UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
101 MARIETTA STREET, N.W., SUITE 2900
ATLANTA, GEORGIA 30323-0199

Report No.: 50-261/95-07

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

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

Facility Name: H.B. Robinson

Inspection Conducted: February 27 - March 3, 1995

Inspector: H. Whitener 4-9-95
H. Whitener Date Signed

Accompanying Personnel: M. Miller, Inspector
O. Mazzoni, Contractor

Approved by: C. A. Casto 4/9/95
C. A. Casto, Chief Date Signed
Engineering Branch
Division of Reactor Safety

SUMMARY

Scope:

This routine, announced inspection was conducted in the areas of design changes and plant modifications, engineering and technical support and follow up on previous inspection findings (IP 37550).

Results:

In the areas inspected, one Violation and one Unresolved Item was identified.

Violation 50-261/95-07-01: Failure to consider the impact of a plant configuration change on Technical Specification 4.4.1.2 leakage limit.

Unresolved Item 50-261/95-07-02: Inadequate evaluation for removal of safety pump room coolers from service: Review the re-analysis.

Enclosure

The reorganization of engineering services at the H. B. Robinson Plant has improved the quality of engineering services and products as evidenced by the improved quality of recent design change packages. (Paragraph 3)

The design change process was, in general, acceptable and the design packages were technically correct and well prepared. (Paragraph 3)

The administrative, prioritization, and design review processes have been maintained and provide sufficient control of plant modifications and design changes to assure plant reliability. (Paragraph 2.)

Engineering has played an active roll in day-to-day plant support, operability and reliability. (Paragraphs 8, 9)

Management initiatives to provide improved working conditions, increase management knowledge of engineering capibility and establish oversight of system conditions were viewed as positive action toward improving engineering performance. (Paragraphs 6, 7)

Pressure/thermal binding of containment sump valves was identified by the licensee and resolved in 1982 by drilling holes in valve discs. (Paragraph 11)

The licensee had successfully resolved six motor operated valve issues. (Paragraph 12)

REPORT DETAILS

1. Persons Contacted

Licensee Employees

*R. Barnett, Manager, RESS TESTING
*J. Boska, Manager, Instrumentation & Controls Design Engineering
*M. Brown, Manager, Design Engineering, RESS
*G. Castleberry, Manager, Electrical Plant Engineering, NED
*B. Clark, Manager, Maintenance
*T. Fay, Technical Support, RESS
*D. Gudger, Senior Specialist, Regulatory Programs
*C. Hinnant, Vice President, H. B. Robinson
*P. Jenny, Manager, Emergency Preparedness
*K. Jury, Manager, Licensing/Regulatory Programs
*M. Knaszak, Manager, Project Management and Engineering Services
*R. Krich, Manager, Regulatory Affairs
*E. Martin, Manager, Document Services
*S. McCutcheon, MOV Coordinator
*W. McGoun, Senior Engineer
*G. Miller, Manager, Robinson Engineering Support Section
*F. Modlin, Engineer, Electrical Plant Engineering
*P. Musser, Plant Operations Assessment
*B. Nauhria, Design Engineer, Mechanical
*D. Taylor, Controller
*R. Wehage, Manager, NSSS Design Engineering
*R. Williamson, Component Engineer
*L. Woods, Manager, Technical Support
*D. Young, Plant General Manager

Other licensee employees contacted during this inspection included engineers, operators, technicians, and administrative personnel.

NRC Resident Inspector(s)

*W. Orders, Senior Resident Inspector
*C. Ogle, Resident Inspector

*Attended exit interview

Acronyms and initialisms used throughout this report are listed in the last paragraph.

2. Planning and Development of Plant Modifications (37550)

In a previous inspection in September 1994 (NRC Inspection Report No. 50-261/94-25) the inspectors reviewed in detail the design change development process used for identifying, screening, evaluating, and approving plant modifications including the plant modification (PM) prioritization process. The inspectors concluded at that time that the licensee had developed sufficient administrative controls, engineering

controls and process reviews to ensure adequate control over the design change process.

Selected modification and design control packages and other documents were reviewed by the inspectors during this inspection to verify that the adequacy of the modification program was maintained. Documents which were reviewed totally, or in part, or discussed with engineering personnel included, but was not limited to, the following:

- Carolina Power and Light (CP&L) Nuclear Plant Modification Program (NPMP).
- PLP-064, Engineering Service Requests, Revision 3.
- MOD-002, Modification And Design Control Procedure, Design Calculations, Revision 8.
- MOD-018, Modification And Design Control Procedure, Temporary Modifications, Revision 13.
- PLP-026, Corrective Action Management, Revision 21.
- PLP-032, 10 CFR 50.59 Reviews Of Changes, Tests, And Experiments, Revision 5.

Based on discussions with licensee engineering personnel and review of the documentation, the inspectors concluded that the licensee had maintained adequate prioritization and review processes in place for identifying and implementing plant design changes.

3. Review of Modification Packages

The inspectors selected nine modification packages for review. Of the selected packages, four modification packages were performed during the last (15th) Refueling Outage (RO), one between RO 15 and 16 and three were to be performed during RO 16. In all cases, the design phases of the packages had been completed. Execution of the packages was also completed except for some phases of the ones to be installed during the upcoming RO 16.

