



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

Report No.: 50-261/85-11

Licensee: Carolina Power and Light Company  
 411 Fayetteville Street  
 Raleigh, NC 27602

Docket No.: 50-261

License No.: DPR-23

Facility Name: H. B. Robinson

Inspection Conducted: February 11 - March 10, 1985

Inspectors:	<u><i>H. E. P. Krug</i></u>	<u>4/9/85</u>
	H. E. P. Krug, Senior Resident Inspector	Date Signed
	<u><i>P. K. Harden</i></u>	<u>4/9/85</u>
	H. C. Whitcomb, III, Resident Inspector	Date Signed
Approved by:	<u><i>P. E. Fredrickson</i></u>	<u>4/9/85</u>
	P. E. Fredrickson, Section Chief Division of Reactor Projects	Date Signed

SUMMARY

Scope: This routine, announced inspection involved 269 resident inspector-hours on site in the areas of technical specification compliance, plant tour, operations performance, reportable occurrences, housekeeping, site security, surveillance activities, maintenance activities, design change activities, quality assurance practices, radiation control activities, outstanding items review, IE Bulletin and IE Notice followup, organization and administration, independent inspection and enforcement action followup.

Results: One violation was identified in the six major areas inspected (Violation 50-261/85-11-01, Failure to Perform Modifications in Accordance With Regulatory Requirements).

## REPORT DETAILS

### 1. Licensee Employees Contacted

- \*P. Beane, QA Supervisor
- G. Beatty, Manager, Robinson Nuclear Project Department
- C. Crawford, Manager, Maintenance
- \*J. Curley, Manager, Technical Support
- B. Flanagan, Engineering Supervisor - Nuclear
- J. Jefferies, Manager, Corporate Nuclear Safety
- F. Lowery, Manager, Operations
- \*A. McCauley, Principal Engineer, Onsite Nuclear Safety
- R. Morgan, Plant General Manager
- \*M. Page, Plant Engineering Supervisor
- B. Reick, Manager, Control and Administration
- \*D. Stadler, Director, Regulatory Compliance
- \*J. Sturdavant, Technician, Regulatory Compliance
- A. Wallace, Director, Onsite Nuclear Safety
- \*C. Wright, Senior Specialist, Regulatory Compliance
- H. Young, Director, QA/QC

Other licensee employees contacted included technicians, operators, security force members and office personnel.

\*Attended exit interview

### 2. Exit Interview

The inspection scope and findings were summarized on March 8, 1985, with those persons indicated in paragraph 1 above. The licensee acknowledged the inspection findings. No written material was provided to the licensee by the resident inspectors during this report period. The licensee also confirmed that the information provided the inspectors did not include proprietary information.

### 3. Licensee Action on Previous Enforcement Matters

This subject was not addressed in the inspection.

### 4. Plant Tour (71707, 62703, 71710)

The inspectors conducted plant tours periodically during the inspection interval to verify that monitoring equipment was recording as required, equipment was properly tagged, operations personnel were aware of plant conditions, and plant housekeeping efforts were adequate. The inspectors determined that appropriate radiation controls were properly established, excess equipment or material was stored properly, and combustible material was disposed of expeditiously. During tours the inspectors looked for the

existence of unusual fluid leaks, piping vibrations, pipe hanger and seismic restraint abnormal settings, various valve and breaker positions, equipment clearance tags and component status, adequacy of fire fighting equipment, and instrument calibration dates. Some tours were conducted on backshifts.

The inspectors performed valve lineup verifications and system status checks on the following systems:

- a. Component Cooling Water System
- b. Service Water System
- c. Main Steam Isolation Valves
- d. Emergency Electrical Busses

Within the areas inspected, no violations or deviations were identified.

5. Technical Specification Compliance (71707, 62703, 61726)

During this reporting interval, the inspectors verified compliance with selected limiting conditions for operation (LCOs) and reviewed results of selected surveillance tests. These verifications were accomplished by direct observation of monitoring instrumentation, valve positions, switch positions, and review of completed logs and records.

Within the areas inspected, no violations or deviations were identified.

