UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
101 MARIETTA STREET, N.W.
ATLANTA, GEORGIA 30323

Report No.: 50-261/92-19

Licensee: Carolina Power and Light Company
P. O. Box 1551
Raleigh, NC 27602

Docket No.: 50-261 License No.: DPR-23

Facility Name: H. B. Robinson

Inspection Conducted: June 15-19, 1992

Inspectors: C. Smith 7/15/92 Date Signed

for M. Hunt Frank Jape 7/15/92 Date Signed

Approved by: Frank Jape 7/15/92 Date Signed
F. Jape, Chief
Test Programs Section
Engineering Branch
Division of Reactor Safety

SUMMARY

Scope:

This special, announced inspection was conducted to examine the licensee's program for self-assessment of problems, followup of a previous inspection item, and to review results of testing as related to Generic Letter 89-10.
Results:

The requirements of T.S. Section 6.5.1.6, concerning oversight activities of the PNSC with regard to reviews of LERs, ACRs and SCRs, were verified as having been satisfied. The licensee performs root cause analysis of LERs, ACRs, and SCRs using investigative techniques such as Change Analysis, Barrier Analysis, and Events and Causal Factors Charts. These root cause analyses were found to be generally acceptable with a few exceptions. The developed corrective action plans were consistent with the identified root causes and the corrective

action program monitored implementation of the corrective actions to assure completion. A previously identified item related to GL 89-10, NRC Report No. 50-261/91-201 was closed. The review of the calculation for MOV FW-V2-6A, related to the Generic letter 89-10 MOV Program identified one violation, 50-261/92-19-01, of Criterion III to Appendix B 10 CFR 50. (paragraph 4a.)

REPORT DETAILS

1. Persons Contacted

   Licensee Employees

   *R. Barnett, Manager, Outages and Modification
   *C. Baucom, Project Specialist, Regulatory Compliance
   *W. Biggs, Manager, Nuclear Engineering Department Site Unit
   *S. Billings, Technical Aide, Regulatory Compliance
   *R. Chambers, Plant General Manager
   *S. Farmer, Manager, Engineering Programs
   *W. Gainey, Jr., Manager, Plant Support
   *M. Grantham, Nuclear Engineering Department, HESS/Mechanical
   *J. Harrison, Manager, Regulatory Compliance
   *P. Musser, Manger, Engineering and Technical Support, Nuclear Assurance Department
   *J. Pearson, Nuclear Engineering Department, HESS/Mechanical
   *D. Stadler, Onsite Licensing Engineer/Nuclear Licensing

   NRC Resident Inspectors

   *L. Garner, Senior Resident Inspector
   *C. Ogle, Resident Inspector

   *Attended exit interview

2. Action on Previous Inspection Findings (92702)
   Closed, VIO, 50-261/91-201

   During an inspection of the licensee's Generic Letter (GL) 89-10, "Safety-Related Motor-Operated Valve Testing and Surveillance Program," a violation was identified. Report No. 50-261/91-201, was issued on July 25, 1991, and the notice of violation was forwarded in a letter issued October 4, 1991. The violation concerned the lack of documentation of corrective actions taken for MOV FW-V2-6A, Feedwater Isolation Valve, which had a galled valve stem. As a result of the violation the licensee has revised its corrective action program to require complete documentation. In addition the operations procedure for shutdown of the unit was revised to require cycling of the three feedwater block valves during cool down to prevent thermal binding of these valves which was determined to be a contributor to the opening difficulties on valve FW-V2-6A. Additionally, during the recent outage the valve stem was replaced. This violation is closed.