The packages, discussed below, were evaluated to determine that design input was adequate (appropriate design factors considered), design basis was maintained, impact on plant systems was determined, engineering calculations were accurate and complete, 10 CFR 50.59 safety evaluations were adequately performed and appropriate post modification testing was specified. The inspectors concluded that the design change process was, in general, acceptable and the design packages were technically correct and well prepared.

a. Modification Package #M1165 - Resolve Cables EQ Issue (Submergence)

The modification package was approved 1/13/95, and was to be fully implemented during RO 16. It required the replacement of cables unqualified for submersion, which could occur during a LOCA, in compliance with 10CFR 50.49. The inspectors raised several questions, as follows:

1) Cable installation procedures required that the cable be spliced and then pulled into conduit. This installation step required that the splices in the tray be suitable for the pulling forces. However, the inspectors found no objective evidence that the splices had been qualified for the stresses induced in pulling the cable into conduits. In response to the inspectors' question the licensee responded that the installation instructions were not correct. However, instead of correcting the installation instructions, the licensee decided to add a note to the general instructions and notes (CP&L HBR2-0B060) indicating that cables will be pulled first and spliced second, thereby imposing no pulling forces on the splice, since the splice was not qualified to take the pulling forces. The inspectors were of the opinion that the correction of the actual installation step was more pertinent and more important than the correction of the general drawing. At the exit interview the licensee stated that both the general drawing and the installation instructions would be corrected.

2) There was no objective evidence that the splices in trays had been seismically qualified. In response to the inspectors' question, the licensee prepared ESR 95-00221 which concluded that, ..."Since the supports are flexible, then differential movement between cable support systems within the same building would be insignificant due to their low resonance...." The inspectors still had concerns with the possibility of induced loosening of connections at the splice crimping, as could be induced by cable "galloping" from the seismic event. In further discussions with CP&L, the licensee indicated that cable "galloping" would be minimal based on the low probability of induced significant movement and also on the fact that the cable tie wraps are anticipated to remain unaffected by the seismic event.

The inspectors found the licensee's explanation acceptable.

3) The inspectors questioned whether the existing cables could be removed from conduits, and if not, how would the conduit be made ready for installation of new cables. The licensee indicated that pulls were short and straight and there should be no problems. If problems come up there will be field revisions written to handle special situations.

The inspectors considered the licensee's response satisfactory.

b. Modification Package #M1166 - Replace Inverters For Instrument Bus.

The modification was approved 12/9/94, and was to be fully implemented during RO 16. The new inverters had the same output but operated at higher efficiency. Also, they allowed for operation with 100-140 VDC input, which provided greater margin to accommodate the end of life voltage level of the DC battery. The inspectors reviewed the specifications of the new inverters and inspected the prototype inverter which had been delivered to the site. The inspectors questioned the advisability of testing the

inverters with the battery charger and without a battery connected in parallel. As the battery acts as a filter for the charger output, the question was based on the possibility of subjecting the inverters to high output ripple with the possibility of inducing harm to the inverter components. The licensee investigated the inspectors' concern by consulting the charger manufacturer, who indicated that the ripple content was only 2% or less, which was low enough to prevent any concern with damage mechanisms affecting the new inverters under test.

c. Modification #M1125 - Reactor Protection Instrumentation Power Supply Change.

The modification was approved 8/11/93 and executed during RO 15. The inspectors found it adequately documented and executed.

d. Modification #-M1133 - Replace 230 kV Generator Breakers.

The modification was approved 8/11/93 and executed during RO 15. The modification required that the Station Auxiliary Transformer be taken out of service in order to relocate the associated wiring. In the process, the plant had to be backfed through the main step up transformers. Under these backfeeding conditions, the inspectors noted that the Isolated Phase Bus becomes an ungrounded system without any ground detection capability. The licensee indicated that this condition was recognized and would only occur for a limited time when the plant was in cold shutdown. The inspectors were able to verify that TS paragraph 3.7.3 and OP-603, "Electrical Distribution," paragraphs 4.21 and 8.9.1, provided sufficient guidance to alert the operators of degraded protection status and required meggering of the generator Isolated Phase Bus Duct.

The breaker replacement was necessitated by grid stability conditions. The previous breakers were of the three pole, three cycle type which presented stability problems. The new breakers were 2 cycle and single pole operation, which allowed for a lower critical clearing time. The Inspectors reviewed the System Planning & Operations Department System Stability Study, which indicated that the Robinson plant would remain stable under the assumed upset system conditions.

e. Modification #M1074 - Replace Containment Penetrations

The replacement of penetrations was required in order to comply with cable environmental qualification. The program of penetration replacement spanned several outages. This modification included Phase II of implementation, which replaced 5 instrument/control penetrations. Phase I modification package, MOD 1016, was implemented during RO 14. The modification reviewed by the inspectors was approved 1/24/92, and was to be implemented during RO 16. New penetrations were required because the pigtails on existing Crouse-Hinds penetrations had a qualified life up to only 1997 (CP&L PIR PCN 87-085/00). Testing to be performed by IEEE 317-1983, required that penetrations be leak tested at design

pressure and ambient temperature. Leak rate tests could be performed as part of the integrated leak rate testing. The inspectors reviewed the process of transferring of cables from the existing penetrations and found it acceptable.

f. Modification #M1153 - Removal Of Power For The Auto Transformer #1 Control Panel From MCC-3 And MCC-4

