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ACCESSION NBR:9708040125 DOC.DATE: 97/07/29 NOTARIZED: NO DOCKET #
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CP&L

Carolina Power & Light Company
Robinson Nuclear Plant
3581 West Entrance Road
Hartsville SC 29550

Robinson File No: 13510
Serial: RNP-RA/97-0167

JUL 29 1997

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

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2
DOCKET NO. 50-261/LICENSE NO. DPR-23
SUBMITTAL OF INFORMATION REQUESTED BY GENERIC LETTER 97-01,
"DEGRADATION OF CONTROL ROD DRIVE MECHANISM NOZZLE
AND OTHER VESSEL CLOSURE HEAD PENETRATIONS"

Gentlemen:

NRC Generic Letter (GL) 97-01, "Degradation of Control Rod Drive Mechanism Nozzle and Other Vessel Closure Head Penetrations," dated April 1, 1997, requested certain information be submitted in a response 120 days from the date of the GL. By letter dated May 1, 1997, Carolina Power & Light Company submitted the required response to the GL that committed to provide a submittal that addresses the requested information for the H. B. Robinson Steam Electric Plant, Unit No. 2. The attachment to this letter provides that submittal.

If you have any questions concerning this matter, you may contact me or Mr. H. K. Chernoff of my staff at (803) 857-1437.

Very truly yours,

Terry M. Wilkerson

T. M. Wilkerson
Manager - Regulatory Affairs

JSK/jk
Attachment
c: Mr. B. B. Desai, USNRC Senior Resident Inspector, HBRSEP
Ms. B. L. Mozafari, USNRC Project Manager, HBRSEP
Mr. L. A. Reyes, Regional Administrator, USNRC, Region II

A075

H. B. Robinson Steam Electric Plant, Unit No. 2
Information Requested by Generic Letter 97-01, "Degradation of
Control Rod Drive Nozzle and Other Vessel Closure Head Penetrations"

INTRODUCTION

Generic Letter 97-01, "Degradation of Control Rod Drive Mechanism Nozzle and Other Vessel Closure Head Penetrations," was issued to request licensees to describe their program for insuring the timely inspection of Pressurized Water Reactor (PWR) control rod drive mechanism (CRDM) and other closure head penetrations. This response provides H. B. Robinson Steam Electric Plant (HBRSEP), Unit No. 2 information relative to the information requested by the Generic Letter.

Prior to issuance of the Generic Letter, Carolina Power & Light (CP&L) Company worked with the Westinghouse Owners Group (WOG), the Electric Power Research Institute (EPRI), and the Nuclear Energy Institute (NEI) to understand the operational experience, identify technical issues, causal factors, relative importance, and solutions regarding degradation of control rod mechanism nozzle and other vessel closure penetrations. One of these tasks was the development of safety evaluations that characterized the initiation of cracks, their propagation, and consequences. These safety evaluations are contained in WCAP-13565, Revision 1, "Alloy 600 Reactor Vessel Adapter Tube Cracking Safety Evaluation," and are applicable to HBRSEP. The NRC reviewed the WCAP and issued a safety evaluation to the Nuclear Utilities Management and Resources Council dated November 19, 1993. The WCAP and the NRC safety evaluation establish the basis for HBRSEP continued operation.

REQUESTED INFORMATION ITEM 1.1

"1.1 *A description of all inspections of CRDM nozzle and other VHPs performed to the date of this generic letter, including the results of these inspections."*

"Footnote: Those licensees that have previously submitted the requested information need not resubmit it, but may instead reference the appropriate correspondence in their response to this Generic Letter."

Response

No inservice volumetric inspections have been performed on the HBRSEP reactor vessel closure head penetrations.

Prior to issuance of Generic Letter 88-05, "Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components in PWR Plants," visual inspections for leakage of the HBRSEP reactor vessel head area had been performed as part of the primary pressure boundary walkdowns following the refueling outages. The inspections included the control rod drive housing area and the reactor vessel stud area.

In response to Generic Letter 88-05, a program was established to identify, evaluate, repair, and prevent boric acid corrosion of carbon steel components forming the Reactor Coolant System (RCS) primary pressure boundary. The Generic Letter 88-05 program was based on walkdown inspections performed during refueling outages, walkdown inspections performed during normal opening of the RCS for maintenance activities, trending of daily RCS leakage evaluations, and monitoring for leakage during power operations. This program is implemented by Plant Program Procedure (PLP) - 040, "Program For Prevention Of Boric Acid Corrosion Of RCS Carbon Steel Bolting (Generic Letter 88-05)."

