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ACCESSION NBR:9711070205 DOC.DATE: 97/11/03 NOTARIZED: NO DOCKET #
FACIL:50-261 H.B. Robinson Plant, Unit 2, Carolina Power & Light C 05000261
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SUBJECT: Responds to NRC 970814 ltr re violations noted in insp rept 50-261/97-201 on 970407-0523.Corrective actions:completed assessment of configuration mgt,review conducted on use of uncertainties in setpoint calculations by 980306.

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H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2
DOCKET NO. 50-261/LICENSE NO. DPR-23
NRC INSPECTION REPORT NO. 50-261/97-201
REPLY TO NRC DESIGN INSPECTION

Gentlemen:

The attachment to this letter provides the Carolina Power & Light (CP&L) Company response to the Design Inspection conducted at the H. B. Robinson Steam Electric Plant (HBRSEP), Unit No. 2, from April 7, 1997, through May 23, 1997. Results of the inspection were documented in NRC Inspection Report No. 50-261/97-201, transmitted by NRC letter dated August 14, 1997. The NRC requested, within 60 days of receipt of the letter transmitting the Inspection Report, a schedule for completion of the corrective actions transmitted by the report. The submittal of this reply was postponed until November 3, 1997, as documented by CP&L letter dated October 17, 1997.

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This inspection provided CP&L with valuable insights into various aspects of both specific system performance and programmatic controls. Attachment I to this letter provides information related to proposed CP&L activities that will be focused on addressing broader implications of issues identified in the report. Attachment II addresses the specific inspector follow-up items and unresolved items identified in the report.

As delineated in the Attachments to this letter, CP&L intends to use the lessons learned from this inspection to enhance safety performance at HBRSEP. Collectively the actions taken and proposed actions resulting from the NRC Design Inspection, in conjunction with the completion of initiatives already underway, will result in enhanced safety performance at HBRSEP.

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If you have any questions regarding this matter, please contact me or Mr. H. K. Chernoff of my staff.

Very truly yours,

John S. Keenan

J. S. Keenan

Attachment

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H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2
REPLY TO NRC DESIGN INSPECTION REPORT NO. 50-261/97-201

GENERIC IMPLICATIONS

As a result of the insights gained during the inspection and upon subsequent review of the inspection report, areas warranting additional review have been identified.
These areas consist of the following: 1) updating and control of calculations; 2) nonconservative design inputs and assumptions; 3) incorporation of design basis information into maintenance and test procedures; and 4) deficiencies and inconsistencies in the UFSAR and other documents. Each of these areas is discussed below. Included in the discussion are actions that will be taken to strengthen the performance of HBRSEP, Unit No. 2 in each area.

Updating and Control of Calculations

As shown by several examples identified during the inspection, improvements are warranted to strengthen both the programmatic control of calculations and enhance the performance of individuals completing or reviewing calculations. While none of the specific examples cited resulted in a significant safety concern, collectively the examples point to the need for additional reviews and action. Additionally a self-assessment focusing on control of calculations and personnel performance, including the selection and use of inputs and assumptions, will be completed by December 19, 1997. The intent of this self-assessment is to identify specific actions that can be taken to effect an overall performance improvement in this area. An open purchase order has been implemented with Westinghouse specifically for retrieval of historical calculations. This is intended to enhance the availability of selected HBRSEP design basis calculations.

As committed in the response to the NRC's October 9, 1996, 10 CFR 50.54(f) letter CP&L has completed an assessment of configuration management. This assessment also identified control of calculations as an issue. CP&L is committed to self-identification and resolution of problems.

Non-Conservative Design Inputs and Assumptions

As described above, a self-assessment focusing on control of calculations, and personnel performance, including the selection and use of inputs and assumptions, will be completed by December 19, 1997. The intent of this self-assessment is to identify specific actions that can be taken to effect an overall performance improvement in this area.

A review will be conducted on the use of uncertainties in setpoint calculations. This review will be completed by March 6, 1998. In the interim, EGR-NGGC-0153, "Engineering Instrument Setpoints," has been revised to specifically delineate a requirement for including

seismic considerations. This action addresses several of the specific examples of problems with design inputs and assumptions.

A CD-ROM copy of many of the Westinghouse WCAP reports applicable to HBRSEP has been obtained. This will allow for electronic search and retrieval of the assumptions and inputs of calculations.

Incorporation of Design Basis Information into Maintenance and Test Procedures

On February 11, 1997, CP&L responded to the NRC's October 9, 1996, 10 CFR 50.54(f) letter regarding the adequacy and availability of design basis information. In this response, three commitments were made that are intended to enhance performance related to the incorporation of design basis information into maintenance and test procedures.

The first commitment was to provide additional training on licensing and design basis to procedure technical reviewers; personnel performing unreviewed safety question determinations pursuant to 10 CFR 50.59; and engineers performing design modifications. This activity is scheduled for completion by March 31, 1998.

The second commitment was to perform self-assessments (Safety System Functional Inspection type inspections) on the effectiveness of design basis translation into procedures and configuration maintenance. HBRSEP has scheduled a self-assessment in 1998, using the Safety System Functional Inspection (SSFI) methodology, that will be similar in approach to the NRC Design Inspection. This assessment will be performed on the Component Cooling Water System. This system was selected based on its level of risk significance. The CCW system is identified by the Probabilistic Risk Assessment as the system with the highest risk significance. This assessment will be completed by December 15, 1998. The NRC Design Inspection and the scheduled CCW self-assessment will be used in conjunction with findings from the corrective action program, the operating experience program, and management judgment in determining other specific assessments to be performed.

