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

Report No.: 50-261/91-20

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: September 7 - October 11, 1991

Lead Inspector: H.O. Ch— for 10/25/91
L. W. Garner, Senior Resident Inspector Date Signed

Other Inspector(s): K. R. Jury, Resident Inspector

Approved by: HO Ch— 10/25/91
H. O. Christensen, Section Chief Date Signed
Division of Reactor Projects

SUMMARY

Scope:

This routine, announced inspection was conducted in the areas of operational safety verification, maintenance observation, onsite review committee activities, engineered safety feature system walkdown, and followup.

Results:

An apparent violation was identified involving inadequate Engineering design control and interfaces (paragraph 5).

Three non-cited violations were identified involving: a security watchperson's failure to perform an adequate personnel search; non-licensed operators performing licensed operator duties; and a failure to provide adequate procedures for performing Technical Specification surveillance testing (paragraph 2).

An unresolved item was identified relating to Loss of Coolant Accident analyses conformance with 10 CFR 50.46 requirements (paragraph 5).

Operator performance (i.e., command and control) was professional during two reactor power reductions (paragraph 2)).

Immediate corrective actions taken in response to the OT Delta T issue were timely and extensive (paragraph 5).

REPORT DETAILS

1. Persons Contacted

*R. Barnett, Manager, Outages and Modifications
*C. Baucom, Senior Specialist, Regulatory Compliance
*F. Bishop, Principal Engineer, Nuclear Assessment Department
**R. Chambers, Plant General Manager
*C. Dietz, Vice President, Robinson Nuclear Project
*D. Dixon, Manager, Control and Administration
*W. Gainey, Manager, Plant Support
*W. Jackson, Engineer, Technical Support
**J. Kloosterman, Manager, Regulatory Compliance
*A. Padgett, Manager, Environmental and Radiation Control
*M. Page, Manager, Technical Support
*R. Parsons, Manager, Robinson Engineering Support
*D. Stadler, Onsite Licensing Engineer, Nuclear Licensing
*R. Steele, Shift Supervisor, Operations
*A. Wallace, Acting Manager, Operations
*L. Williams, Manager, Emergency Preparedness, Security

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

*H. Christensen, Section Chief, Division of Reactor Projects was on site October 9, 10, and 11, 1991, to meet with the resident inspectors and plant management.

*Attended exit interview on October 11, 1991.
**Attended exit interview on October 21, 1991.

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

2. Operational Safety Verification (71707)

The inspectors evaluated licensee activities to confirm that the facility was being operated safely and in conformance with regulatory requirements. These activities were confirmed by direct observation, facility tours, interviews and discussions with licensee personnel and management, verification of safety system status, and review of facility records.

To verify equipment operability and compliance with TS, the inspectors reviewed shift logs, Operation's records, data sheets, instrument traces, and records of equipment malfunctions. Through work observations and discussions with Operations staff members, the inspectors verified the staff was knowledgeable of plant conditions, responded properly to alarms, adhered to procedures and applicable administrative controls, cognizant of in-progress surveillance and maintenance activities, and aware of inoperable equipment status. The inspectors performed channel verifications and reviewed component status and safety-related parameters to verify conformance with TS. Shift changes were observed, verifying that system status continuity was maintained and that proper control room staffing existed. Access to the control room was controlled and operations personnel carried out their assigned duties in an effective manner. Control room demeanor and communications were appropriate.

Plant tours and perimeter walkdowns were conducted to verify equipment operability, assess the general condition of plant equipment, and to verify that radiological controls, fire protection controls, physical protection controls, and equipment tagging procedures were properly implemented.

Non-Licensed Operators Standing Watch

On September 5, 1991, the licensee identified that two individuals who had recently completed SRO training performed the duties of RO license positions (stood watch) prior to receiving their NRC SRO licenses. The docket numbers for these individuals was telephonically transmitted on August 19, 1991. At that time, the licensee erroneously believed that a docket number was sufficient basis to allow these individuals to stand watch (i.e., docket number issuance was equivalent to license issuance). One individual had an inactive RO license and was in the process of license reactivation when he received his docket number. This individual stood watch as RO and BOP operator on seven occasions prior to license issuance. The other individual was an "instant" SRO who was previously licensed at another licensee facility (Shearon Harris). He stood watch on three occasions as RO and once as BOP operator prior to receiving his license.