The modification package was prepared to allow for another source of power to the Auto Transformer #1 Control Panel, to enable the Transmission Department to perform maintenance activities without interfacing with the Nuclear Engineering Department. The inspectors found the modification well prepared and executed.

g. Modification #M1149 - Turbine Redundant Overspeed Trip System (TROTS) Removal

The TROTS function is to initiate a turbine/reactor trip upon turbine overspeed in excess of 111%. TROTS was completely disabled by a previous modification which disconnected the power supply but left the solenoids in place. The disabling of the system was based on the premise that the Primary Turbine Overspeed Trips were not affected. The removal of the solenoids was found required because they experienced leaking. The Inspectors found the modification well executed.

h. Modification #M1151 - Activation Emergency Bus E2 Undervoltage Trip Function For "C" CCW Pump Supply Breaker

The modification was approved on 10/26/93 and accomplished the scope of temporary modification 92-704, which was installed during Refueling outage #14. The tripping of load breakers in case of Loss of Offsite Power (LOOP), except for the "C" CCW pump breaker, was accomplished by a 1 out of 2 logic. The purpose of the modification was to bring the "C" CCW breaker undervoltage trip in line with the other load breakers by installing a redundant undervoltage trip function for the "C" CCW pump. The redundant trip accomplished a greater assurance of the trip function. The Inspectors found the modification package well prepared and executed.

4. Safety Pump Room Coolers

Background

Due to the deteriorating condition and increased leakage of the AFW room fan coil units HVH-7A and HVH-7B, these were taken out of service during the current operating cycle, pending installation of new tube bundles. A previously performed evaluation, EE 89-018 was utilized to justify taking the units out of service.

Two separate evaluations were performed by the licensee in 1988 and in 1991, to determine compliance with the requirements of 10CFR50.49(d) for safety pump areas which will experience high ambient temperatures and radiation under accident conditions.

At the Robinson plant the SI and CS pumps are in the same pump room, while the RHR and the AFW pumps are in their own separate rooms. Post accident high temperature and radiation ambient conditions would be expected in the SI/CS, and the RHR pump rooms. Post accident high temperature ambient conditions would be expected in the AFW pump room.

The licensee performed evaluation ENG 88-080, dated 6/30/88, initiated by NCR 88-090, to analyze the qualification of the Safety Injection (SI) Pumps, the Residual Heat Removal (RHR) Pumps, and Containment Spray (CS) Pumps. The evaluation for the Auxiliary Feed Water (AFW) pumps was performed by EE 89-018, dated 2-05-91.

In addition, while the inspectors were at HBR, the licensee initiated a reevaluation of the earlier calculations to address the failure of EE 88-080 to, "Adequately consider the effect of post accident temperatures in the rooms enclosing the SI/CS and RHR pumps, on the pumps' lubricants when fan-coil units HVH-6A/6B and HVH-8A/8B are inoperable...."

Evaluation Of SI and CS Pumping Systems

As part of the Environmental Qualification (EQ) program at HBR, ENG 88-079 evaluated the effects of the postulated post accident radiation environment on the operation of the SI and the CS pump motors. As found by the licensee when performing ENG 88-080, ENG 88-079, ..."failed to consider the potential failure of the room fan-coolers and did not fully document the performance of PVC insulated cables exposed to radiation and elevated temperatures...."

Failure Modes and Effects Analyses (FMEA) were conducted in ENG 88-080 to consider the effect of the loss of the fan-cooler units HVH-6A/6B in the SI/CS room and HVH-8A/8B in the RHR room.

The failure mechanisms for the loss of the fan-cooler units were identified as: a) loss of the drive belt due to high radiation, b) loss of the motor, c) failure of the cables. The position taken regarding the FMEA was that the fan-coil units could be taken out of service without jeopardizing plant safety.

In the current reevaluation the licensee addressed the failure of EE 88-080 to, "adequately consider the effect of post accident temperatures in the rooms enclosing the SI/CS and RHR pumps, on the pumps' lubricants when fan-coil units HVH-6A/6B and HVH-8A/8B are inoperable...." With the intent of evaluating these issues, the current evaluation reviewed the qualification of the motors and the cables feeding the pump motors.

No information was provided on the periods of time that the fan-coil units were valved out of service.

Evaluation Of AFW Pumping System

Engineering Evaluation (EE) 89-018 provided justification for continued plant operation with the AFW motor-driven pump room fan-coil units HVH-7A/7B out of service. The evaluation was prepared to resolve a potential non compliance with the single failure criterion relative to

the fan-coil units control circuitry. This evaluation was later found by the licensee (see CR# 95-0031) to be defective in that it relied on temperature values calculated with an unverified calculation, that it did not provide adequate documentation to support the ability of the lubricating oil to sustain high temperatures and that it did not provide sufficient basis to support the qualification of the cables.

The failure mechanisms for the loss of the fan-cooler units were identified as: a) loss of the drive belt due to high radiation, b) loss of the motor, c) failure of the cables. The position taken regarding the FMEA was that the fan-coil units could be taken out of service without jeopardizing plant safety.

CR# 95-0031, provided a summary table of the outages taken for the fan-coil units for the period 1989 to 1994.

Findings Relative To Licensee Past And Current Evaluations

The inspectors reviewed the 1988 and 1991 evaluations as well as the current reevaluation of the earlier analyses.