3. Evaluation of Licensee Self-Assessment Capability (40500)

T. S. Section 6.5.1.6 describes the responsibilities of the PNSC and specifies reviews to be performed by this onsite organization. Violations of the TS are required to be investigated and a report prepared that evaluates the event and provides recommendations to prevent recurrence. The PNSC reviews these reports and evaluates the adequacy of the developed corrective action plan. Additionally, station problems documented as ACRs, or SCRS, are reviewed by the PNSC to ensure that an adequate root cause analysis has been performed and an effective corrective action plan has been developed. The inspectors performed an evaluation of the licensee's self-assessment capability by reviewing monthly PNSC meeting minutes covering a period from March 15, 1991 to March 20, 1992. Selected action items dispositioned by the PNSC at these meetings were independently reviewed using the investigative techniques delineated in the "NRC HPIP Procedure and Module Manual."

a. Licensee Event Reports

The inspectors reviewed the events and their root causes documented on the following LERs and evaluated the proposed corrective actions to determine their adequacy.

- LER No. 91-004, Rod Control System Urgent Failure
- LER No. 91-007, Failure to Perform Surveillance Test
- LER No. 91-009, Over Temperature Delta Temperature Channel Inoperable due to Summator Module Lag Constants
- LER No. 91-013, Diesel Driven Fire Pump Inoperable
- LER No. 92-002, Failure to Test all Circuits Associated with the Auxiliary Feedwater Auto-Start
- LER No. 92-004, T.S. Violation During ILRT

The root causes for these plant events ranged from random hardware failure to human errors and procedural deficiencies. The inspectors determined that the root causes identified by the licensee were for the most part correct with exception of the following examples. The root cause for the event documented on LER 91-007 was given as human error. Application of the guidance delineated in the NRC HPIP manual identified the root cause to be inadequate communications which resulted in less than adequate shift-turnover. The licensee's developed corrective plan was considered adequate, however, in that administrative controls were established to ensure adequate communication and work control during shift changes. Additionally, plant personnel were indoctrinated on the use of the new administrative controls. Another example of inadequate root cause analysis was identified on LER 91-013. The licensee identified the root cause as failure of the design engineering program to replace existing 100°-120°F thermostats with 120°-140°F thermostats. The inspectors discovered, however, that the primary causal factor was failure of the CAP to initiate corrective action for a station problem that was identified in October 1989 and which was documented on WR/JO 89-AJMF1. Discussions with licensee's personnel revealed that the CAP was in a process of transition at the time the deficiency was identified. This probably was the reason why it failed to initiate corrective action for an identified and documented station problem. A definitive evaluation of this root cause can not be made, however, because of its indeterminate status. The licensee's implemented corrective action for LER 91-013 was considered adequate based on change out of the thermostats and an increase in the power rating of the associated heater.

b. Adverse Condition Reports/Significant Condition Reports

ACRs and SCRs dispositioned by the PNSC during regular monthly meetings were independently reviewed by the inspectors for root causes in order to evaluate the licensees self-assessment capability. Objective evidence reviewed during this effort are listed as follows:

SCR No. 89-022

SCR No. 89-015

SCR No. 89-018

SCR No. 90-013

ACR No. 91-009

ACR No. 91-034

ACR No. 91-283

ACR No. 91-286

ACR No. 92-20

The inspectors determined that the licensee used various investigative techniques during the performance of root cause analyses. Among these were Barrier Analysis, Changes Analysis, and Events and Causal Factor Charts. Based on review of the above ACRs/SCRs the inspectors concluded that the root causes identified by the licensee were generally correct. ACR No. 91-286 was a typical example and involved failure of the narrow range OTDT RTD instrumentation circuit to meet TS requirement of 0.75 seconds time delay. This event was also reported to the NRC on LER 91-009-01. The licensee correctly identified the root causes documented on ACR No. 91-286. Additionally, the inspectors reviewed the close-out package and verified that the developed corrective action plans for the three primary causal factors specified in the LER had been completed in accordance with the Licensee's commitments. SCR No. 88-022 further demonstrated the licensee's capability to perform effective self-assessments. This SCR involved an event wherein the reactor vessel cavity boron concentration was inadvertently diluted to less than 1950 ppm during a RFO with fuel off loaded. The inspectors used the guidance of the NRC HPIP Module and determined the near root causes to be in the functional areas of training, procedures, supervision, and communications. The licensee's developed corrective action plans for the seven causal factors identified in the Event and Causal Factor Chart fell within the functional areas identified by the NRC HPIP module.

