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

Report No.: 50-261/89-23

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: August 15 - August 18
September 11 - November 10, 1989

Inspectors: H C Dance / for 12/15/89
L. W. Garner, Senior Resident Inspector Date Signed
H C Dance / for 12/15/89
K. R. Jury, Resident Inspector Date Signed

Contributing Inspector: R. H. Lo, Senior Project Manager, August 15-18, 1989

Approved by: H C Dance 12/15/89
H. C. Dance, Section Chief Date Signed
Division of Reactor Projects

SUMMARY

Scope:

This routine, announced inspection was conducted in the areas of operational safety verification, surveillance observation, maintenance observation, onsite followup of events at operating power reactors, 10 CFR 50 safety reviews, onsite review committee, and onsite followup of written reports of nonroutine events at power reactor facilities.

Results:

A violation with two examples was identified for failure to adequately establish measures for the suitability of processes essential to safety-related functions involving AFW acceptance testing and of weld processes on SW piping.

A violation with two examples was identified for failure to make a 50.72 report within four hours as required.

A violation with two examples was identified for failure to use a thread sealant as required by procedure when EQ transmitter FT-494 was replaced and not providing instructions appropriate to the circumstances when instrument manifold valves were installed.

Actions taken in anticipation of Hurricane Hugo were timely and prudent.

Operator error resulted in actuation of LTOPP.

Both the A and B EDG exhaust line expansion bellows had not been included in seismic analysis. Reanalysis indicated that the lines could fail during a DBE. Additional supports were added as required.

Fifty-eight of 98 installed Patel conduit seals were found with the seal grommet to be larger than recommended by the vendor and were replaced with the correct size grommets.

With the unit in a prolonged outage, the operators were not paying adequate attention to equipment status nor to the importance of accurate and complete narrative logs.

The A MDAFW pump motor rotor bars were staked to prevent occurrences of rotor bar cracking.

The licensee identified cleanliness control problems and has initiated corrective actions.

Procedures contained inadequate guidance for safety evaluations. The quality and completeness of existing reviews were inconsistent. Implementation of a draft procedure, modeled after NSAC-125, should improve quality and consistency of 50.59 reviews.

REPORT DETAILS

1. Persons Contacted

R. Barnett, Shift Outage Manager, Outage Management
C. Baucom, Senior Specialist Regulatory Compliance
*J. Cribb, QC Manager, Quality Assurance/Quality Control
D. Crook, Senior Specialist, Regulatory Compliance
*J. Curley, Manager, Environmental and Radiation Control
*C. Dietz, Manager, Robinson Nuclear Project
*J. Eaddy, Support Supervisor, Environmental and Radiation Control
R. Femal, Shift Foreman, Operations
*W. Flanagan, Outage Manager, Outage Management
*S. Griggs, Technical Aide, Regulatory Compliance
E. Harris, Director, Onsite Nuclear Safety
R. Johnson, Manager, Control and Administration
*J. Kloosterman, Director, Regulatory Compliance
D. Knight, Shift Foreman, Operations
E. Lee, Shift Outage Manager, Outage Management
D. McCaskill, Shift Foreman, Operations
R. Moore, Shift Foreman, Operations
*R. Morgan, Plant General Manager
D. Nelson, Shift Outage Manager, Outage Management
*M. Page, Manager, Technical Support
D. Quick, Manager, Plant Support
D. Seagle, Shift Foreman, Operations
*J. Sheppard, Manager, Operations
*R. Smith, Manager, Maintenance
R. Steele, Shift Foreman, Operations
*K. Williams, Senior Engineer, Onsite Nuclear Engineering Department
H. Young, Director, Quality Assurance/Quality Control

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

*Attended exit interview on December 5, 1989.

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

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 (including CV) 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, operations' 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, (except as described below), cognizant of in-process 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 occasionally 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.

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.

During the daily review of the control room operator logs, the inspectors have identified deficiencies with log-keeping practices. Log-keeping requirements are delineated in OMM-001, Conduct of Operations, revision 23, sections 3.6, 5.5.3, and 5.7.3. The deficiencies identified primarily relate to the status of safety-related equipment (HVH coolers, DS Diesel Generator, etc.) status as they were taken out-of-service and placed back in-service. These evolutions have not been consistently logged, nor has the equipments' status been routinely and accurately carried over from shift to shift within the logs. Section 5.5.3 of the above referenced procedure describes the control operator's review of previous shift logs. Since shift logging anomalies were frequently not recognized for several shifts or were pointed out by the inspectors, it appears that the operators were not paying adequate attention to equipment status nor to the importance of accurate and complete narrative logs during the current prolonged outage. The inspectors discussed this situation with the responsible shift operators, shift foremen, and the Operations Manager. The fact that violation 88-01-03 is currently open involving this area was stressed. The Operations Manager informed the the inspectors that he had taken corrective actions and will take further corrective actions. The adequacy and effectiveness of these actions will be verified prior to the closure of violation 88-01-03.