The third commitment was to implement the Improved Technical Specifications. Implementation activities, including procedure reviews and revisions, provide an opportunity to improve the confidence level that the design bases are properly translated into plant procedures. The HBRSEP Improved Technical Specifications were approved by the NRC on October 24, 1997. Implementation is scheduled for November 1997.

These actions are structured to address both systemic improvement of the procedure development and the modification processes and to further assess the existing condition of the plant.

Deficiencies and Inconsistencies in the UFSAR and Other Documents

The response to the NRC's October 9, 1996, 10 CFR 50.54(f) letter regarding the adequacy and availability of design basis information also provided commitments addressing the potential for inconsistency between various licensing and design basis documents. These included the following actions:

First, completion of the UFSAR initiative as described in the May 30, 1996, presentations to the NRC. This review has been completed and the changes to the UFSAR identified during the review will be incorporated into the next required revision of the UFSAR. The UFSAR revision will be submitted within 6 months of the end of RO 18 in accordance with the requirements of 10 CFR 50.71(e). This review effort was effective in identifying a large number of existing inaccuracies in the UFSAR. An additional UFSAR review initiative will be completed by October 15, 1998. The follow-on review will include an examination of correspondence to verify that required information has been translated into the UFSAR, and an analysis of the Regulatory Guide positions included in the UFSAR.

Second, the completion of validations, in accordance with the Design Basis Document (DBD) program plan, of the DBDs and the Generic Issues Documents (GIDs) that were not validated previously as part of the Design Basis Reconstitution Program. This action will be completed by October 15, 1998.

Third, less significant discrepancies identified as part of the Design Basis Reconstitution Program will be resolved by December 15, 1997. Discrepancies identified as part of the Design Basis Reconstitution Program were evaluated and dispositioned during the program. This action resolves the remaining less significant discrepancies.

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2
REPLY TO NRC DESIGN INSPECTION REPORT NO. 50-261/97-201

RESPONSE TO SPECIFIC ISSUES

97-201-01-IFI: Operating Event Review

NRC Inspection Report (IR) 97-201 (Section E1.2.2.2(b)) questioned the H. B. Robinson Steam Electric Plant (HBRSEP), Unit No. 2 follow up action in regard to NRC Information Notice (IN) 91-38, "Thermal Stratification in Feedwater System Piping."

During 1991, a review of IN 91-38 was initiated. However, the program in place at the time did not adequately track this review to completion. The current Operating Experience Program at HBRSEP provides the methodology for screening, evaluating, and initiating actions for industry operating experience information, to ensure that lessons learned are used to prevent occurrences of such events and to improve plant safety and reliability. NRC Information Notices (IN), as well as other operating experience information, are screened for applicability and evaluated or otherwise dispositioned in accordance with instructions in Plant Procedure, PLP-107, "Operating Experience Program." INs requiring evaluation are assigned an evaluation due date of 60 days with approved extensions required for additional time. The review of IN 91-38 has been reinitiated and is currently scheduled by OE 97-01071 to be completed by February 15, 1998. IN 91-38 will be reviewed with the assistance of contractor resources.

An additional review of the OE program will be conducted to determine the likelihood that other items were not adequately tracked to completion prior to implementation of the current program. This review will be completed by February 26, 1998.

97-201-02-URI: Single Failure of CST Level Instrumentation

IR 97-201 (Section E1.2.4.2(a)) identified that the design of the Condensate Storage Tank (CST) level measurement system may not provide redundant power supplies, since both supplies are powered from the "A" safety train.

HBRSEP's Regulatory Guide (RG) 1.97 compliance submittals list CST level as both a type A1 and D3 variable. In the December 31, 1984, submittal, Table A, page 1 of 4 incorrectly lists LT-1454B as being provided with diesel backed (i.e., "DB") power. In Table D, page 10 of 24, LT-1454B is correctly listed as being battery backed (i.e., "BB"). The error in the Table A listing is apparently typographical. Both Table A and Table D show the CST level as being redundant. In the section of the submittal entitled, "Compliance Table - Legend," HBRSEP states, that "Redundant =Yes," is used "for instrumentation that has more than one channel; [sic] does not necessarily imply compliance with single failure, RG 1.97 separation

criteria, or channel independence." As defined in RG 1.97, Revision 3, Type A variables are those that provide primary information required to permit the control room personnel to take specific manually controlled actions for which automatic control is not provided and that are required for safety systems to accomplish their safety functions for design basis accidents.

In the worst case scenario, which includes a Loss of Offsite Power, and a failure of Emergency Diesel "A," the CST remaining level indicator would fail "low," after one hour. CST Level Transmitter (LT)-1454A is powered from Instrument Bus 1, which uses Emergency Diesel Generator (EDG) "A" as its standby power source. CST LT-1454B is powered from Instrument Bus 2, which uses Station Battery "A" as its standby power source. A Loss of Offsite Power (LOOP) combined with failure of EDG "A" would result in the immediate loss of Instrument Bus 1 and the loss of Instrument Bus 2 in one hour when Station Battery "A" was drained.

Operating Procedure (OP) - 402, "Auxiliary Feedwater System," contains a precaution that informs the operator that if CST level decreases to 10% during AFW operation, a backup water supply should be placed in service. If the CST level indication were lost, it would clearly be interpreted that the backup water supply (i.e., Service Water) should be placed in service. In this case, the information provided by the CST level indication, that is a failure of the level indication, meets the Type A definition by providing information that would permit personnel to take specific manually controlled actions for which no automatic control is provided and that are required for safety systems to accomplish their safety functions for design basis accidents. There is sufficient inventory in the CST to insure water is available for two hours. An alternate water source can be selected. CP&L believes this meets our commitments in response to RG 1.97 for this instrumentation.