Some inadequate root cause analyses were identified by the inspectors. Typical of this small sample was ACR No. 91-034. This ACR involved an event related to inadequately revised calibration procedures required per plant modification M-959. Licensee management determined the root cause to be indeterminate. The inspectors, however, identified the near root causes as inadequate design control. Specifically, the postmodification/calibration test requirements and test acceptance criteria were not adequately specified in the plant modification package. The immediate corrective action of revising the loop calibration procedures to be technically adequate was necessary but not sufficient to prevent recurrence of a similar problem. Discussions with the senior resident inspector revealed that the licensee's response to an NOV involving a civil penalty more adequately addressed the required corrective actions for the deficiency documented on ACR 91-034.

The inspectors also selected ACR No. 92-186 for review of problem assessment activities. The ACR was written to identify a tripping condition which occurred while the A Emergency Diesel Generator

(EDG) was undergoing an over-speed trip test. A similar condition had occurred on a B EDG while it was undergoing post maintenance overspeed trip testing. In each instance, the same condition, fuel rack unlatched, was found. The A EDG was operational at the time the trip occurred, while the B EDG was still in post maintenance test status.

As a result of the EDG A trip, an investigation team was organized to determine the cause of the fuel rack unlatched condition. The team consisted of knowledgeable engineering personnel, an operations person and a maintenance supervisor and craftsman. Additional personnel provided special assistance when needed.

The team conducted root causes analysis which included a description of the event, equipment failure/conditions affecting the event, a chronological description, and summarized the factors that influenced human behavior. A list of proposed corrective actions to preclude recurrence was prepared. Corrective actions for contributing factors and improvements based on the investigation were recommended.

The team used various causal factors check sheets to assess each aspect of the event. These check sheets were applicable to any investigation and contained a group of questions that could be applied to any situation.

The inspectors concluded that the investigation was thorough and the method of reaching the solution was acceptable. The recommendation seemed to fit the findings of the investigating team.

c. Conclusion

The inspectors concluded that the licensee management generally performed an adequate root cause analysis for LERs. ACRs and SCRs. The developed corrective action plans were also consistent with the identified root causes to ensure implementation of effective corrective actions. Additionally, the corrective action program monitors implemented corrective action plans to verify completion of corrective actions. The inspectors attended the PNSC monthly meeting on June 17, 1992 to observe the depth of review of overall plant performance. The meeting was well conducted with a prepared agenda. The agenda items were presented to the committee members in a clear and understandable manner, and the committee reviews were thorough and in depth.

Within this area no violations or deviations were identified.

4. Generic Letter 89-10. Safety-related Motor-Operated Valve Testing and Surveillance (TI 2515/109)

The inspectors performed a limited review of the licensee's Motor Operated Valve (MOV) program.

a. Differential Pressure Testing (DP)

The testing of MOVs, either static or under DP conditions is performed using VOTES diagnostic equipment. The acceptance criteria is furnished to the testing personnel by the licensee's Nuclear Engineering Department (NED). The traces produced by the diagnostic equipment are screened by on-site personnel to verify the required thrust is developed and is within the thrust window. The traces are also reviewed for acceptable motor current, packing load, and verification that the maximum thrust at torque switch trip (TST) is below the maximum allowable thrust. Once these items are verified, the trace data is forwarded to NED for detailed analysis. The inspectors were advised that if during the NED review a discrepancy is found, the site is notified and corrective action is initiated.