No violations or deviations were identified.

3. Monthly Surveillance Observation (61726)

The inspectors observed certain safety-related surveillance activities on systems and components to ascertain that these activities were conducted in accordance with license requirements. For the surveillance test procedures listed below, the inspectors determined that precautions and LCOs were adhered to, the required administrative approvals and tagouts were obtained prior to test initiation, testing was accomplished by qualified personnel in accordance with an approved test procedure, test instrumentation was properly calibrated, the tests were completed at the required frequency, and that the tests conformed to TS requirements. Upon test completion, the inspectors verified the recorded test data was complete, accurate, and met TS requirements, test discrepancies were properly documented and rectified, and that the systems were properly returned to service. The inspectors witnessed/reviewed portions of the following test activities:

*OST-051 (revision 11)	Reactor Coolant System Leakage
*OST-207 (revision 12)	Motor Driven Auxiliary Feedwater Pump Flow Tests
*OST-401 (revision 24)	Emergency Diesels
*OST-702 (revision 13)	ISI Secondary Side Valve Test
*OST-703 (revision 21)	ISI Primary Side Valve Test
*OST-910 (revision 11)	Dedicated Shutdown Diesel Generator
*OMM-21 (revision 24)	Operation During Adverse Weather Conditions
*EST-013 (revision 8)	Auxiliary Feedwater Pump Bearing Temperature Test
*MST-022 (revision 2)	Safeguard Relay Rack Train A
*MST-023 (revision 3)	Safeguard Relay Rack Train B

No violations or deviations were identified.

4. 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. The inspectors observed/reviewed the following maintenance activities:

*WR/JO 89-AFGM1	Disassembly/Inspection and Repair of the SDAFW Pump
*WR/JO 89-AHXL1	Disassembly/Inspection and Repair of A AFW Pump
*WR/JO 89-AHXL2	Disassembly/Inspection and Repair of B AFW Pump

*WR/JO 89-AJTT1 Removal/Reinstallation of A and B SW Booster Pump Suction and Discharge Lines

*WR/JO 89-AJUS1 SW System Piping Flush/Inspection

*CM-007 (revision 4) Electric Driven Auxilary Feedwater Pump Overhaul

*CM-008 (revision 6) Steam Driven Auxilairy Feedwater Pump Overhaul

*CM-306 (revision 2) Replacement of Test Switches in Nuclear Safeguard Systems

*CM-310 (revision 2) Installation of Patel Conduit Seals

*CM-625 (revision 1) Rotating Shaft Flexible Coupling Alignment

In addition, twenty work request activities were observed/reviewed involving all phases of work associated with Patel seals including torque, replacement of grommets, inspection, and leak testing.

ALARA

On November 9, 1989, the inspectors observed Patel conduit seal inspection of FT-474 and FT-475. The inspectors observed that the maintenance technicians were working in a radiation field of between 80 and 100 mrem/hr. Most of this exposure was caused by a nearby letdown line. The work activity took approximately one hour to perform. However, the inspection revealed that the wiring to the transmitters required replacement prior to re-installation of the Patel seal. In addition, four additional transmitters in the area also required inspection. The inspectors discussed with the onshift HP that due to the length of the projected work, shielding should be considered for the area. Temporary shielding was subsequently placed on the line. This reduced the radiation to approximately the background levels for the general area, 25 mrem/hr. The inspectors discussed this item with the E&RC Manager. Specifically, the inspectors expressed concern about the ALARA planning for work in this area and the ALARA criteria for placement of shielding. The E&RC manager indicated that he would review the circumstances surrounding this work and the current shielding practices to determine if additional measures need to be taken to implement ALARA. This is an IFI: Review Use Of Shielding For Implementation Of The ALARA Program, 89-23-01.

AFW System Maintenance/Modification

In addition to the AFW suction piping modifications detailed in paragraph 2, the following maintenance/modifications were performed on the MDAFW pumps, SDAFW pump, and the A MDAFW pump motor, respectively.