The Inspectors found that the current evaluation of the lubricating oil properly demonstrated the oil to be suitable for the expected maximum ambient temperatures.

The inspectors could not agree with the licensee's analyses relative to the qualification of the electrical insulation systems for the motors and their associated cables, on the basis of nonconservative assumptions and omissions in performing the evaluations. The inspectors were concerned in that there was only one failure mode considered: the gradual aging of the insulation. At elevated temperatures, the aging effect is accelerated in accordance with a logarithmic expression, which depends upon the type of insulation material. While gradual aging is an appropriate failure mode for end of life estimation on exposure to rated temperatures, or to relatively small increments over rated temperatures, the possibility of "catastrophic failure" was not considered as a failure mode when the ambient temperatures are much higher than rated. Since catastrophic failure could occur prior to the estimated end of life due to gradual aging, the inspectors considered the failure to analyze this abrupt failure mode to be in the non-conservative direction.

The inspectors had additional concerns with the licensee's approach in their review and acceptance of the tests performed on cables. The inspectors found that the samples of cables subjected to testing were not of the same construction as the cables used to feed the pump motors. While the cables were of the same insulation material, the thickness of the insulation, and the conductor size, were totally different. Since the motor feeder cables are of much greater thickness (94 mils) than the samples tested, the conclusions drawn from the test could not be readily applied to the actual cables being reviewed. In addition, the samples tested did not have any cable jacket, while the motor feeder cables had a 45 mill asbestos jacket. The absence of a cable jacket and the thinner insulation were considered by the inspectors to be two factors which would produce results in a non-conservative direction in respect

to the issue being evaluated, i.e., the ability of the cable to dissipate heat. Other factors which would have biased the test results in the non-conservative direction were the lack of current and voltage on the tested samples. The continuing voltage stresses upon applied voltage would have tended to produce an insulation failure mechanism that did not get evaluated in the test. Likewise, the passage of current in the cable conductor would have produced heat losses, which would have tended to increase the insulation temperature. The inspectors felt that the absence of these considerations had invalidated the applicability of the test results to the motor feeder cables.

In addition, the inspectors found no objective evidence that all relevant factors had been considered to arrive at the maximum cable insulation temperature (the maximum insulation temperature that must be used to establish insulation withstand is the maximum conductor temperature). The inspectors found that the cable heat dissipation was not correctly considered by the licensee's past or current evaluations. In the evaluations performed, rather than including the cable temperature rise when evaluating the maximum insulation temperature, the licensee considered only the effect of the ambient temperature alone, which resulted in lower than actual temperatures, i.e., non-conservative results were obtained. The inspectors estimated that due consideration of the temperature rise would have given a considerably higher conductor temperature than that established by the licensee. This was based on the fact that the maximum conductor temperature is to be found as the sum of the maximum ambient temperature plus the temperature rise above ambient, due to load current. The maximum conductor temperature was, as a result, dependent on the motor load current, the ambient temperature, and the heat dissipation through the cable insulation and jacket.

Newly calculated ambient temperature excursions were shown to the inspectors. These ambient temperature calculations appeared to have been more carefully performed than the previous 1988 estimates. However, the inspectors did not see a final approved and verified calculation. Aside from the concerns noted, none of the related conclusions could be validated because of the lack of duly approved temperature calculations as back up data. The licensee stated that the new calculation will be verified via CR 95-00389, C/A #1.

No information was available to the inspectors on the periods of time that the fan-coil units for the SI/CS and RHR rooms were valved out of service. However, some information was available for the periods of time that the AFW room fan-coil units were valved out of service which showed that the two redundant fan-coil units for the AFW room were simultaneously out of service in June 1992, and in May 1989.

The licensee has initiated a more rigorous analysis to determine if the safety pump room coolers are required to be operational to ensure the function of safety related equipment in these rooms post-accident. When the licensee identified the discrepancies in the original analysis all pump room coolers were placed in service. The coolers will remain in service until a re-analysis is performed to determine if the room coolers are required for operability of the safety related equipment under post-accident conditions.

At the exit interview, the inspectors identified this issue as an unresolved item pending the finding of the re-analysis on the need for room coolers to be operable to ensure the operability of safety related equipment in these rooms post-accident.

Unresolved Item No. 50-261/95-07-02: Review the licensee's re-analysis to determine if room coolers are required to be operable to ensure the function of safety related equipment in the safety pump rooms under post-accident conditions.

5. Temporary Modifications

In a previous inspection (NRC Inspection Report No. 50-261/94-25) the inspectors had reviewed in detail the licensee's temporary modification (TM) process to determine its adequacy for controlling and tracking temporary changes to the plant's configuration.

Although no attempt was made to reexamine this area extensively during this inspection, the inspectors performed a detailed walkthrough of a TM package with the System Engineer, reviewed the revised procedures MOD-018 and PLP-064, reviewed the status log for TMs and verified that periodic review of TMs were performed quarterly in accordance with procedure OST-913, Local Clearance And Test Request, Caution Tag, And Temporary Modification Log Audit.