<u>97-201-03-URI: Lack of Design Verification for Calculations</u>

IR 97-201 identified weaknesses concerning updating and control of calculations, non-conservative design inputs and assumptions, and incorporating design bases into maintenance and test procedures. Potential generic implications related to 97-201-03 will be addressed in Section 2.0 of this Attachment.

Calculation RNP-I/INST-1015, "CST Level Alarm Setpoints," was revised to include the effects of seismic error uncertainty in the channel uncertainty calculation. (Section E1.2.4.2(c))

Calculation RNP-I/INST-1023, "RWST Level Indication Accuracies," did not consider the instrument error uncertainty and potential vortexing in the evaluation of the Refueling Water Storage Tank (RWST) low level operator decision point as it is used in End Path Procedure 9 (EPP-9), "Transfer to Cold Leg Recirculation." Condition Report (CR) 97-00988 was initiated to address this item. The evaluation of this CR will be completed and the calculation will be revised, if necessary, by the end of RO-18, scheduled to begin on March 7, 1998. This issue has been examined in sufficient detail to determine that adequate margin exists. (Section E1.3.2.2(b))

IR 97-201 (Section E1.3.4.2(c)) identified that calculation RNP-I/INST-1058, revision 0, "Containment Water Level Instrument Uncertainty Calculation," used an incorrect tolerance, range, and required water level above the containment floor. This calculation was revised and was provided to the NRC inspection team. In addition RNP-I/INST-1109, "Containment EOP Setpoint Parameters," was updated.

IR 97-201 (Section E1.3.4.2(e)) identified that calculations RNP-I/INST-1040, "Main Steam Flow Accuracy and Scaling Calculation," and RNP-I/INST-1043, "Main Steam Pressure Accuracy and Scaling Calculation," did not account for seismic error uncertainty. CRs 97-01209 and 97-01042 were initiated to address this issue. The calculations have been revised to include the seismic error uncertainty term.

IR 97-201 (Section E1.3.5.2(a)) identified that calculation RNP-M/MECH-1620, "Evaluation of the Effects of High Energy Pipe Ruptures on the CCWS," was not in accordance with the Quality Assurance Program. Calculation RNP-M/MECH-1620 was voided, and ESR 97-00014 was revised to include information from Westinghouse and Ebasco to support that the Component Cooling Water (CCW) System inside containment is a closed system.

CRs 97-01163, 97-00993, 97-00363, 97-00996, 97-01085, 97-01070, 97-01126, 97-02175 and 97-01074 have been initiated for other calculations identified in IR 97-201 as being in error. CRs 97-01163, 97-00363, and 97-01085 have been closed. CR 97-02175 will be complete by February 18, 1998, CR 97-01074 will be complete by September 1, 1998, the remainder will be complete by March 2, 1998. (Section E1.5.2.2)

<u>97-201-04-URI: Instrument Sensing Line Slope</u>

IR 97-201 (Section E1.2.4.2(g)) identified deficiencies on 6 instruments for improper slope of the sensing lines.

The slope for one of these instrument sensing lines (i.e., sensing line for Flow Indicator (FI)-944) has been corrected. A modification to correct other identified instrument sensing lines slopes will be implemented prior to return to power following RO 18.

A walkdown of other safety related systems to determine if other differential pressure transmitters are installed in accordance with engineering procedures will be conducted during RO 18. Corrective actions will be taken as appropriate. (CR 97-01005) (Section E1.3.4.2(g))

97-201-05-URI: UFSAR Discrepancies

IR 97-201 noted deficiencies and inconsistencies in the Updated Final Safety Analysis Report (UFSAR). Approximately half of these items had been identified by HBRSEP as part of the first phase of the UFSAR review prior to the inspection.

HBRSEP is in the process of making changes to the UFSAR to correct items identified as part of the UFSAR review process. In addition to the self-identified changes being made, the following changes have been initiated: identifying the ATWS Mitigation System Actuation Circuitry (AMSAC) as a start signal for AFW has been initiated, reflecting the correct horsepower for the SDAFW Pump, correcting the Station Battery "A" cell type, accurately reflecting Motor Control Centers (MCCs) and the number of phases for the transformer supply to MCCs 9 and 10, reflecting Emergency Diesel Generator (EDG) loading calculation RNP-E-8.016, Revision 5, "Emergency Diesel Generator Static and Dynamic Analysis," and changing Figures 8.3.1.3 and 8.3.1.4 to reflect the correct CCW pump motor horsepower. (Section E1.2.6)

Engineering is evaluating other UFSAR discrepancies identified in the Inspection Report to determine if changes are necessary. The reviews involve:

The effect on the Safety Injection (SI), Residual Heat Removal (RHR), and Containment Spray (CS) pumps' Net Positive Suction Head (NPSH) due to system design and operational changes, such as having two instead of three SI pumps in service, modifying the RHR pump minimum flow recirculation line, and realigning system valves. (Section E1.3.6)

UFSAR Table 6.2.4-1 did not indicate that the discs of containment isolation valves SI-860A and B or SI-861A and B were drilled. As shown on drawing 5379-1082 sheet 5, "SI System Flow Diagram," Revision 34, the upstream discs of SI-860A and B and the downstream discs of SI-861A and B were drilled for pressure relief. (Section E1.3.6)