After recognizing this problem, the licensee immediately relieved the one individual who was on shift. An ACR, 91-317, was generated to document and perform a root cause analysis on this problem. Apparently, the licensee believed the docket number to be sufficient documentation to allow potential ROs or SROs who had taken and passed the NRC license examination to stand watch. Discussions were held between the licensee and the Region II Operations Branch Chief to resolve this issue and to discuss the historical basis for the licensee's position. The licensee was informed that ROs and SROs are not permitted to perform licensed duties before license issuance. 10 CFR 55.3 requires that a person must be authorized by a license to perform the function of an operator or senior operator. 10 CFR 55.53 (e) delineates the requirements to reactivate a license. Failure of these individuals to meet the requirements of 10 CFR 55 is a violation. However, this violation meets the criteria specified in Section V.G.1. of the NRC Enforcement Policy (10 CFR 2, Appendix C) for not issuing a Notice of Violation and is not cited. This violation is identified as an NCV: Non-Licensed Operators Performing Licensed Operator Duties, 91-20-01.

Inadequate Personnel Search

On September 12, 1991, the inspector observed an inadequate personnel search at PAP West. The search was inadequate, in that, a watchperson assigned to the metal detector did not perform a visual inspection of an item which was not processed through the X-ray detector. The inspector questioned the watchperson and determined the watchperson was unaware of what the unsearched item was. The watchperson then stopped the employee prior to him accessing the PA and searched the item. The watchperson determined that the item was permitted inside the PA.

After discussion of this event with security management, the licensee performed a comprehensive investigation, took appropriate disciplinary action, and provided emphasis (through a "Lessons Learned" memorandum) to all security personnel on the necessity to follow procedures.

Security procedures require that all items or equipment be searched by visual inspection or by processing through special purpose detectors. The watchperson's failure to perform the search as described above is a violation. However, this violation meets the criteria specified in Section V.A. of the NRC Enforcement Policy (10 CFR 2, Appendix C) for not issuing a Notice of Violation and is not cited. This violation is identified as an NCV: Failure To Perform An Adequate Personnel Search, 91-20-02.

Emergency TS Amendment Due To Inadequate Surveillance Testing

During internal electrical distribution system reviews on September 13, 1991, the licensee questioned whether or not each channel associated with the loss of power load shed feature had been tested as required by TS Table 3.5-3, item 3.a. The licensee additionally questioned if load shedding upon a simulated loss of all normal AC coincident with a safety injection signal, had been demonstrated as required by TS 4.6.1.2. An operability determination (91-20) was initiated to determine the status of testing and compliance with TS requirements.

On September 14, based upon engineering reviews, the licensee determined that the applicable TS surveillance requirements had not been fully satisfied. These reviews identified instances in which the ability of both channels to initiate load shedding had not been tested. In addition, breakers were identified which had not been demonstrated to open when a load shed signal was initiated. Upon the determination that these surveillance requirements had not been properly implemented, the loss of voltage instrument channels were declared inoperable (TS Table 3.5-3, item 3.a). An 8 hour to hot shutdown LCO was appropriately entered at 4:50 p.m. Based upon the circuit components known to have been tested and the reliability of the components which were either not tested or still under evaluation, the licensee determined that a high confidence level existed that the loss of voltage instrumentation safety function would be accomplished if called upon(i.e., the condition had minor safety significance). However, due to the complexity of developing a test procedure, testing could not be performed prior to LCO expiration. Thus, the licensee requested a Wavier of Compliance from Region II to determine the feasibility of developing a test procedure which could be performed with the plant on line and/or the need for emergency TS relief. Regional management, after consultation with NRR, agreed to a Waiver of Compliance which authorized continued power operation until midnight on September 18. A condition of the waiver was to train Operations personnel on compensatory actions which could be taken if load shedding would fail to occur. The inspectors verified, through attending training sessions and interviews with personnel, that the committed training was provided.

On September 16, Engineering recommended, and plant management concurred, that on-line testing was not desirable. Thus, emphasis was shifted to emergency TS change development to allow continued operation until RO 14 or an outage of sufficient length to allow time for test procedure development and performance.