6. Plant Operations Review (71707, 62703, 92700)

Periodically during the inspection interval, the inspectors reviewed shift logs and operations records, including data sheets, instrument traces, and records of equipment malfunctions. This review included control room logs, maintenance work requests, auxiliary logs, operating orders, standing orders, jumper logs, and equipment tagout records. The inspectors routinely observed operator alertness and demeanor during plant tours. The inspectors conducted random off-hours inspections during the reporting interval to assure that operations and security remained at an acceptable level.

On February 3, 1985, an event occurred in which an electrical short, resulting from water being sprayed on the R-21 Radiation Monitoring Vacuum Pump motor cabinet, caused the loss of the safety-related ESF equipment receiving electrical power from Motor Control Center (MCC) 5 (one of the two 480 VAC Motor Control Centers). Details of the event are described in paragraph 6 of Inspection Report 50-261/85-08 and Licensee Event Report 85-008. The electrical short destroyed the R-21 motor controller and subsequently caused another short circuit on the line side of the R-21 motor feeder breaker located inside the same cubicle in MCC 5. These events caused electrical current to rise to an excessive level resulting in an overcurrent trip of the MCC 5 feeder breaker which isolates MCC 5 from the Emergency Bus E1. The E1 bus receives power from the "A" emergency diesel generator when offsite power is unavailable. MCC 5 was isolated from the E1 bus for approximately 1 hour.

The licensee determined that the R-21 motor feeder breaker instantaneous overcurrent trip device was oversized at 100 amps nominal and therefore unable to (1) adequately protect the R-21 motor controller and, (2) prevent the failure of a single faulted load from tripping the MCC 5 feeder breaker. A review of all loads on both MCC 5 and 6 was conducted and it was found that a total of three safety-related loads had oversized overcurrent trip devices. These loads were identified as the R-21 Radiation Monitoring Vacuum Pump, the R-20 Radiation Monitoring Vacuum Pump and the "A" Emergency Diesel Generator Fuel Oil Transfer Pump. This review also noted that the "B" Emergency Diesel Generator Fuel Oil Transfer Pump was set at 20 amps nominal. Drawing B-190627, Revision 5 identified these loads to be "as-built" to a 100 amp nominal setting. Corrective action consisted of replacing the 100 amp overcurrent trip devices of these three loads with overcurrent trip devices with a 20 amp trip rating, using an engineering evaluation, as described in MOD-001, to perform the change.

10 CFR 50, Appendix B, Criterion III states, in part, "Design changes, including field changes, shall be subject to design control measures commensurate with those applied to the original design ..." 10 CFR 50.59, in part, entitles the holder of a license authorizing operation of a utilization facility to make changes in the facility as described in the safety analysis report, without prior Commission approval, unless the proposed change involves an unreviewed safety question. Finally, Technical Specification 6.5.1.2 requires that a safety analysis be prepared for all modifications that affect nuclear safety. The method used to make this modification did not adequately evaluate the trip device modification with respect to a change to the as-built drawing and did not evaluate the change with respect to the involvement of an unreviewed safety question. Instead, the licensee determined that the use of an engineering evaluation was appropriate for this modification.

MOD-001, "Procedure for Preparing Engineering Evaluation", states that "the purpose of the engineering evaluation is to provide a method for documenting technically oriented evaluations and ensuring that these evaluations are reviewed for technical and quality assurance concerns." This procedure does not require that a safety analysis be performed. MOD-001 further states that "Examples of typical items for which engineering evaluations are required are:

- a. Evaluation of items which require a quality classification which exceeds that to which it was procured.
- b. Evaluation of the acceptability of the results of tests where technical problems exist (such as test gauges out of calibration).
- c. Evaluating the interchangeability of safety-related parts having different part numbers/descriptions, etc."