UFSAR Table 6.3.2-5, "Safety Injection Pump Design Parameters," indicated that the maximum SI pump flow rate was 550 gpm. WCAP-12070, calculation RNP-M/MECH-1556, "Safety Injection/Residual Heat Removal Hydraulic Model," Revision 2, as well as test results for SP-986, Revision 0, dated January 26, 1991, "Safety Injection System Flow Test," reflected maximum SI pump flows greater than 550 gpm. Additionally, the manufacturer's information stated that each SI pump could operate up to a maximum flow of 650 gpm. (Section E1.3.6)

UFSAR Section 6.3.2.2.8 (page 6.3.2-8) implied that a minimum of 300,000 gal was "available for delivery" from the RWST. The tank curve shows 277,999 gal. (Section E1.3.6)

UFSAR Section 6.3.2.2.17 (page 6.3.2-15) stated that the SI system high pressure branch lines were designed for a pressure of 1500 psig. The appropriate value is 1750 psig. (Section E1.3.6)

Evaluation of these items will be completed and any additional UFSAR changes will be initiated by March 2, 1998.

97-201-06 & 07-URI: Notification of Changes in PCT - Reporting of Significant PCT Changes

IR 97-201 (Section E1.3.2.2(a)) discusses inconsistent reports made to the NRC pursuant to 10 CFR 50.46. HBRSEP has not been reporting significant errors or changes in the application of each evaluation model for both Large Break and Small Break Loss of Coolant Accidents, including recirculation cooling applications. Additionally, HBRSEP has not been reporting nonsignificant errors and changes in the application of evaluation models annually as required by 10 CFR 50.46.

The most limiting Peak Clad Temperature (PCT) from analyses of a Loss-of-Coolant Accident (LOCA) prior to May 31, 1995, was 2134°F for the Large Break LOCA (LBLOCA) analysis during the injection phase. The PCT value of 2134°F was reported to the NRC by letter dated May 20, 1993. By letter dated June 28, 1995, a reduction in calculated PCT from 2134°F to 2006°F was reported to the NRC for the LBLOCA during the injection phase. This report did not include information regarding the PCT associated with application of the LOCA Evaluation Model (EM) for ECCS recirculation cooling after a LBLOCA. As stated in CP&L letter dated October 16, 1997, unreported significant changes were made in the application of a LOCA EM in January 1992, October 1993, and September 1994, for ECCS recirculation cooling after a LOCA. The latest resulting PCT from ECCS recirculation cooling prior to May 31, 1995, was a value of 2102°F for the LBLOCA during switchover to recirculation. By letter dated April 24, 1996, an increase in calculated PCT from 2006°F to 2064°F was reported to the NRC for the LBLOCA during the injection phase. This report did not include information regarding the PCT associated with application of the LOCA EM for ECCS recirculation cooling after a LBLOCA. By letters dated October 14, 1996, and October 25, 1996, the reported limiting PCT for the LBLOCA was raised from 2064°F to 2128°F, and the reported limiting PCT has remained above the PCT for ECCS recirculation cooling after a LBLOCA since that time. Therefore, during the time period from June 28, 1995, and October 14, 1996, the most limiting PCT from a LOCA analysis was not adequately reported to the NRC.

An annual notification of nonsignificant errors and changes in application of evaluation models will be submitted by December 4, 1997.

97-201-08-IFI: Evaluation of Transfer to Cold Leg Recirculation

IR 97-201 (Section E1.3.2.2(a)) identified a concern regarding the acceptability of a second Peak Cladding Temperature (PCT) peak that occurred during the switchover from injection to recirculation during a LBLOCA. The IR stated that the issue was referred to NRR for resolution.

By letter dated August 26, 1997, the NRC requested additional information regarding the justification for implementation of the emergency procedure that provides guidance for the switchover from injection to recirculation during a LBLOCA. The HBRSEP response was provided by letter dated October 14, 1997. The response discussed the methodologies used for analysis of the LBLOCA transient, conservatisms in the assumptions used, a discussion of previous research regarding heating and quenching effects on Zircaloy, and some basic testing conducted by Siemens Power Corporation.

The response stated that, as a result of the conservatism in the LBLOCA analysis, the data contained in the literature references, and the limited experimental data obtained by Siemens that show minimal impact on materials properties following the two short duration temperature excursions, it is concluded that long-term coolable geometry is not jeopardized.

97-201-09-URI: SI, RHR, and CS Pump NPSH

IR 97-201 (Section E1.3.2.2(c)) expressed a concern with the ability to demonstrate the most limiting values for NPSH for the SI, RHR, and CS pumps.

A review of calculations related to the available Net Positive Suction Head (NPSH) for the SI system determined that following a postulated LBLOCA, and assuming a single active failure of one SI pump, the remaining operating SI pump might not have sufficient NPSH under certain operating equipment configurations to respond to a design basis accident. The NRC Operations Center was notified of this condition on June 27, 1997. This condition was caused by personnel error. Personnel involved in a 1988 design change to the SI system did not adequately assess the impact of a single pump on SI system flow and pump NPSH requirements. Assuming a single failure, failure of the other SI pump could lead to increased fuel cladding temperatures and fuel damage during a LOCA, which could result in an increase in consequences beyond those considered in the current safety analysis. Therefore, during the period of inadequate available NPSH, the plant operated in a condition outside of the design basis. Modifications have been implemented to assure adequate NPSH is available to respond to a design basis event. ECCS flow modeling and calculations will be completed by December 1, 1997, to document the capability of the ECCS system's pumps to perform their design functions. A preliminary review indicates adequate NPSH for both injection and recirculation. The SI system piping will be modified to gain additional NPSH margin during RO 18.