On September 18, per letter NLS-91-245, the licensee submitted an emergency request for license amendment. Enclosure 5 of this letter identified the circuit components and functions which were not fully tested.

The PNSC meetings which approved the emergency request for license amendment is discussed in paragraph 5. A Waiver of Compliance was granted on September 18, 1991, to allow continued operation pending NRC review and approval of the emergency request. On September 27, 1991, the NRC issued Amendment No. 136 to grant authorization to operate until testing can be performed on, or no later than, RO 14.

The failure to perform surveillance testing as required by TS is a violation. A previous violation (90-11-01) was issued in June 1990 for failure to take adequate corrective action to preclude inadequately established surveillance procedures. In response to this violation, a program was initiated to ensure that established procedures adequately implement instrumentation surveillance test procedures. At the time of discovery, this process had not yet been performed on the TS surveillance requirements in question. The identification of this violation by engineering personnel was considered a strength. No violation is being cited as the criteria of Section V.G.1 of the NRC Enforcement Policy (10 CFR 2, Appendix C) was met. The violation is identified an an NCV: Failure To Provide Adequate Procedures For Performing TS Surveillance Testing, 91-20-03.

B Condensate Pump Motor Ground

On September 19, 1991, at 11:41 p.m., a 4160V switchgear ground alarm was received on the RTGB. Although I & C verified that the alarm input was valid, there were not any other indications that a ground existed. Additional troubleshooting early the following morning, revealed that the 4160V buss 4 ground device had actuated as indicated by closed relay contacts; however, the normal visual indicator, a red flag, did not drop.

Shortly after 9:00 a.m., smoke was smelled in the vicinity of the power cable conduit associated with the B condensate pump motor. At 9:18 a.m., a reduction to 50 percent power was initiated to allow removal of the B condensate pump from service. When the B condensate pump motor breaker was opened, the ground alarm reset. Inspection of the field power cables to the motor stabs revealed damage indicative of a current path in this area (i.e., electrical short). The motor windings were found to be undamaged. The motor was shipped offsite for repairs. Shop inspection revealed that a breakdown had occurred in the B phase stab's insulation which had initiated a current pathway through the fiberglass stab holder block to one of the stab holder block's mounting bolts. The damaged components were replaced or repaired as necessary. The motor was re-installed, successfully tested, and returned to service on September 24, 1991. Power escalation to 100 percent was initiated, but was delayed due to repairs to a leaking relief valve on the 5B FW heater and the 6A FW heater's level indicating column and level control valve. Full power operation resumed on September 26, 1991.

Investigation into the failure of the ground relay flag to actuate disclosed that a design problem exists. The amount of current which activates the flag's positioning relay is determined by the high resistance of the annunciator circuit. Due to this high resistance, the current was not sufficient to activate the relay. The licensee was in the process of evaluating possible design changes to correct the condition, as well as reviewing other circuits to determine if similar conditions exist. This is an IFI: Review Corrective Actions Associated With Failure Of A Ground Relay Flag To Drop, 91-20-04.

Turbine Drain Line Steam Leak

On September 27, 1991, a leak developed in the number 4 governor valve drain line. After consideration of possible on-line repairs, the licensee decided to remove the unit from service to replace the affected line and inspect similar lines from the other governor valves. Since repairs were expected to require less than a day, reactor power was maintained at approximately 0.01 microamperes as indicated on the intermediate range power monitors. The affected line and two others which had unacceptable wall thinning, as determined by ultrasonic inspections, were replaced. The unit was returned to service the next day and resumed 100 percent power operation on September 29. The line failure was attributed to errosion. The licensee is re-evaluating the scope of their errosion/corrosion program.

Both power reductions described above were witnessed by the inspectors. The Operations staff demonstated good command and control during these evolutions, as both power reductions were professionally performed without incident.

Three non-cited violations were identified.

3. Monthly Maintenance Observation (62703)

The inspectors observed safety-related maintenance activities on systems and components to ascertain that these activities were conducted in accordance with TS, approved procedures, and appropriate industry codes and standards. The inspectors determined that these activities did not violate LCOs and that required redundant components were operable. The inspectors verified that required administrative, material, testing, radiological, and fire prevention controls were adhered to. In particular, the inspectors observed/reviewed the following maintenance activities:

CM-507	Emergency Diesel Lube Oil Strainer
PM-007	Emergency Diesel Generator Inspection Number 1
WR/JO 91-ALNA1	Exhaust Turbocharger Oil Leak
WR/JO 91-ALEMI	Air Receiver Relief Valve Replacement
WR/JO 91-FMA392	EDG B Compresser Oil Change

No violations or deviations were identified.