The licensee performed Engineering Evaluation # 85-009 on February 7, 1985, which prescribed the necessary corrective action. The implementation of the corrective action was completed on February 9, 1985, and plant start-up

commenced on February 10, 1985. A revision to Engineering Evaluation #85-009, which included a safety analysis, was completed on February 15, 1985. Thus, the licensee modified three safety-related breaker settings, including one protecting the "A" emergency diesel generator fuel oil transfer pump, without: (1) such changes being subject to design control measures commensurate with the original design; and (2) performing a safety analysis prior to implementing the changes.

This is identified as Violation 50-261/85-11-01, "Failure to perform modifications in accordance with regulatory requirements."

#### 7. Design Changes and Modifications (37700)

Based on the operations problems previously discussed, relating to plant modifications, the inspectors reviewed the measures presently established which specify and provide controls for plant modifications affecting nuclear safety and which specify the performance of safety reviews prior to implementation of the modification.

As was discussed in paragraph 6, 10 CFR 50, Appendix B, Criterion III requires that design changes (i.e., modifications) be subject to measures consistent with those applied to the original design. Also, Technical Specification 6.5.1.2 requires that a safety analysis shall be prepared for all modifications that affect nuclear safety. The analysis shall include a written determination of whether or not the modification is a change in the facility as described in the FSAR, involves a change to the Technical Specifications, or constitutes an unreviewed safety question as defined in 10 CFR 50.59(a)(2). This analysis constitutes a first party safety review and may be accomplished by the individual who prepared the modification. Prior to approval, a second safety review shall be performed on all modifications that affect nuclear safety. This review shall be performed by a qualified individual other than the individual who was the original preparer.

The following established procedures were reviewed during the inspection:

MOD-001, "Procedure for Preparing Engineering Evaluation,"  
Revision 1

MOD-005, "Modification Package Development and Revision,"  
Revision 3

MOD-013, "Safety Review", Revision 1

MMM-013, "Temporary Repairs", Revision 0

The licensee has defined in MMM-013, "Temporary Repairs", a plant modification to be "a planned change in plant design accomplished in accordance with the requirements and limitations of applicable codes, standards, specification, licenses and predetermined safety restrictions." MOD-005, "Modification Package Development and Revision", applies to modification packages for

both new installations and modifications to existing installations. During the process of modification package development, a safety analysis and review shall be performed by qualified personnel in accordance with the guidelines specified in MOD-013, "Safety Review". The requirements of MOD-013 appear to be consistent with the safety analysis requirements of Technical Specification 6.5.1.2. During the performance of the safety analysis, the safety concerns addressed as outlined in 10 CFR 50.59 shall be evaluated for applicability and "A simple statement of conclusion is not sufficient." The licensee emphasized this statement (by underlining it) in MOD-005.

After reviewing two changes recently made to the plant, a consistent mechanism by which plant changes are effected involving safety-related systems does not appear to be present. Corrective action for a MSIV control logic problem was prescribed through the use of a Temporary Repair Procedure (TRP) utilizing the criteria specified by MMM-013. Corrective action for the inappropriately sized overcurrent trip devices found in the three safety-related loads discussed in paragraph 6 was prescribed through the use of an Engineering Evaluation utilizing the guidance specified by MOD-001, "Procedure for Preparing Engineering Evaluation." In both the MSIV logic and safety-related breaker modifications, changes were made to safety-related systems which affect nuclear safety utilizing mechanisms which have different criteria with respect to 10 CFR 50, Appendix B and 10 CFR 50.59 requirements and to the licensee prescribed procedure, MOD-005, which meets these two requirements.

As described in paragraph 6, MOD-001, "Procedure for Preparing Engineering Evaluation", states that the purpose of the Engineering Evaluation is to provide a method for documenting technically oriented evaluations and ensuring that these evaluations are reviewed for technical and quality assurance concerns. MOD-001 does not require that a safety analysis, as required by Technical Specification 6.5.1.2, be performed during the completion of the engineering evaluation.