CR 97-01217 evaluated NPSH concerns relative to the SI and RHR pumps. Westinghouse calculation RFS-CP-872 compares required and available NPSH for the SI, RHR, and CS pumps and states that "the high head SI pump is the limiting factor" during injection. This means that whenever the SI pumps have adequate NPSH, the CS pumps will also. In addition, in the injection mode, the calculation shows RHR with a large NPSH margin. Calculation RNP-M/MECH-1637, "Determination of Available NPSH for ECCS Safety Injection Pumps A, B, C in the Injection Mode of Operation," performed an analysis of the current plant configuration (incorporating level changes made in 1997) and concluded that acceptable NPSH is available for the SI pumps. The RHR and CS pumps have lower required NPSH than the SI

pumps and, therefore, are not analyzed further for the injection configuration. LER 97-008-00 was issued addressing the period of time when SI pump NPSH was not acceptable.

97-201-10-URI: SI Valve Testing

IR 97-201 (Section E1.3.2.2(e)) raised questions concerning the American Society of Mechanical Engineers (ASME) Code requirement to classify valves SI-851A, SI-851B, and SI-851C as "active," and concerning testing these valves in accordance with ASME Code requirements.

Technical Management Manual (TMM) procedure TMM-004, "Inservice Inspection Testing," has been revised to reclassify valves SI-851A, SI-851B, and SI-851C as "active," with a requirement to stroke time test closed, fail safe test, and full stroke exercise at cold shutdown intervals, in addition to remote position indication testing at a two year interval. This has been determined to be a conservative action based on NUREG-1482, "Guidelines for Inservice Testing at Nuclear Power Plants," section 2.4.2, which states, "If a valve is routinely operated during power operations, it is considered an 'active' valve. A valve need not be considered 'active' if it is only temporarily removed from service for a short period of time." The operation of these valves is performed on an infrequent basis, when they are opened for a short period of time. The reclassification of these valves considered the impact of failure, and it is believed that the performance of this test is a prudent and conservative action. Therefore, a test procedure will be developed to test these valves and the valves will be tested prior to restarting from the next cold shutdown of sufficient duration.

97-201-11-IFI: Classification of Valves in IST

IR 97-201 identified several valves that perform accident mitigating functions and that are not included in the in-service testing program. These included SI valves used to realign the SI system in the event of an accident while filling the accumulators, and the CCW and SW system valves used to isolate leaks in the room containing the RHR pumps.

CR 97-01981 evaluated the requirements and determined that valves SI-856A, SI-856B, SI-864A, and SI-864B should be ASME Code Category A. The Category A list in TMM-004 will be revised by December 11, 1997, and the in-service testing program will test these valves in accordance with Category A requirements. (Section E1.3.2.2(e))

Valves CC-927 and 928 and SW-906 and 907 will be tested as active valves. (Note: incorrect valve numbers were used in IR 97-201 for SW valves. Valve numbers SW-75 and 77 as stated in the Report should be SW-906 and 907). (Section E1.3.2.2(f))

97-201-12-URI: 10 CFR 50.59 Screening Deficiency

IR 97-201 (Section E1.3.2.2(f)) noted that the valve stroke time for valves RHR-744A and B was increased from 10 seconds to 16 seconds. The 10 CFR 50.59 evaluation for the activity that changed the stroke times incorrectly stated that the change did not affect the UFSAR. Section 6.3.2.2.12 of the UFSAR states that valves that receive an SI signal were supplied with 10 second operators.

CR 97-01261 was initiated due to ESR 96-00012 not correctly identifying the need to perform an unreviewed safety question determination, since it was not recognized that the change would result in a change to the UFSAR. The evaluation is complete and the conclusions remain unchanged that the change would not invalidate the assumptions regarding delivery of RHR flow. Action has been taken to address the oversight with the personnel that were responsible. A change to the UFSAR has been initiated to clarify the valve stroke time description.

97-201-13-IFI: RAB Flooding due to SW Passive Failure

IR 97-201 (Section E1.3.2.2(g)) questioned how a Service Water (SW) system passive failure was controlled such that flooding would not disable safety related equipment within the Reactor Auxiliary Building (RAB).

In the event of a SW system passive failure in the RAB, leakage would be routed to the floor drain system. If the leakage exceeded the floor drain system's capability, the water would be distributed throughout the RAB. If the leak could not be isolated, the water could eventually overflow the building's door dikes, and spill outside the RAB. However, safety-related equipment in the RAB is elevated above the height of the door dikes. Therefore, no additional action is necessary to be taken.

97-201-14-URI: SI Cable Separation Deficiency

IR 97-201 (Section E1.3.3.2(a)) noted a weakness regarding 10 CFR 50, Appendix B, Criterion III, "Design Control," in that the design basis for electrical systems was not properly implemented into work instructions dictating the correct routing for SI pump control cables.

On May 21, 1997, with HBRSEP operating at 100% power, engineering personnel discovered that auto-start and manual start cables for SI Pump "C" were routed in the same cable tray stack as the auto-start cables for SI Pumps "A," and "B" when aligned in the safety Train A configuration. This condition was outside of the design basis of the plant. When the discovery was made the SI Pump "C" was taken out of service and the SI Pump "B" was place in service. This eliminated the separation concern since wiring control wiring for SI pump "A" and "C" was the wiring not separated. This condition was reported as LER 97-006-00.