4. Onsite Review Committee (40500)

The inspectors evaluated certain activities of the PNSC to determine whether the onsite review functions were conducted in accordance with TS and other regulatory requirements. In particular, the inspectors attended the September 16 and 17, 1991 PNSC meetings in which the request for a Waiver of Compliance from the requirements of TS Table 3.5.3 and surveillance requirement 4.6.1.2 was reviewed. (See paragraph 2 for details concerning this issue.) During the September 16 meeting, the inspectors observed that the PNSC members mixed their safety review function with their management function. The focus of this meeting shifted from evaluating the safety implications (risk and potential consequences) of the issue to improving the wording of the Waiver of Compliance to maximize the probability it would be approved. This was manifested by the PNSC expending almost all of the two and one half hour PNSC meeting in rewording and rewriting the proposed request for regulatory relief. The PNSC recognized that another review would be necessary and scheduled one for the next day. After the first meeting, the inspectors expressed concerns about the conduct of the meeting with plant management. The inspectors subsequently determined that even though the potential scope of the problem was recognized on September 16, the degree to which various components had been tested, if at all, was not known by the PNSC members. Prior PNSC meetings, both routine and special, have typically been well conducted with the appropriate emphasis on safety. The September 17, 1991 meeting which approved the request for Waiver of Compliance was more typical of the manner in which PNSC meetings have been conducted. On September 27, 1991, the inspectors met with the Plant General Manager and the Regulatory Compliance Manager to discuss concerns with the PNSC's performance. The inspectors expressed concern about the focus of the September 16 PNSC meeting, the potential conflict due to possible ownership by the PNSC for a document which the PNSC had essentially generated, and the lack of a total understanding of what had not been tested (i.e., could not make an adequate evaluation of the issue's safety significance).

The Plant General Manager acknowledged the inspectors concerns. Though not totally agreeing with the inspectors comments, he did indicate that the September 16 PNSC meeting was atypical and had not met his expectations. Both the inspectors and plant management agreed that between the two meetings, the PNSC had adequately discharged its duties during the review of the Waiver of Compliance request. It was determined that provisions of the TS dealing with membership, review process, frequency, and qualifications were satisfied.

No violations or deviations were identified.

5. Followup (92700, 92701, and 92702)

(Closed) URI 91-14-01, Review Impact Of Entrainment Losses On LOCA Analysis, and URI 91-19-01, Determine Safety Significance and Root Causes Of Excessive OT Delta T and OP Delta T RPS Time Delays.

These issues (discussed in IRs 91-14 and 91-19, respectively) involved numerous design/engineering breakdowns. However, at the time the associated IRs were issued, questions involving ECCS performance during a SBLOCA had not been fully resolved. Also, the licensee's review into the significance and root causes of the excessive OT Delta T and OP Delta T RPS time delays had not been completed. The following paragraphs summarize the LOCA issues and related analyses (SI issue), as well as, provide an update to the excessive OT Delta T and OP Delta T RPS time delays .

The SI issue involved the effects of entrainment inventory losses and securing ECCS flow during the transfer from the injection phase to the recirculation phase of a LOCA. The amount of inventory prior to and the rate of inventory loss during the time ECCS is interrupted, determines the PCT during the transfer. The transfer is performed via Emergency Procedure EPP-9, Transfer to Cold Leg Recirculation. EPP-9 is required to be initiated when the RWST level reaches 27 percent and the actual transfer is to be completed prior to RWST level reaching 9 percent. The procedure restricts the time that ECCS flow is interrupted to ten minutes for SI system component re-alignments when RCS system pressure is above the shutoff head of the RHR pumps and to three minutes when the RCS system pressure is below this value. The former case corresponds to small SBLOCAs while the latter case corresponds to the larger SBLOCAs and LBLOCAs.