The purpose of the TRP is "to eliminate personnel hazard, equipment hazard, prevent the spread of contamination, and improve plant operability, etc. A temporary repair also may be used to prevent lost generation." The TRP is presently being utilized to initiate immediate short term corrective action until permanent corrective action (such as a plant modification) can be fully developed. However, MMM-013 states that "a temporary repair procedure must show that the repair will not constitute a technical specification violation, a change or addition to the FSAR, or an unreviewed safety question without the necessary approvals." During the TRP development process, MMM-013 requires that a safety analysis be performed. MMM-013 does not require, however, that the analysis and reviews be performed in accordance with the requirements of MOD-013. Thus, they are not consistent with Technical Specification requirements nor with 10 CFR 50.59. Additionally, no criteria is provided specifying how many reviews must be performed, if any, or when it is necessary that qualified personnel of a particular technical discipline must perform the review. To ensure that all safety

issues are resolved, the TRP safety analysis and review should, as a minimum, require that the safety analysis and reviews be independently performed and recorded by qualified personnel as required by MOD-013.

Although a violation has been identified for the Engineering Evaluation breaker modification, the inspectors determined that in this particular case, the Temporary Repair Procedure change on the MSIV logic did meet the "intent" of Criterion III and Technical Specification 6.5.1.2.

The inspectors concluded that the established program whereby modifications are implemented via the Engineering Evaluation and Temporary Repair Procedure system does not appropriately incorporate administrative features which adequately implement the requirements of 10 CFR 50 Appendix B or 10 CFR 50.59. The inspectors determined that the program failed to provide plant personnel with those guidelines which must be used in the determination of whether a change to a system, which affects nuclear safety, is completed utilizing a Temporary Repair Procedure, Plant Modification or Engineering Evaluation. This is essential, because only plant modifications conducted under MOD-005 receive appropriate measures under Criterion III and receive a safety analysis and review performed to the requirements of Technical Specification 6.5.1.2. The licensee has within its organization, three levels of management review which are responsible for ensuring that modifications and safety reviews are being conducted in accordance with 10 CFR 50 Appendix B and 10 CFR 50.59 requirements. These are identified as the (1) Technical Support organization, (2) Plant Nuclear Safety Committee, and (3) Corporate Nuclear Safety organization which performs part of its function by utilizing Onsite Nuclear Safety personnel at individual plants. All of these groups have shared responsibilities in that they are required to perform an overview of plant modifications and associated to 10 CFR 50.59 reviews to assure that these processes meet the requirements of the regulations. Discussions with the licensee about the issues described above revealed that the licensee had previously recognized certain weaknesses in the program and that improvements are currently being developed to correct these programmatic deficiencies. As these weaknesses are directly linked to the violation identified in paragraph 6, the licensee will be requested to address programmatic changes in addition to specifics with respect to the identified violation.

During this review, the inspectors also noted that Technical Specification 6.5.1.2 does not accurately characterize the requirements of 10 CFR 50.59 in that Technical Specification 6.5.1.2.1 states, in part, that "The analysis shall include a written determination of whether or not the modification is a change in the facility as described in the FSAR, involves a change to the Technical Specifications, or constitutes an unreviewed safety question as defined in 10 CFR 50.59(a)(2)." 10 CFR 50.59 uses the term "involves" instead of "constitutes" with respect to the unreviewed safety question. Where the licensee can meet both a Technical Specification and a 10 CFR requirement, both are required to be met. In this case, although there is a wording difference, both can and shall be required to be met.

8. Physical Protection (71707)

The inspectors verified by observation and interview during the reporting interval that measures taken to assure the physical protection of the facility met current requirements. Areas inspected included the organization of the security force, the establishment and maintenance of gates, doors and isolation zones in the proper condition, that access control and badging was proper, that search practices were appropriate, and that escorting and communications procedures were followed.

Within the areas inspected, no violations or deviations were identified.

9. Licensee Action On Previously Identified Inspection Findings (92701)

(Closed) Inspector Followup Item 50-261/85-08-02; Motor Control Center 5 Isolation. The inspectors reviewed licensee actions with respect to the event which caused isolation of MCC 5. As previously described in paragraph 6 of this report, the inspectors identified a violation of regulatory requirements which pertain to this event. Specific inspector findings will be addressed in the licensee responses to the violation. Therefore, this inspector followup item is considered closed.