A modification was implemented on May 25, 1997, that replaced the cables with cables using

new routes that achieved normal design basis separation. This condition was caused by an original plant design installation error. The results of a review concluded that the cable routing error did not introduce a condition that had significant adverse safety consequences. A review of other SI pump cables was performed, and additional deviations to the redundant cable separation criteria that were identified were dispositioned. Current electrical installation practices provide conservative separation criteria that is based on the concepts of safety train separation. Additional emphasis will be placed on incorporating the review of electrical separation during appropriate engineering self-assessments.

97-201-15-IFI: SI Pump Motor Load Evaluation

IR 97-201 (Section E1.3.3.2(a)) noted that a service factor of 1.0 was used for SI pump motors A, B, and C in calculation RNP-E-5.004, "Ampacity Evaluation of Safety Related Power Cables on 480V and 208V AC MCC's and Busses." Investigation revealed that the actual service factor was 1.15 and that the three SI pump motors could operate in excess of their nameplate rating. CR 97-01194 was issued to track this condition, and ESR 97-00276 was issued to evaluate the effects on the motors, cables, breaker protection, and supporting systems.

CR 97-01194 and ESR 97-00276 evaluations were completed. They demonstrated operability of the SI pump motors, cables, breaker protection and supporting systems. As a corrective action, a review of design calculations to determine if other similar safety-related motors may be required to operate at greater than rated horsepower under an accident condition will be completed by December 1, 1997.

97-201-16-IFI: Seismic Qualification of 480VAC Circuit Breakers

IR 97-201 (Section E1.3.3.2(c)) noted that the seismic qualification of the 480V breakers feeding the B SI pump motor while the breakers are in the removed position had been identified as outliers in the verification of seismic adequacy of electrical equipment. HBRSEP stated that the breakers would either be modified, or an analysis would be performed to demonstrate seismic adequacy in the removed position.

Calculation RNP-C/EQ-1137, "E1/E2 Switchgear Modifications and Seismic Qualification of DB-50 Circuit Breakers," has been revised to include the analysis of the acceptability of an Emergency Bus Circuit Breaker in the "Disconnect" and the "Test" positions. This calculation concludes that seismic braces are not required for lateral restraint in the "Connect," "Test," "Disconnect," or the "Racked Out" positions. As stated during the inspection, the concern with the breakers as seismic outliers involved a question regarding whether seismic retrofitting was needed when the breakers were in the racked out position. The breakers are not seismically qualified when in the removed position (i.e., physically removed from the cabinet); however, they are attended by plant personnel.

97-201-17-URI: SI Accumulator Pressure Alarm Setpoint

IR 97-201 (Section E1.3.4.2(a)) questioned, when considering instrument uncertainty, the SI Accumulator high pressure alarm setpoint and the Accumulator pressure relief valve setpoint overlap. This setpoint was changed in 1972, and no setpoint calculation or uncertainty calculation is available to support the new setpoints. This is considered a weakness with regard to 10 CFR 50, Appendix B, Criterion III, "Design Control," and Criterion XVI, "Corrective Actions."

To address this concern, a modification was developed and completed which changed the Accumulator high pressure alarm from 680 psig to 646 psig.

97-201-18-URI: RWST Level Instrument Uncertainty

IR 97-201 (Section E1.3.4.2(b)) questioned whether the seismic uncertainty factor should have been included in the uncertainty calculation for the RWST level instrument, and in similar calculations for other accident-mitigation instruments.

Calculation RNP-I/INST-1023, "RWST Level Indication Accuracies," did not consider the instrument uncertainty and potential vortexing on the evaluation of RWST low level operator decision point as it is used in EPP-9. Condition Report (CR) 97-00988 was initiated to address this item. The evaluation of this CR will be completed and the calculation will be revised, if necessary, by the end of RO 18. This issue has been examined in sufficient detail to determine that adequate margin exists.

There are two classes of accidents that take credit for the RWST water: a Loss of Primary Inventory and a Loss of Secondary Inventory. The limiting event in each category is a LOCA and a Main Steam Line Break (MSLB). The RWST level indication channels perform their only safety function following a LOCA. At HBRSEP, a LOCA coincident with a seismic event is outside the design basis of the plant. A MSLB requires enough water from the RWST to deal with shrink of the RCS inventory and to ensure shutdown margin. This is a relatively small quantity. A large inaccuracy in RWST level for MSLB would be of little importance. Consideration of seismic error in determination of RWST level indication uncertainty is not considered necessary.

97-201-19-IFI: Containment Water Level Setpoint and Instruments used in Emergency Operating Procedures (EOP) and Abnormal Operating Procedures (AOP)

IR 97-201 identified several discrepancies in the plant procedures. For example, the containment floor water level setpoint used in an emergency plant procedure to initiate post-accident recirculation did not agree with the UFSAR value. The emergency procedure value may not have included the appropriate instrument uncertainty.

The IR identified that calculation RNP-I/INST-1058, revision 0, used an incorrect tolerance, range, and required water level above the containment floor. Revision 1 to this calculation was completed and provided to the NRC during the inspection. (Section E1.3.4.2(c))

CR 97-01221 addressed the use of non-Category 1 RG 1.97 instruments specified for use for operator actions as part of the AOPs/EOPs. CR 97-01221 concluded that HBREP will follow the Westinghouse Owners Group (WOG) guidance concerning accuracy of these instruments, where it is provided. There are no regulatory commitments made by HBRSEP concerning the use of non-Category 1 RG 1.97 instruments in the AOP/EOP applications.