On June 20, 1988, TS Amendment 119 was issued to support single SI pump operation. The analysis to support the amendment failed to address the consequences of having only one SI pump available during implementation of EPP-9. In January 1989, NFS Design Activity 89-001 was generated to provide justification for a revision to EPP-9 to recognize that only one SI pump may be available during a LOCA . Utilizing a simple decay heat model from a text book, written by El-Wakil, Design Activity 89-001 demonstrated that the flow rate from one SI pump would be slightly greater than the steaming rate due to decay heat. Consequently, the licensee concluded that core cooling is maintained during performance of EPP-9; thus operation with one SI pump was deemed acceptable. However, on May 14, 1991, as a result of IPE activities, the licensee determined that Design Activity 89-001 did not consider the inventory loss due to entrained water in determining the flow out of the break. The previous analyses associated with a minimum operation of two SI pumps during EPP-9, the plant configuration prior to February 1988, had appropriately incorporated entrainment losses. Upon this discovery, reactor power was reduced to 60 percent as previous analysis supported. Following an interim Westinghouse analysis justifying operation at 95 percent power (assuming a 700 degree F maximum PCT after the injection phase and a core heat load based on ANS 1979 decay heat plus 2 sigma), power was increased to 90 percent on May 15, 1991. Later that day, the NRC informed the licensee that use of the ANS 1979 decay heat model was not in compliance with 10 CFR 50 Appendix K requirements. Accordingly, on May 16, the licensee informed the staff that the interim analysis was reperformed using the ANS 1971 decay heat plus 20 percent model and that acceptable results were obtained for power operation up to 92.5 percent.

On May 29, 1991, based on a more rigorous re-analysis which used the ANS-1971 plus 20 percent model, Westinghouse determined that the maximum PCT while ECCS flow is interrupted per EPP-9 during a LBLOCA was approximately 1250 degrees F. The results of this re-analysis were discussed with the NRC, and the unit was subsequently returned to full power later that day. (Based on questions by the NRC regarding mass quantity and distribution in the core, the maximum PCT while ECCS flow is interrupted per EPP-9 during a LBLOCA was subsequently revised to approximately 1400 degrees F.)

Since the licensee and Westinghouse assumed the LBLOCA to be the most limiting case, none of the analyses discussed above addressed the affects of a SBLOCA during the EPP-9 time frame. This bounding of the SBLOCA was based upon the engineering judgement that prior to securing ECCS flow per EPP-9 during a SBLOCA, the inventory in the vessel would be as great or greater than that which would exist for a LBLOCA. Prompted by the NRC, subsequent SBLOCA analysis revealed that during the transfer to recirculation with only one SI pump available, the SBLOCA is not bounded by the LBLOCA. Specifically, this new analysis indicated that the SBLOCA inventory level is significantly less than previously assumed (i.e., core uncovery is greater and exists for a longer period of time than that previously anticipated). Thus, as calculated for the worst case SBLOCA (a one and one-half inch break), the maximum PCT of 1936 degrees F during the time ECCS flow is interrupted exceeds the worst case LBLOCA PCT of approximately 1400 degrees. The SBLOCA PCT during EPP-9 is less than the 2200 degrees F 10 CFR 50.46 ECCS acceptance criteria and less than the values calculated for both the SBLOCA and LBLOCA PCT during the initial injection phase (i. e., 2096 and 2178 degrees F, respectively). When the plant license was amended in June 1988 to allow the configuration of only one SI pump being available to mitigate the consequences of a design basis accident, the licensee failed to analyze the consequences of only one SI pump operation during ECCS switchover. This issue was identified by the NRC during their review of re-analyses associated with the entrainment issue. As determined, the SBLOCA is independent of the entrainment issue due to the higher RCS pressures experienced. Accordingly, URI 91-14-01 regarding entrainment losses is considered closed. However, the acceptability of a second peak (i.e., approximately 1400 and 1936 degrees F for the LBLOCA and SBLOCA, respecitvely) is still under review by the NRC. This is an URI: Determine If The Existance Of Second Peak Is In Accordance with 10 CFR 50.46, 91-20-05.