The Generic Letter 88-05 program has augmented the existing surveillance tests for: (1) inspection of the RCS prior to cooldown for refueling; (2) inspection of the vessel head after removal; and (3) inspections associated with the reactor vessel pressure testing conducted prior to startup, in conjunction with the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section XI requirements. The inspections of the vessel head include the control rod drive housing area and the vessel stud area. The procedures specifically require inspection of the canopy seal welds, penetration tube surfaces, penetration tube/head insulation interface - particularly the outer three rows, around the inside of the CRDM cooling duct shroud, the conoseal bolting, and the closure studs and nuts. The inspector is instructed to view as much of the component as possible from the accessible areas.

Although, none of these inspections specifically examine the interface area between the reactor vessel head and the CRDM penetrations, they do require that inspectors perform these activities within the reactor vessel head penetration area. Therefore, if a reactor vessel head penetration experienced a leak, it is likely that the resulting boric acid build-up/corrosion would have been identified by the inspector performing one of these inspections.

The augmented inspections which implemented the commitments made in accordance with Generic Letter 88-05 have been performed during the past five refueling outages, RO13 through RO17. None of these past inspections have identified evidence of leakage or boric acid corrosion originating from the reactor vessel closure head penetration areas.

REQUESTED INFORMATION ITEMS 1.2 THROUGH 1.4

"1.2 If a plan has been developed to periodically inspect the CRDM nozzle and other VHPs:

a) Provide the schedule for first, and subsequent, inspections of the CRDM nozzle and other VHPs, including the technical basis for this schedule.

b) Provide the scope for the CRDM nozzle and other VHP inspections, including the total number of penetrations (and how many will inspected), which penetrations have thermal sleeves, which are spares, and which are instrument or other penetrations."

1.3 If a plan has not been developed to periodically inspect the CRDM nozzle and other VHPs, provide the analysis that supports why no augmented inspection is necessary.

1.4 In light of the degradation of CRDM nozzle and other VHPs described above, provide the analysis that supports the selected course of action as listed in either 1.2 or 1.3, above. In particular, provide a description of all relevant data and/or tests used to develop crack initiation and crack growth models, the methods and data used to validate these models, the plant-specific inputs to these models, and how these models substantiate the susceptibility evaluation. Also, if an integrated industry inspection program is being relied on, provide a detailed description of this program."

Response

CP&L currently has no plans or schedule for conducting volumetric inspections of reactor vessel head penetrations. However, in a effort to establish when volumetric inspections might have been warranted, CP&L previously evaluated the susceptibility of reactor vessel head penetrations to Primary Water Stress Corrosion Cracking (PWSCC) for HBRSEP. Based on the evaluation results, volumetric examinations were not considered to be warranted prior to Refueling Outage 19 (RO19), planned for the fall of 1999.

The plant-specific evaluation, performed in 1994, was based on crack initiation and crack growth prediction methods, including the Peter Scott model, described in WCAP - 13565, Revision 1, "Alloy 600 Reactor Vessel Head Adapter Tube Cracking Safety Evaluation", and WCAP-14024, "Inspection Plan Guidelines for Industry/Plant Inspection of Reactor Closure Head Penetration Tubes."

In addition to the CP&L plant-specific evaluation, each of the three PWR owners groups, EPRI, and NEI are cooperatively working to compile information on the estimated operating time needed to initiate and propagate a 75% through-wall crack in a vessel head penetration. This information will be evaluated to determine if an

adequate number of plants have, or are planning, to inspect as a part of an integrated industry inspection program. CP&L is a member of EPRI, NEI, and the WOG. This work is scheduled to be completed by the end of 1997 and provided to the NRC through NEI.

Additionally, Dominion Engineering, Inc., is currently working under contract with EPRI to develop a computer software module for use by utilities in assessing their plant-specific PWSCC risk probabilities. This software product is being based on the PWSCC risk model which has been previously applied to several domestic plants by the contractor. CP&L is participating as a member of the Utility Advisory Group for this EPRI product. Once the beta testing version of the software is available in late 1997, CP&L plans to use the product for further evaluation of PWSCC risk at HBRSEP. The final version of the software is anticipated to be available for use in 1998.

CP&L will continue with ongoing visual examinations of the reactor vessel head penetrations in accordance with previous commitments made in response to Generic Letter 88-05, in addition to other inspection activities noted in the response to Item 1.0, above. The need and timing for future volumetric inspections at the CP&L nuclear plants will continue to be evaluated and updated, based on the results of the plant-specific evaluation and the current industry programs described above. Once plans for conducting volumetric examinations at the CP&L plants are established, the NRC will be informed regarding the schedule and scope for these activities.

