 HBRSEP, Unit No. 2, Operating License 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. The current HBRSEP, Unit No. 2, 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.

HBRSEP, Unit No. 2, TS 3.4.13, "Reactor Coolant System Operational Leakage," provides criteria and limits regarding primary system leakage, including LCO 3.4.6.a, which prohibits RCS pressure boundary leakage. Verification that RCS operational leakage is within limits is by performance of an RCS water inventory balance. This is performed on a 72-hour frequency in accordance with SR 3.4.13.1. Should pressure boundary leakage exist, Condition "B" would be entered, which requires the unit to be in MODE 3 within 6 hours and in MODE 5 within 36 hours.

Table A: HBRSEP, Unit No. 2, Alloy 600 and Alloy 82/182 Welds

Component	Qty.	Inspection Technique	Personnel Qual.	Extent of Coverage	Frequency	Type/Degree of Insulation Removal	Technical Basis	Note
Bottom Head Instrument Nozzles	50	Visual (VT-2)	VT-2 Level II	100%	Each refueling outage	Reflective/as required by IWA-5250 & IWB-3522	ASME Section XI (IWB-2500, Examination Category B-P)	
Top Head Vent Line	1	Visual (VT-2)	VT-2 Level II	100%	Each refueling outage	N/A (Not insulated)	ASME Section XI (IWB-2500, Examination Category B-P)	
Control Rod Drive Housings	69	Visual (VT-2)	VT-2 Level II	100%	Each refueling outage	N/A (Not insulated)	ASME Section XI (IWB-2500, Examination Category B-P)	
Control Rod Drive Housing Welds	69	Surface (PT)	PT Level II	10% of peripheral housings	Each Ten -Year ISI Interval	N/A (Not insulated)	ASME Section XI (IWB-2500, Examination Category B-O)	
Leakage Monitor Tubes	2	Visual (VT-2)	VT-2 Level II	100% of Class 1 sections	Each refueling outage	N/A (Not insulated)	ASME Section XI (IWB-2500, Examination Category B-P)	Limited Exposure
Inlet Nozzle to Safe-End Welds	3	Visual (VT-2)	VT-2 Level II	100%	Each refueling outage	Fiberglass/As required by IWA-5250 & IWB-3522	ASME Section XI (IWB-2500, Examination Category B-P)	
Inlet Nozzle to Safe-End Welds	3	Volumetric (UT)	UT Level II	100% Code required volume	Each Ten -Year ISI Interval	Fiberglass/As required by IWA-5250 & IWB-3522	ASME Section XI (IWB-2500, Examination Category B-F)	Surface exam exempted per 3rd ISI Interval Relief Request RR-32
Outlet Nozzle to Safe-End Welds	3	Visual (VT-2)	VT-2 Level II	100%	Each refueling outage	Fiberglass/As required by IWA-5250 & IWB-3522	ASME Section XI (IWB-2500, Examination Category B-P)	
Outlet Nozzle to Safe-End Welds	3	Volumetric (UT)	UT Level II	100% Code required volume	Each Ten -Year ISI Interval	Fiberglass/As required by IWA-5250 & IWB-3522	ASME Section XI (IWB-2500, Examination Category B-F)	Surface exam exempted per 3rd ISI Interval Relief Request RR-32

References:

1. NRC Letter to Mr. J.W. Moyer, Subject: "H. B. Robinson Steam Electric Plant, Unit No. 2 - Response to NRC Bulletin 2002-02, 'Reactor Vessel Head and Vessel Head Penetration Nozzle Inspection Programs' (TAC NO. MB5916)," dated December 30, 2002.
2. HBRSEP, Unit No. 2, Submittal of 30-Day Response to NRC Bulletin 2002-01, "Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity," dated December 13, 2002.
3. Westinghouse Owners Group Letter WOG-02-223, Subject: "Westinghouse and Combustion Engineering Owners Group, Transmittal of Response to NRC Request for Information, Bulletin 2002-01: Vendor Recommendations for Visual Inspections of Alloy 600/82/182 Component Locations (MUHP-5035, CEOG 2046)," dated December 13, 2002.
4. NRC Letter to Mr. J.W. Moyer, Subject: "Bulletin 2002-01, 'Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity,' 60-day response for H. B. Robinson Steam Electric Plant, Unit No. 2 (HBRSEP2) - Request for Additional Information (TAC No. MB4570)," dated November 22, 2002.
5. HBRSEP, Unit No. 2, Submittal of Information Requested by NRC Bulletin 2002-02, "Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity," dated September 9, 2002.
6. HBRSEP, Unit No. 2, "Reactor Vessel Head Inspection Plan for Refueling Outage-21," dated August 12, 2002.
7. HBRSEP, Unit No. 2, Submittal of 60-Day Response to NRC Bulletin 2002-01, "Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity," dated May 17, 2002.
8. HBRSEP, Unit No. 2, Submittal of Information Requested by NRC Bulletin 2002-01, "Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity," dated April 1, 2002.
9. CP&L Response to NRC Generic Letter 88-05, dated May 27, 1988.
10. NRC Letter to Mr. E. E. Utley, Subject: "H. B. Robinson Steam Electric Plant, Unit No. 2, and Shearon Harris Nuclear Power Plant, Unit 1 - Response to Generic Letter 88-05 (TAC Nos. 68493, 68922)."