The inspectors reviewed the test traces with the licensee representatives for the following valves:

Valve ID Number	Thrust Range LBS	Thrust at Flow cutoff LBS	Thrust at TST LBS
CC-735	8,883 - 12600	833	9359
CC-730	3351 - 12600	3427	5775
CC716B	3351 - 12600	3560	4236
RHR-744B	6669.6 - 126000	4461	9693
RHR-744A	6669.6 - 126000	6020	23,422
SI-870B	8099 - 12600	2552	8571
SI-870A	8099 - 12600	5616	9595
SI-864B	12856 - 21600	Static only	19085
CVC-350	1482 - 7200	Static only	2239
FW-V2-6B	38211 - 63000	Static only	44863
FW-V2-6C	38211 - 63000	Static only	38573
FW-V2-6A	38211 - 63000	Static only	43010

No problems were identified with these tests.

The inspectors reviewed the status of feedwater MOV FW-V2-6A to examine the corrective actions taken to improve the operability of the valve. The licensee had made changes to improve the operation of the valve in the open direction. The TOLs had been tripping during the opening stroke of FW-V2-6A. The tripping was determined to be the result of thermal binding of the valve. Thermal binding occurs when the valve is closed while at high temperature and allowed to cool in the closed position which causes the seats to tighten on the wedge gate disc. Then on the next attempt to open the valve after cool down, a high thrust is required to unseat the valve. This high thrust requirement causes the valve actuator motor to stall, causing the TOLs to trip.

On June 15, 1991, an operability review determinated that thermal binding was occurring. In January 1992, the licensee performed calculations Nos. RNP-M/MECH-1398, 1399 and 1400 which recommended a lighter spring pack to stay within the torque rating of the actuators based on the postulated accident differential pressures of 50 psid. The operating procedures were revised to require cycling of the feedwater valves FW-2V-6A, B and C, during unit cool down. Lighter spring packs were installed during the June 1992 outage in the valve actuator of each valve to reduce the thrust at the end of the close cycle. The lighter spring packs were installed as the result of the calculated differential pressure across the valves of 50 psid. This value was based on the postulated assumption that feedwater regulating valves located down stream of each block valve will close in seven seconds after a safety injection (SI) signal is received and the reactor feedwater pump trips and coasts to a stop. The design basis differential pressure report DP-027FW for the motor operated valves (MOVs) in the feedwater system for the Robinson Nuclear Plant acknowledged that if the feedwater regulating valves were in the manual mode at the time the SI signal was received, the block valve would see some, "substantial though indeterminate ΔP." This report assumed that the flow control valve would close and cause minimum flow across the block valve, but in any case did mention that the line pressure is assumed to be 580 psig during accident conditions with the feedwater regulating valves in the manual position. The assumption held to in the evaluation is that the feedwater regulating valves will always be closed first.

On June 15, 1992, Feedwater Regulating Valve FCV-478 was given a close command but did not close sufficiently to reduce the differential pressure across Block Valve FW-2V-6A. FW-2V-6A torqued out before completely closing off the flow to Steam Generator A. Work request WR/JO 92-AJEH2 was written to check the stroke and adjust the positioner of FCV-478. The inspector inquired about the condition of FW-V2-6A and why it did not close fully. The reason given was the discharge pressure of the condensate pump was greater than the 50 psid. The inspector then questioned the licensee concerning the basis for the assumption that the reactor feedwater pump tripping would cause the DP across the block valve to be less than 50 psid. It appears that an unverified assumption was made by the licensee that the condensate pump is also tripped when a safety injection signal is received. The condensate pump trip is a manual action taken by the operator, and is not initiated by an automatic trip signal.

The H. B. Robinson FSAR Table 15.1.5.2, ACTUATION SIGNALS AND DELAYS FOR MSIV, SIS AND FEEDWATER SAFETY ACTIONS, states that the main feedwater regulating valve closure occurs seven seconds after the SI signal. When the licensee recalculated the differential across the valve with the condensate pump still operating the DP was calculated to be 480 psid at closing and 375 psid at opening. The licensee immediately issued the necessary work requests to reset the torque switches on these three valves to enable the actuator to develop the required thrust without tripping the torque switch before closure is accomplished.