97-201-20-IFI: CCW System Overpressurization

IR 97-201 (Section E1.3.5.2(a)) questioned whether the CCW system would function properly with the vent valve gagged open and with the design basis inleakage of 260 gpm from a ruptured Reactor Coolant Pump (RCP) thermal barrier. The Report also questioned the possibility of overpressurizing the system using the demineralized or primary water systems used to supply makeup to the surge tank.

In the event of a ruptured Reactor Coolant Pump (RCP) thermal barrier cooling coil, a radiation monitor (i.e., R-17) would alert operators to the RCS leakage into CCW System. Inleakage in the CCW System would also be apparent from a high CCW Surge Tank level alarm set at 55 percent of the tank level. The 2000 gallon capacity tank is normally maintained at a level between 45-55 percent. Upon high RCP thermal barrier cooling water flow (i.e., 90 gpm), a high flow alarm received from FIC-626 would alert operators. If the flow rate reached 100 gpm, containment isolation valve FCV-626 would automatically close preventing inleakage into the CCW System. If the automatic isolation does not occur, valve CC-735 could be closed from the control room prior to the tank becoming water solid. Closure of FCV-626 or operator action will prevent the CCW Surge Tank and CCW pump discharge piping from becoming overpressurized. The limiting inleakage flow rate from a ruptured RCP thermal barrier cooling coil to the CCW Surge Tank is 260 gpm. Westinghouse letter VPA-89-54 dated May 31, 1989, regarding a similar plant, indicates this maximum leakage would be only 7.5 gpm, this letter was not used to evaluate this condition. Conversations with Westinghouse indicated that at HBRSEP, this value would not exceed 10 gpm. This indicates that the 260 gpm, from an earlier letter, is very conservative.

The maximum makeup flow rate is 150 gpm. In addition, the pressure of the RCS is significantly greater than the makeup pumps discharge pressure (i.e., the demineralized and primary water pumps). Inleakage/pressurization as a result of a ruptured RCP thermal barrier cooling coil is therefore more limiting than inleakage/pressurization from the makeup system. This issue was examined in sufficient detail to conclude that overpressurization is not a concern.

97-201-21-URI: Translation of Design Bases into Drawings, Procedures, and Installed Components

IR 97-201 noted that the design basis for thermal relief valves was not correctly translated into the drawings that controlled the as-built plant as required by 10 CFR 50, Appendix B, Criterion III, "Design Control." Also, discrepancies were noted in the procedures regarding testing of the station batteries, that appeared to be in violation of the requirements of 10 CFR 50, Appendix B, Criterion XI, "Test Control." Additionally, discrepancies related to the sizing of battery intercell connections, the inclusion of battery hydrometers, and protruding cable appear to be examples of violations of the requirements of 10 CFR 50, Appendix B, Criterion III, "Design Control," in that the design basis was not correctly translated into installed components because of incorrect drawings, procedures, and work instructions.

A clarification of code requirements was obtained from ASME and Reedy Engineering, Inc. No Section of the ASME Boiler and Pressure Vessel Code requires that thermal relief devices be used to protect either vessels or piping from overpressurization. The design bases for the thermal relief valves are included in the CCW Design Basis Document (DBD). The CCW DBD was revised to delete errant statements regarding CCW pump suction pressure.

The Maintenance Surveillance Test (MST) - 921, "Station Battery Service Test," voltage limit will be revised to a value greater than 1.0 VDC in accordance with IEEE 450-1980. Voltage values from the last completed MST did not drop below 1.0 VDC. MST-921 will be revised by March 2, 1998 and will be performed during RO-18. (Section E1.4.2.3)

Benchmark values for intercell resistance have been computed from readings taken when the present batteries were installed. In accordance with the vendor technical manual, any subsequent reading which exceeds the benchmark by more than 20 percent requires correction. This 20 percent value is 31 micro-ohms and 50 micro-ohms for the "A" and "B" batteries, respectively, and has been incorporated into Preventative Maintenance (PM) procedures PM-410, "Inspection of Battery Bank and Cell Connections" and PM-411, "Disassembly, Cleaning, Assembly and Testing of A and B Station Battery Cell Connections." A review of data from the last PM indicates that these values had not been exceeded. (Section E1.4.2.3)

Vendor drawings for the batteries, 063969D and 063970D, and CP&L drawings HBR2-8237 and HBR2-8236, were revised on May 22, 1997, to match the vendor technical manual. (Section E1.3.5.2(a))

The Station Battery hydrometer wells were removed on May 20, 1997. (Section E1.4.2.5)

The lugs that connect the cables to the Station Battery will be replaced during RO 18. (Section E1.4.2.5)

CRs 97-01152, 97-01012, 97-01138, 97-01070 determined that the latest design for Station Battery "A" and "B" is not reflected in the battery MSTs. The CR evaluations determined that no operability issues exist. However, the Station Battery "A" and "B" calculations will be

revised and the design details will be incorporated into affected MST procedures prior to RO 18. (Section E1.4.2.3)

97-201-22-IFI: Ampacity Derating of Cables

IR 97-201 noted that in Calculation RNP-E5.004, "Ampacity Evaluation of Safety-Related Power Cables on 480V and 208V AC Motor Control Centers and Buses," Revision 4, no ampacity derating was included for cables routed in fire stops and seals or for cables with fire wrapping. Cables were not derated because the impact was considered to be negligible, but no basis for this conclusion was provided. The Calculation was not updated to reflect the fire wrapping of certain cables in the CCW system. Further, the report questioned the need to provide design guidance for ampacity due to localized heat sources.