Summarizing the SI issue, there were significant breakdowns in the technical reviews/analyses performed throughout the chronology. These included: (1) Failure to perform an analysis to support single SI pump operation in June 1988; (2) Inadequate analysis performed in January 1989 (Design Activity 89-001) to support single SI pump operation; (3) Inadequate analysis performed by Westinghouse to permit power operation up to 95 percent power, and this was not identified by the licensee prior to power accession from 60 percent to 90 percent power; and (4) Incomplete SBLOCA analysis performed due to the belief that the LBLOCA analysis was more limiting.

Engineering/design control concerns similar to the SI issue discussed above were also seen in the OT Delta T and OP Delta T excessive time delay problem addressed in URI 91-19-01. The OT Delta T septpoint provides for the on-line protection against DNB. This setpoint is part of the RPS circuitry, and provides a reactor trip when the core Delta T, a measure of reactor power, exceeds the setpoint value. The setpoint is calculated based on inputs of core temperature, pressure, and power distribution. The plant safety analyses assume the OT Delta T trip function is used to mitigate three UFSAR Chapter 15 events.

As discussed in IR 91-19, the problem resulted from capacitors (filters) being installed in the OT Delta T and OP Delta T (which is not used in accident analyses) RPS circuitry. These capacitors imparted an additional approximate two second time delay on RTD system response time. As a result, the OT Delta T protection circuitry response time exceeded that used in the accident analyses approximately 6.75 seconds versus the 4.75 seconds analyzed. Based on a review by the fuel vendor (Siemens) for cycles 13 and 14, the licensee has concluded that sufficient margin was available to compensate for the additional channel response time, such that the OT Delta T trip function would have performed as required to maintain the MDNBR greater than the TS value of 1.17. However, actual RCS flow rates had to be utilized for one of the three Chaper 15 events (Uncontrolled Control Rod Assembly Withdrawal at Full Power) to obtain an MDNBR greater than 1.17.

As identified by the licensee, the root cause of this event was that Westinghouse, the modification designer, failed to include capacitor removal in the modification work instructions (FCN) or to perform a post modification transient test of the associated circuitry. It appears that an excessive reliance was placed on Westinghouse for FCN completeness and neither the reviews performed on the FCN nor subsequent modification reviews identified the fact these capacitors needed to be removed. Additionally, even though the PLS document change recommendation identified the change in the capacitors' time delay (identified in PLS document to be zero), the hardware changes required were not communicated nor independently identified. These problems indicate design control breakdowns between Westinghouse and the licensee's engineering organizations.

The licensee's immediate corrective action after identifying this issue was both timely and appropriate. An independent investigation team, including NSD and NAD, performed a thorough review/root cause analysis of this issue and previous related OT Delta T issues. While this issue's safety significance (discussed above) was not great, the licensee recognized the necessity of accurate RPS setpoints. Long term corrective actions will be tracked via LER 91-009 and the violation below (91-20-06). Accordingly, URI 91-19-01 is considered closed.

The chronology of engineering review/communication breakdowns for the OT Delta T and the SI issue indicates that there has been, and continues to be, significant deficiencies in engineering design control and interfaces. This is contrary to 10 CFR 50 Appendix B, Criterion III, and is identified as an apparent Violation: Inadequate Engineering Design Controls and Interfaces, 91-20-06.

The following historical examples demonstrate that the adequacy of design controls and interfaces has been a continuing problem for the licensee:

- Inadequate engineering communications and reviews were issues in 1988 and 1989 related to OT Delta T concerns previously addressed in IRs 88-03 and 89-12. These issues resulted in a violation of 10 CFR 50 Appendix B Criterion III in 1989 (Vio 89-12-02). Inadequate calculational, design, and modification testing review problems were identified with the AFW NPSH issue in late 1989 (EA 89-188 and IRs 89-11, 89-18, and 89-20). In addition, inadequate engineering reviews were also identified in 1989 involving the Agastat relay tolerance issue (IR 89-12).

- During 1990 (IR 90-22), concerns were identified with the interdepartmental communications during development of two RMS modifications. Engineering also failed to identify that one of the modification's proposed radiation monitors was inadequately ranged.

- Also in 1990, errors involving LBLOCA analysis computer codes and interpretation of TS figure 3.10-5 (IR 90-23) indicated inadequate oversight/communications between the licensee and the fuel vendor.