Procedure MOD-018, Temporary Modifications, Revision 13 in conjunction with Procedure PLP-064, Engineering Service Requests, Revision 3 provided the requirements and controls for the initiation, approval, evaluation, installation, removal, and tracking of temporary plant changes. The ESR process initiated the development of the design and implementation packages which contained the essential reviews, evaluations and calculations. These packages are accompanied by a Modification Traveler which identified the required reviews and approvals to be obtained.

Based on this review the inspectors determined that:

- The design analysis and implementation packages for temporary modification ESR 95-00125 were thorough and comprehensive.

- The revised procedures, MOD-018 and PLP-064, provide adequate instructions.

- Installed temporary modifications are inspected periodically by Operations.

- The current number of installed temporary modifications was twenty. The majority of these will be restored at RO 16 which begins in April 95.

6. Plant Walkdown, Plant Conditions

A cable tray discontinuity was observed during the walkdown of the Reactor Auxiliary Building El. 226', cable trays PR100-SA and CR100-SA. The Inspectors questioned if the trays were structurally adequate and whether the continuity of the grounding would require that grounding straps be provided. The licensee determined that the trays were structurally adequate but agreed that a ground strap was required in accordance with CP&L HBR2-08060, Electrical Installation Practices, Notes and Details. Work request, WR/JO 95-ADLT1, was initiated to have the ground straps installed.

During the walkthrough of the plant the inspectors noticed that major improvements had been made in the plant appearance including overall cleanliness, painted surfaces and color coding of equipment. Also, the engineering manager explained the plans to perform a major enlargement of the control room. These management initiatives to provide cleaner, more attractive and better adapted work stations were viewed as positive action toward improving job satisfaction, personnel morale and attitudes.

7. Diesel Generator Systems

The Inspector reviewed the Emergency Diesel Generator (EDG) system, performing a field walk down, witnessing of a management assessment of the System Engineer and review of the Reliability Program in TMM-034, Technical Management Manual, Emergency Diesel Generator Reliability Program, Revision 0, dated 2/5/93. The inspectors found the cognizant engineer to be well versed and capable. The reliability program was considered to be adequate.

The inspectors considered the practice of management conducting periodic walkdowns of systems with the System Engineer as a positive indication of management initiatives to improve engineering standards and performance and to maintain management oversight of plant conditions. During these walkdowns the System Engineer is questioned relative to his/her knowledge of system operation, function, problems and condition.

8. Engineering and Technical Support Activities

a. Organization

CP&L has reorganized the engineering functions from a Central Design Centered organization located at the corporate offices to an on-site centered organization located at each of the nuclear plants. With the exception of a small contingent of Chief Engineers at the Corporate Office all Robinson engineering functions are constituted in the Robinson Engineering Support Section (RESS) located on-site. The various engineering groups and their assigned responsibilities are discussed in detail in NRC Inspection Report 50-261/94-25.

On a trial basis RESS has implemented a pilot program in the Electrical Engineering area. System, Design and Component Electrical Engineers have been combined into one integrated unit

Engineering and etc. under various managers. Data from this pilot program will be evaluated and factored into future engineering reorganizations.

A second organization change was to establish a Rapid Response Team. This team consists of Mechanical, Civil and Electrical Senior Design Engineers. The function of this team includes screening ESRs, resolving short term ESRs rapidly, assigning longer term ESRs to the appropriate engineering group, interpreting design requirements for plant organizations such as work planning, maintenance and operations. Where feasible the team provides immediate response to questions which saves time and eliminates unnecessary paper work.

Based on self assessments and NAD assessments in 1994, engineering management had developed a systematic approach to address identified problems which consisted of scheduled action items assigned under the sponsorship of engineering managers for accountability. That program, Engineering Excellence '94, has been extended with the intent of becoming corporate wide in 1996. The program, Engineering Excellence'96, was in developmental stage at the time of this inspection and appeared to address the responsibilities of senior level management.

b. Engineering Support

The inspectors interviewed licensee personnel and reviewed station records to evaluate engineering involvement in support of day-to-day plant operations. The type of records reviewed included, but were not limited to, the following (some documents are listed by character of information rather than title):

- RNP Organizational Charts
- CP&L MEMO, RESS Organizational Announcement, dated January 18, 1995.
- CP&L Memo, Pilot Organization for Electrical Engineering, dated January 18, 1995.
- CP&L Memo, Engineering Rapid Response Team, dated February 1, 1995.
- RESS Monthly Performance Report, January 17, 1995.
- RESS Monthly Performance Report, February 17, 1995.
- Trending Plots on Performance Indicators.
- Top Ten Equipment Issues.
- Engineering Excellence'96(Incomplete)
- ACR 95-00243/CR Evaluation

- ESR 95-00064 (Minor Modification)

- CR 95-00443 Evaluation/OP-925 Procedure Change

- Electrical Penetration-Phase II, Testing Requirements, Plant Modification 1074.

- Engineering Evaluation 94-094, Isolation Of Air To Main Steam Containment Penetrations.

- ESR 95-00236, Corrective action for PPS leakage.

Based on the review of the above documents and discussions with licensee personnel, the inspectors concluded that engineering provided adequate and timely support to maintenance and operations for day-to-day activities and emergent issues. Certain of the above Condition Reports, CR Evaluations and Modifications are discussed in the following paragraph.