REQUESTED INFORMATION ITEM 2

2. *" Provide a description of any resin bead intrusions, as described in IN 96-11, that have exceeded the current EPRI PWR Primary Water Chemistry Guidelines recommendations for primary water sulfate levels, including the following information:*

   2.1. Were the intrusions cation, anion, or mixed bed?
   2.2. What were the durations of these intrusions?
   2.3. Does the plant's RCS water chemistry Technical Specifications follow the EPRI guidelines?
   2.4. Identify any RCS chemistry excursions that exceed the plant administrative limits for the following species: sulfates, chlorides or fluorides, oxygen, boron, and lithium.
   2.5. Identify any conductivity excursions which may be indicative of resin intrusions. Provide a technical assessment of each excursion and any follow-up actions.
   2.6. Provide an assessment of the potential for any of these intrusions to result in a significant increase in the probability for IGA of VHPs and any associated plan for inspections"

Response

CP&L has reviewed plant historical records to determine if incidents of resin ingress, similar to those which occurred in 1980 and 1981 at the Jose Cabrera (Zorita) plant, occurred at the HBRSEP. This data search was structured to identify resin intrusion events into the primary coolant system with a magnitude greater than 1 ft^3 (30 liters). The threshold of 1 ft^3 was chosen as a conservative lower bound since it represents less than 15% of the estimated volume of resin released into the RCS during the two events at the Jose Cabrera plant.

For the period of plant operation prior to the routine analysis for sulfate in the RCS, the data search was based on a review of the plant's reactor coolant chemistry records relative to specific conductance (conductivity) of the reactor coolant. An elevation of > 28 µS/cm in conductivity was the value used as an indicator of cation resin ingress equivalent to a volume of 1 ft^3. Such a resin intrusion would also have produced a pH depression of > 1.0 pH unit, a lithium increase of > 2 ppm, a suspended solids increase of > 1.0 ppm and a sulfate spike of 15 - 17 ppm.

Chemistry data at power from October 1, 1970 to July 9, 1997, were evaluated to look for conductivity increases above the conductivity value calculated from measured boron and lithium and also to look for pH decreases below the pH value calculated from measured boron and lithium values. Only one conductivity increase greater than 28 µS/cm above the calculated value was found at power. An increase of 35.1 µS/cm above the calculated value of 7.9 µS/cm was observed on March 8, 1971. The pH was elevated by + 0.74 pH units, indicative of the presence of ammonia and hydrazine used to eliminate oxygen. While conductivity increases above calculated values were common, no case could be attributed to resin ingress. No cases were found that combined large conductivity increases with pH depression, lithium increases, and, when sulfate analysis was available, large sulfate increases.

Conductivity data were not available for the period May 27, 1986 to March 29, 1987. Evaluation of boron, lithium and pH correlations were used to show that no resin ingress had occurred during that period.

Routine analysis for sulfate in reactor coolant was performed for plant operation from October 6, 1988, to the present. A sulfate concentration in the range of 15 to 17 ppm peak concentration was used as the indicator of cation resin ingress. This concentration is approximately equivalent to a resin ingress of 1 ft^3. No sulfate value at power exceeded 50 ppb. This is < 0.33 % of the threshold indicator. Conductivity, pH, and lithium data were also evaluated for this period as confirmation that no resin ingress had occurred. Patterns of conductivity increase and decrease from calculated values and patterns of pH increase and decrease from calculated values were observed that were similar to those seen before sulfate analysis was available (i.e., before October 6, 1988). This technical

evaluation of chemistry data after October 6, 1988, confirms the validity of the methods used prior to that date without sulfate data.

It was unnecessary to review plant records for chloride, fluoride, and oxygen. These species are not viewed as valid indicators of cation resin ingress and degradation within the primary coolant system of a PWR. Borate, chloride and fluoride anions could be associated with the anion portion of mixed bed resin (cation plus anion); however, if mixed bed resin leakage to the RCS occurred, the cation portion of the resin would contain the sulfate, pH and lithium indicators described above. Detectable dissolved oxygen in reactor coolant, during power operation with appropriate hydrogen overpressure on the volume control tank and specified residual dissolved hydrogen in the reactor coolant, could not occur and, therefore, could not be associated with the resin in-leakage.

Based on the results of this review, it is concluded that a Zorita type resin intrusion has not occurred in the past at HBRSEP.

The HBRSEP Chemistry Program has followed EPRI "PWR Primary Water Chemistry Guidelines," as applicable. Action levels have not been formally implemented, since most action level 2 items already correspond to Technical Specifications requirements, and most action level 1 items already correspond to administrative limits. Many parameters are analyzed less frequently than recommended because of the large database accumulated over 27 years of operation. HBRSEP Chemistry Program limits and frequencies are evaluated to assure that each component/system is adequately sampled and protected.