Additionally, a 1991 issue (VIO 91-01-03) was identified involving inadequate engineering review of modification M-1016, Electrical Penetration Replacement.

(Closed) VIO 89-09-05, Design Control Measures Were Not Adequately Established to Assure That The 50 GPM Leak Isolation Capability Design Basis Was Correctly Translated Into Specifications, Drawings, and Procedures for the RHR System. The inspectors reviewed the licensee's response dated July 26, 1989, to the NOV. The inspectors agreed that the root cause was a lack of adequate design basis and that the DBD process should correct this deficiency. The inspectors reviewed modification M-1017, Eliminate RHR Pump Common Mode Failure, which provided enhanced leak detection and isolation capabilities. This consisted of: installation of RHR pit sump level indicators on the RTGB; control room annunciation of high water level in the pit; installation of motors on the RHR-752 A, B, valves; and movement of the isolation valves for the CCW supply lines to the RHR pump Hxs and SW supply lines to the RHR room coolers into areas which would be accessible during the recirculation phase of an accident. This modification provides adequate indication of a leak in the RHR pit, as well as providing remote leak isolation capability. This item is closed.

(Open) LER 89-11, Auxiliary Feedwater System Flow Rate Could Exceed Limits of Accident Analysis. The SDAFW pump discharge flow control valve, FCV-6416, has been mechanically limited such that under steam line break accident conditions, the maximum flow rate does not result in a mass input into the CV in excess of that used in the safety analysis. This limitation results in the SDAFW pump being limited to approximately half its flow capacity during normal plant transients. Modification 1025, Upgrade AFW FCV 6416, is being developed for installation in RO 14 to correct the design deficiency. This item remains open pending installation and successfully testing of the modification.

One violation was identified.

6. Organizational Changes

On September 10, 1991, Mr. R.H. Chambers, Operations Manager, was named as General Manager - Robinson Plant. Mr. W. J. Flanagan, previosly Manager, Modification Projects, and recently licensed as an SRO, was appointed to the position of Operations Manager. On September 18, 1991, Mr. C.R. Dietz, Manager - Robinson Nuclear Project, was elected to the position of Vice President.

7. ESF System Walkdown (71710)

This inspection was performed during the EDSFI conducted the weeks of September 23-27, October 7-11, and 21-25, 1991. The details of this inspection will be documented in IR 91-21.

e.g.	For Example
ECCS	Emergency Core Cooling System
EDG	Emergency Diesel Generator
EPP	End Path Procedures
F	Fahrenheit
FCN	Field Change Notice
FCV	Flow Control Valve
FW	Feedwater
Hx	Heat Exchanger
I&C	Instrumentation & Control
IPE	Individual Plant Evaluation
IR	Inspection Report
KV	Kilovolt
LBLOCA	Large Break Loss of Coolant Accident
LCO	Limiting Condition for Operation
LOCA	Loss of Coolant Accident
M	Modification
MDAFW	Motor Driven Auxiliary Feed Water
MDNBR	Minimum Departure from Nucleate Boiling Ratio
NCV	Non-cited Violation
NED	Nuclear Engineering Department
NFS	Nuclear Fuels Section
NLS	Nuclear Licensing Section
NOV	Notice of Violation
NPSH	Net Positive Suction Head
NRC	Nuclear Regulatory Commission
NRR	Nuclear Reactor Regulation
OT Delta T	Overtemperature Delta Temperature
p.m.	Post Meridiem
PA	Protected Area
PAP	Personnel Access Portal
PC	Protective Clothing
PCN	Project Change Notice
PCT	Peak Cladding Temperature
PLS	Precautions, Limitations, and Setpoints
PNSC	Plant Nuclear Safety Committee
REV	Revision
RHR	Residual Heat Removal
RMS	Radiation Monitoring System
RNP	Robinson Nuclear Project
RO	Reactor Operator
RO	Refueling Outage
RPS	Reactor Protection System
RTD	Resistance Temperature Detector
RTGB	Reactor Turbine Generator Board
RWST	Reactor Water Storage Tank
SBLOCA	Small Break Loss of Coolant Accident
SI	Safety Injection
SRO	Senior Reactor Operator
SW	Service Water

TS	Technical Specification
UFSAR	Updated Final Safety Analysis Report
URI	Unresolved Item
VIO	Violation