9. Engineering Support Reviews

ACR 95-00243

ACR 95-00243 documents leakage from Charging Pump C through a through-the-wall pipe crack in the 3/4 inch suction stabilizer vent line. A work order was written and the weld and adjacent pipe section exhibiting the crack were replaced. Engineering evaluation determined that system pressure and hole size were such that Charging Pumps A and B were not rendered inoperable. Subsequent evaluation of the crack indicated metal fatigue as a result of cyclic stress. Engineering walkdown of Charging Pumps A, B and C verified continued plant operability and identified long term corrective action to support reliability.

CR Evaluation 95-00443

Justification for not applying the criteria for electrical heat tracing issued by Westinghouse to Ebasco in letter CWE-675 to line 16-SI-151R, RWST suction line external to the auxiliary building, could not be found. A document search could not determine if heat tracing this line was an oversight or if an analysis had been performed by Ebasco. Engineering made temperature measurements at freezing conditions which showed the pipe was at about 50 degrees. Also, as an interim action, engineering revised procedure OP-925, Cold Weather Bill, to include monitoring of the pipe when temperature is at or below 32 degrees. If pipe temperature reaches 35 degrees provide additional heating. These actions support plant operational reliability.

Minor Modification ESR 95-00064

The purpose of this modification was to relocate the power feed to the Hagan Rack from circuit No. 4 on Instrument Bus No. 1 to circuit No. 24 on the same Bus. The modification package was well prepared and complete. The package was developed on 1/13/94 and implemented on 1/14/94. This demonstrated rapid engineering response to plant needs.

Plant Modification 1074

This plant modification involved replacing electrical containment penetrations over a period of several outages. Since the containment is not pressurized at each refueling a local leak rate test method had to be developed as a post modification test. The inspector reviewed the post modification test and concluded that an adequate leakage test had been performed. This action supported plant reliability.

10. Penetration Pressurization System

On June 16, 1994 the licensee isolated containment penetrations for Main Steam lines B and C from the Penetration Pressurization System (PPS) because leakage was approaching the allowable leakage (1.57 scfm) for the PPS. The leakage from these penetrations was through the inner bellows into containment. The leakage was measured as 0.18 scfm and the integrity of the outer bellows was verified by local testing. Consequently, containment integrity was maintained.

The Penetration Pressurization System (PPS) is a system which pressurizes certain penetrations between inner and outer boundaries with air at a slightly higher pressure than accident pressure. It is a continuous leakage monitoring system for the penetrations within the system and has a Technical Specification (TS) imposed leak rate of 0.3Lp, i.e., 30 percent of the allowable containment leak rate. The leakage rate from this system is continuously monitored by a flow measuring device on the control board and is alarmed to alert personnel to excessive leakage. The PPS is not a safety system because no credit was taken for this system in the accident analysis.

In Engineering Evaluation (EE) 94-094 the licensee advanced arguments to justify the isolation of certain penetrations from the PPS. The licensee concluded that with one barrier of a two barrier system intact containment integrity is maintained, that no impact on a safety system exists since the PPS is not considered in the accident analysis, that 10 CFR 50, Appendix J allows but does not require continuous monitoring of penetrations and that the function of the double seal is to allow pressurization and consequently monitoring of penetration leakage. The inspector agreed that: (1) containment integrity was maintained but failure of one barrier does result in a degraded condition, (2) PPS is not a safety system however, it does have a TS requirement, (3) Appendix J does allow but does not require continuous monitoring but TS 4.4.1.2 does. The statement that a double seal is only for the purpose of leakage monitoring is incorrect. Double seals on containment penetrations is inherent in containment design. For instance, only two plants have a PPS, however, all containments are designed with the double seal concept as a defence against failure and consequently leakage.

EE 94-094 failed to address the TS Leakage requirement. The TS leakage limit of 1.57 scfm (0.3Lp) applies to the overall PPS. Removal of the penetrations for main steam lines B and C from the system by isolating the air to these penetrations removes the leakage indication from the flow measuring device used to ensure that the TS limit is met. Under these conditions measured leakage could be indicated within the TS limit

when actual leakage is exceeding the limit by as much as the isolated leakage through the steam line bellows (0.18 scfm).

The TS leakage limit is established and maintained in OMM-008, Minimum Equipment List and Shift Relief (MEL). The value of 1.57 scfm (Design Calculation 91-C-0005) was not revised when the plant configuration was changed by isolating the main steam line B and C penetrations from the PPS on June 16 1994. In ESR 95-00236, performed subsequent to the inspection, the licensee provides an analysis and the basis for adjusting the OMM-008 (MEL) leakage limit to meet the requirements of TS 4.4.1.2. The corrected leakage limit in OMM-008 is 1.39 scfm (1.57 - .18 = 1.39 scfm).

Criterion III of 10 CFR 50, Appendix B, states that, "Measures shall be established to assure that applicable regulatory requirements and design basis...are correctly translated into specifications, drawings, procedures and instructions." Criterion III is implemented through the licensee's Quality Assurance Program, FSAR Section 17.2 which embodies the design control requirements of ANSI 45.2.11 - 1974, Quality Assurance Requirements for the Design of Nuclear Power Plants. ANSI 45.2.11 requires that a change in the designed plant configuration consider all aspects of the design including performance requirements and indicating instruments, controls and alarms required for operation, testing and maintenance.