The failure on the part of the licensee to consider the DP across the three feedwater valves with the condensate pumps operating, and setting the feedwater block valves to close at a pressure less than actual is identified as violation 50-261/92-19-01: Inadequate design control involving unverified assumptions related to D/P for Valves FW-2V-6A,B, and C.

10 CFR50, Appendix B, Criterion III states in part "... design control measures shall provide for verifying or checking the adequacy of designs ...". Contrary to the above, on June 15, 1992, Feedwater Block Valve FW-2V-6A did not fully close due to the differential pressure across the valve having been calculated at a lower value than existed in the system. The differential pressure had been calcualted to be 50 psid across each of the three feedwater block valves and the valves had been adjusted for closure at that pressure. Upon inquiry by the NRC Inspectors, the licensee recalculated the differential pressure to be 480 psid. The differential pressure was the result of

the condensate pump operating, which was assumed to trip upon receipt of a safety injection signal allowing the valves to close under the lower diffferential pressure.

b. Schedule

The licensee was returning the unit to operation after the completion of a refueling outage. The MOV testing scheduled during the outage was completed as planned. The licensee differential pressure tested 23 MOVs and static tested 42 MOVs. Other planned maintenance items such as the installation of VOTES sensors, electrical and mechanical preventive maintenance, and the replacement of torque switches was accomplished during the outage as scheduled, with the exception of 4 torque switch replacements which were delayed due to parts availability.

c. Maintenance

The licensee has developed procedure TMM-032, TECHNICAL SUPPORT MANAGEMENT MANUAL PROCEDURE; MOTOR OPERATED VALVE PROGRAM, for the purpose of establishing, implementing, and maintaining an overall program for motor operated valves. This procedure references various maintenance documents as guidelines for maintaining MOVS. The inspector reviewed a draft revision of MMM-003, Appendix A, POST MAINTENANCE TESTING, which defines the post maintenance testing required at the completion of various MOV maintenance activities. The control of switch settings is maintained under procedure CM-111, LIMITORQUE LIMIT SWITCH AND TORQUE SWITCH MAINTENANCE. This appears to be an adequate procedural control for accomplishing the MOV GL 89-10 program.

d. Training

The licensee has a group of craftsmen that travel between the nuclear plants and perform the testing of MOVs during outages. The test data taken are initially reviewed by the site personnel and later by the NED. Licensee representatives informed the inspector that some of the site instrumentation and control (I&C) personnel had been trained in the use of the diagnostic equipment, but due to the heavy work load during the recent outage, they were not involved in the testing completed this outage. The licensee has scheduled classes for training in the use and analyzing of the Votes equipment and traces for selected site personnel. The inspector noted that a MOV training for selected site personnel. The inspector noted that a MOV training class was in session during this inspection.

5. Exit Interview

The inspection scope and results were summarized on June 19, 1992, with those persons indicated in paragraph 1. The inspectors described the areas inspected and discussed in detail the inspection results listed below. Proprietary information is not contained in this report. The violation 50-261/92-19-01, Inadequate design control involving unverified assumptions related to D/P for Valves FW-2V-6A, B, and C, was discussed and no dissenting comments were received.

Acronyms and Initialisms

ACR	Adverse Condition Report
CAP	Corrective Action Program
DP	Differential Pressure
EDG	Emergency Diesel Generator
GL	Generic Letter
HPIP	Human Performance Investigation Process
I&C	Instrumentation and Control
ILRT	Integrated Leak Rate Test
LER	Licensee Event Report
MOV	Motor Operated Valve
NOV	Notice of Violation
NRC	Nuclear Regulatory Commission
OTDT	Over-temperature delta-temperature
PPM	Parts Per Million
PNSC	Plant Nuclear Safety Committee
psid	Pounds Per Square Inch Differential
RFO	Refueling Outage
RTD	Resistance Temperature Device
SI	Safety Injection
SCR	Significant Condition Report
TS	Technical Specification
TST	Torque Switch Trip
TOL	Thermal Overload Limit
VOTES	Valve Operation Test & Evaluation System
WR/JO	Work Request/Job Order