CR 97-01155 determined that there is no immediate adverse impact of Ampacity derating for CCW "A" and "C" conduit firewraps. Corrective actions are to: 1) revise procedure EGR-NGGC-0103 to incorporate additional guidance regarding fire stops, and 2) revise HBRSEP ampacity calculations (i.e., RNP-E-5.001, "Ampacity of 600V Power Cable in Tray," RNP-E-5.004, RNP-E-5.018, "Ampacity Evaluation of 125VDC and 120VAC Power Cables," and RNP-E-5.019, "Ampacity Evaluation of Non-Safety Related 480V AC Bus, MCC and Power Panel Cables") to incorporate revised EGR-NGGC-0103 guidance. These actions are scheduled to be completed by August 31, 1998. (Section E1.4.2.1)

CR 97-01085 determined that no additional derating factors are required for the conduit firewraps on CCW "A" and "C." ESR 97-00489 and Design Change Backup Form (DCBF) RNP-E-5.004-0001 were generated to document the basis for no additional derating. No additional actions are planned to address ampacity derating for CCW "A" and "C" conduit firewraps. (Section E1.4.2.1)

With regard to ampacity derating for Intumastic Fireproof Cable Coatings, CR 97-01219 determined that no additional derating factors are required for the installed conditions of the Intumastic Fireproof Cable Coatings. ESR 97-00488 and DCBF RNP-E-5.038-0001, "Evaluating the Effect of Intumastic 285 Fireproof Cable Coatings," were generated to document the basis for no additional derating. (Section E1.4.2.1)

Ampacity derating for localized heating sources were evaluated and CR 97-01196 determined that no additional derating factor is required for the subject installation. CR 97-01196 corrective action is to revise procedure EGR-NGGC-0103 to incorporate additional guidance regarding cable derating due to localized heat sources. This procedure is scheduled to be revised by June 30, 1998. (Section E1.4.2.5)

CR 97-01090 was initiated to track resolution of cable pulling activities, including monitoring cable sidewall pressure and tension during cable pulling exercises. These issues are currently scheduled to be resolved by May 1, 1998. System engineers responsible for electrical

modifications will delineate appropriate cable pulling techniques prior to completion of this effort by May 1, 1998. (Section E1.4.2.5)

97-201-23-IFI: Agastat Relay Lifetime

IR 97-201 (Section E1.4.2.1) questioned the replacement schedule for Agastat E7000 series relays.

This issue was discussed with the NRC Inspector during a telephone conversation on September 12, 1997. The NRC was concerned that the Agastat Relay Lifetime test report addressed Agastat 7000 series relays, while the application at HBRSEP utilized Agastat E7000 series relays. As explained during the telephone conversation, the Agastat 7000 series relays are no longer available for use in safety-related applications, and the Agastat E7000 relays are the qualified replacement. Documentation to this effect was provided to the Inspector on September 15, 1997.

97-201-24-IFI: Station Battery "B" Rating

IR 97-201 (Section E1.4.2.2) questioned the sizing of station battery "B". A review of the Calculation RNP-E-6.020, "Load Profile and Battery Sizing Calculation for Battery B," revealed that the battery sizing used the published 1 minute discharge rates and *the coup de fouet* effect was not included. Testing by the manufacturer was not performed on the type of battery installed as Station Battery "B".

HBRSEP contacted the battery supplier, and testing was initiated to confirm the manufacturers data. Results indicate that published rates for the MCX-9 type battery at the "B" batteries designed end voltage are valid.

97-201-25-URI: Field Flash Battery Testing DSDG

IR 97-201 (Section E1.4.2.3) identified that the procedures for performance testing of certain batteries had several deficiencies.

Calculations RNP-E-6.028, "DS Battery Sizing Calculation," and RNP-E-6.029, "DSDG Battery Sizing Calculation," have been completed and demonstrate adequate capacity from a design standpoint. A service (i.e., duty cycle) test for both the DS and DSDG batteries will be developed to demonstrate battery operability. Procedures will be developed by March 2, 1998.

97-201-26-IFI: Station Battery Test Control Deficiencies and Test Procedure Revisions

IR 97-201 (Section E1.4.2.3) identified test and procedure discrepancies related to the testing of station batteries.

Documentation of the current applied during performance of MST-921 for station batteries "A" and "B" was inadequate in that duty cycle changes at 1 minute and 59 minutes could not be substantiated with the data provided. The batteries are operable; however, the data gathering device was not appropriate for the test (data gathered at the wrong intervals).

A procedure change will be developed to ensure that MST-920, "Station Battery Performance Testing (Five Year Interval)," capacity calculation is computed from the last data recorded at a terminal voltage greater than or equal to 105VDC. Calculations and procedures will be revised to incorporate actual test values into MST-921. These actions will be complete by March 2, 1998, in support of the next scheduled performance of the affected procedure.

<u>97-201-27-URI: Control of Calculations</u>

IR 97-201 (Section E1.5.2.1) identified that several calculations that performed the same or similar analyses with different input data and conclusions were currently active, with none identified as voided or superseded. Additionally, one calculation was performed without the required design verification. Review of this concern indicates that control of calculations does not meet the level of commitment to UFSAR Section 17.13.2.2.4, which specifies that design documents and procedures are controlled to reflect design modifications and "as-built" conditions. In addition, UFSAR Section 17.3.2.2.5 specifies that design interfaces are defined and controlled, including review, approval, release, and distribution of design documents and revisions. This represents a weakness with the design control measures required by 10 CFR 50, Appendix B, Criterion III, "Design Control."

Each of the examples identified in the IR were evaluated, and corrected as deemed appropriate.