Contrary to the above requirements, when on June 16, 1994, the licensee changed the plant configuration by isolating the Main Steam lines B and C penetration leakage from the PPS based on Engineering Evaluation (EE) 94-094, the evaluation failed to consider the non-conservative impact of the change on instrumentation monitoring TS leakage. This matter was identified as Violation 95-07-01: Failure to translate the impact of a plant configuration change to specify the correct leakage limit required by TS 4.4.1.2.

11. Containment Sump Valves Pressure Locking Event

A preliminary notification of an event, PNO-I-95-004, dated February 17, 1994, concerning pressure locking of the containment sump valves, was issued by the NRC for information only. A preliminary notification constitutes early notice of events of possible safety significance or public interest. The event, not at Robinson, occurred on January 26, 1995, with the reactor shutdown. The site determined that the containment sump recirculation valves could experience pressure locking during a design basis loss of coolant accident (LOCA) and fail in the closed position. These valves must open to provide a water source for the emergency core cooling systems during the recirculation phase of the LOCA, a minimum of 44 minutes into the event. This type of event was applicable to pressurized water reactors.

The inspectors inquired if the Robinson Nuclear Plant had experienced pressure locking with the containment sump valves. The licensee's MOV engineering personnel provided a history of pressure locking. The licensee identified that a similar occurrence with the containment sump MOV SI-861A was identified August 11, 1982, during the performance of the RHR System Component Test. SI-861A would not fully open. It was

found that the valve discs were deformed. This deformation was attributed to severe overpressurization of the disc interspace during RHR heatup. A study was conducted by the engineering group to determine what other valves were susceptible to the pressure locking and thermal binding phenomenon. Modification M677 was written to drill a hole in one of the valve discs on valves SI-861A&B to prevent any further overpressurization. The other two containment sump valves, SI-862A&B were also modified with holes drilled in one of the discs. The inspectors verified that the gate valve drawing indicated the discs had holes drilled for valves SI-861A&B and SI-862A&B. The inspectors concluded the licensee had previously identified and corrected the pressure locking phenomenon with the containment sump valves.

12. Licensee Action on Previous Inspection Findings (92702)

The following inspector followup items (IFI) were identified during GL 89-10 Part 2 MOV inspection conducted March 28 - April 1, 1994, and discussed in NRC Inspection Report 50-261/94-06. The inspectors concluded that the licensee had either taken or was in the process of taking appropriate corrective action that permitted the closeout of the following IFIs.

a. (Closed) IFI 50-261/94-06-01, Margin for Load Sensitive Behavior.

During the review of MOV thrust calculations, it was noted that margin was not included for load sensitive behavior (also known as rate of loading). Licensee personnel stated that their evaluation of test data did not indicate an appropriate amount of margin to be set aside for load sensitive behavior. The licensee stated, "that the addition of the diagnostic and torque switch repeatability accuracies to the calculated minimum thrust requirement increased the conservatism of their minimum required thrust settings and would account for load sensitive behavior." The inspectors concluded the licensee's justification that this margin will be adequate for MOVs that can not be tested at or near design-basis differential pressure would require further evaluation.

The licensee stated that load sensitive behavior was or will be factored back into the thrust calculations to increase minimum required thrust at torque switch trip. In addition, MOVs not tested will be re-evaluated to determine the appropriate amount of load sensitive behavior to be applied. Set up thrust calculations will be revised to incorporate load sensitive behavior to adjust the minimum required torque switch trip setting. The inspectors verified the licensee was in the process of implementing the corrective action and concluded it was appropriate. This IFI was closed.

b. (Closed) IFI 50-261/94-06-02, Correction of Maximum Setting for Torque Switch Repeatability.

During review of MOV thrust calculations, it was noted that the calculated maximum thrust limit was not reduced for torque switch repeatability. However, the limit did include a 10 percent safety

margin. This margin would be adequate to address torque switch repeatability except at a numerical switch setting of 1, where the repeatability error may be as great as 20 percent. (Reference Limitorque Maintenance Update 92-2 for repeatability error). The licensee initiated a procedural change to provide for the repeatability error. This change was not verified by the inspectors at that time.

The inspectors verified that procedure CM-111, "Limitorque Limit Switch & Torque Switch Maintenance," Revision 2 was revised to reflect the appropriate torque switch repeatability error for a setting of 1 (one). The inspectors concluded this corrective action was appropriate. This IFI was closed.

c. (Closed) IFI 50-261/94-06-03, Actions to Ensure Capabilities of RHR-744 Valves.

During review of the thrust calculations for valves RHR-744A&B, the inspectors found the motor torque was marginal and that RHR-744A had recently failed to open during a surveillance test. When the licensee re-evaluated motor torque as the result of limitorque's part 21 (elevated temperature effects reduce torque), it was initially determined the valves did not have adequate torque in the opening direction. The licensee re-calculated the minimum torque under degraded voltage and found the torque marginally adequate. The licensee initiated Plant Improvement Request 94-019/00 that recommended replacing the motor feeder cables with a larger size that would provide the motor with higher voltage and therefore increase the torque. The inspectors considered the implementation of this recommendation an acceptable corrective action.

The inspectors reviewed plant modifications Mod. No. M1167, Replacement of RHR-744A&B Motor Feeder Cables and Mod. No. M1172, MOV Torque Switch Bypass Jumpers. In addition, electrical calculation RNP-E-5.047, Cable Sizing Calculation For The RHR Loop To RCS Cold Leg Valve RHR-744A&B was reviewed to verify that the larger cables would increase motor voltage and torque. The licensee used the degraded grid voltage of 415 VAC for the calculation. Abnormal Operating Procedure AOP-027, Operation With Degraded System Voltage specified a voltage of 430 VAC (This has been changed to 450 VAC per Revision 7 dated December 12, 1994). The inspectors verified that even at 415 VAC the larger cables provide the required voltage and torque. Mod. No. M1172 provides for bypassing the torque switch. The inspectors verified that both modification packages are complete, meet the necessary requirements, will provide the appropriate corrective action, and are scheduled for implementation during the next refueling outage (Spring 1995). In addition, the licensee had submitted a request August 23, 1994, for a license amendment to change the degraded voltage relay setpoint from 415 VAC to 430 VAC. The NRC issuance of this proposed amendment, dated February 23, 1995, was sent to the Federal Register. The inspectors concluded the corrective action was appropriate. This IFI was closed.

d. (Closed) IFI 50-261/94-06-04, Revision of Design Criteria Assumption Based on Test Data.

The inspectors reviewed MOV test procedures, completed test packages and examined diagnostic static and dynamic test traces for the sampled valves. Independent calculations were performed by the inspectors using the licensee's test data. The results of these independent calculations indicated the assumptions used by the licensee for "valve factors" (VF) were not always bounding. The assumed VFs used were considered too low. The licensee agreed a VF greater than 0.40 was appropriate and Design Guide DG I.II would be revised to include the best available data for valve factors.

The inspectors verified that DG I.II, Design Guide For Limitorque MOV Mechanical Evaluation, was revised in Revision 7 to use the appropriate VFs. The changes included the following: 1) New installation - Assume VF of 0.75, 2) Existing MOV that cannot be tested - Assume a VF of at least 0.50, and 3) Westinghouse valves - Assume VF of 0.55. The inspectors concluded the corrective action was appropriate. This IFI was closed.

e. (Closed) IFI 50-261/94-06-06, MOV Post Maintenance Testing.

Appendix A of maintenance procedure MMM-003, Maintenance Work Requests specified the post maintenance test requirements. The inspectors indicated the adequacy of static testing to verify thrust was not fully justified for valve replacement or overhaul.

The inspectors verified that the maintenance procedure MMM-003, Revision 13, dated December 14, 1994, was revised to require testing under both static and dynamic conditions for valve overhaul or replacement. The revisions were in the Motor Operated Valve Test Matrix in Attachment 7.2. of Appendix A. The inspectors concluded the corrective action was appropriate. This IFI was closed.

f. (Closed) IFI 50-261/94-06-08, Setting Closed-to-Open Bypass Switch Limits.

The licensee had issued Plant Improvement Request 93-056100, dated April 30, 1993, which recommended increasing the closed-to-open bypass from 5 percent to 95 percent. Therefore, the torque switch would be bypassed for 95 percent of the valves opening stroke. This change was not fully implemented at that time.

The inspectors reviewed plant modification Mod. No. M1172, MOV Torque Switch Bypass Jumpers to verify that the close-to-open torque switch is to be bypassed. The inspectors verified that M1172 was complete, met the necessary requirements, will provide the necessary corrective action, and is scheduled for implementation during the next refueling outage (Spring 1995). The inspectors conclude the corrective action was appropriate. This IFI was closed.

13. Exit Interview

The inspection scope and results were summarized on March 3, 1995, with those persons indicated in paragraph 1. The inspectors described the areas inspected and discussed in detail the inspection results. Proprietary information is not contained in this report. No dissenting comments were received from the licensee.

Violation 50-261/95-07-01, "Failure to consider the impact of a plant configuration change on Technical Specification 4.4.1.2 leakage limit.

Unresolved Item 50-261/95-07-02, Initial evaluation for removal of safety pump room coolers from service was inadequate. Review the re-analysis

Six MOV inspector followup items were closed.

14. Acronyms and Initialisms

AFW	Auxiliary Feed Water
C/A	Corrective Action
CP&L	Carolina Power and Light
CR	Condition Report
CS	Containment Spray
DBD	Design Basis Document
EDG	Emergency Diesel Generator
EE	Engineering Evaluation
ENG	Engineering
EQ	Environmental Qualification
ESR	Engineering Service Request
FMEA	Failure Modes And Effects Analysis
FSAR	Final Safety Analysis Report
GL	Generic Letter
HBR	H. B. Robinson
IEEE	Institute Of Electrical And Electronics Engineers
IFI	Inspector Followup Item
IP	Inspection Procedure
KV	Kilovolt
LOCA	Loss Of Coolant Accident
LOOP	Loss Of Offsite Power
MCC	Motor Control Center
MOV	Motor Operated Valve
NAD	Nuclear Assessment Department
NCR	Non Conformance Report
NED	Nuclear Engineering Department
NPMP	Nuclear Plant Modification Program
NRC	Nuclear Regulatory Commission
OST	Operations Surveillance Test
PLP	Plant Procedure
PRG	Plant Review Group
QA	Quality Assurance
RESS	Robinson Engineering Support Section
RHR	Residual Heat Removal
RNP	Robinson Nuclear Plant
SI	Safety Injection

TM	Temporary Modification
TS	Technical Specification
VAC	Voltage Alternating Current
VF	Valve Factor