The MDAFW pumps' repairs as delineated in IR 89-17, were performed basically as anticipated. Worn parts/components were replaced or refurbished, with the majority of the motor-driven pumps refurbishment consisting of the various damaged impellers and diffusers replacement. The motor-driven pump casings were weld repaired where damaged and machined as required by the pump vendor. During this process a problem was identified with the pump casings, in that, during the machining of the A and B MDAFW pump housings at the Dresser (vendor) Pump Repair facility, hairline cracks were discovered throughout the housings when the machined surfaces were dye penetrant tested. These cracks were discovered after Dresser had refurbished the housings and were found to be 1/4 inch to 1 inch long, with a depth of up to 1-1/4 inches. These housings had been penetrant tested (witnessed by the inspectors) prior to leaving the site for refurbishment; only minor surface indications were identified. Based on an Engineering Off-Normal Condition Report (EONC-TS-89-016), it appears that the cracks are an inherent characteristic of all commercial grade castings, and that the cracks resulted from the casting cooling process; not from the casing being overstressed. The licensee performed comprehensive actions in repairing the pumps. A licensee welder repaired the inclusion-affected areas after they were ground out, using an approved CP&L procedure and Q-list materials. The weld repairs were inspected and approved by certified QC inspectors, with the entire evolution overseen by a Technical Support engineer. The pump housings will be inspected for future damage the next time the pumps are inspected per the PM schedule the licensee is planning on implementing. This planned PM implementation is also applicable to the SDAFW pump. The damaged SDAFW pump impeller and diffuser was "buffed" to smooth the cavitation-damaged areas, and was subsequently reinstalled. Any potential pump future damage or system anomalies will be monitored through the ISI/IST program and the planned PM schedules. This is identified an an IFI: Review Planned PM schedule for AFW components, 89-23-02.

As discussed in IR 89-17, the MDAFW pump motors were sent to Westinghouse for inspection/refurbishment. Specifically, the A pump motor had exhibited arcing and sparking during starts (IR 89-17 had incorrectly identified B as the affected motor). Results of this inspection revealed that this motor had incurred rotor bar cracking. Per the Westinghouse evaluation, the failure was caused by a fractured and bent rotor bar, which deflected outward through the air gap and shorted with the stator. The cracked rotor bars were observed at both the inboard and outboard sides of the motor. The fracture (cracks) was attributed to metal fatigue. This failure was evidently caused by the fact that the rotor bars were not staked or swaged to the rotor slots (thus precluding slip), coupled with motor starting in excess of the frequency anticipated in the plant fluid systems design and the effective motor start times. Westinghouse determined that since the HBR configuration was designed for low in-rush current, it is likely motor starting voltage may also be low. The pump motor was not designed for low voltage starting capability.

This is significant in that, if starting voltage less than the nominal (90 percent of rated) assumed in the motor's design, lengthy motor start times could have resulted, causing substantial slip during the starting time. This phenomenon may have greatly increased the number of magnetic stress cycles experienced during each start. These three factors apparently caused the A pump rotor cracking; the rotor bars were subsequently replaced and swaged. According to Westinghouse, swaging of the rotor bars should eliminate the mechanism which resulted in the rotor bar cracking. Motor start voltage will be verified during the testing for Modification M-1025, AFW Steam Driven Pump Flow Control Valve Setpoint Change. This is identified as an IFI: Verify MDAFW Pump Motor Starting Voltages, 89-23-03.

After the maintenance/modifications were completed, the inspectors witnessed the performance of Special Procedure SP-896, AFW Hydro, revision 1, Operations Surveillance Test, OST-207, Motor Driven Auxiliary Feedwater Pump Flow Test, revision 12, and Engineering Surveillance Test EST-013, Auxiliary Feedwater Pump Bearing Temperature Test, revision 8, respectively. No significant anomalies were identified during performance of the above procedures.

Service Water System Debris

On October 24, 1989, an escalating differential pressure increase was identified on CV cooling unit HVH-4. Differential pressure was being monitored on this cooling unit due to biological fouling problems identified in 1988 (see Inspection Report 88-28 for details). In July, 1989, a slight increase in HVH-4 differential pressure was noted; however, during the outage the differential pressure had continued to increase. Values taken were 12.0 psid at 910 gpm flow on September 5, 1989, 12.6 psid at 910 gpm flow on October 4, 1989, and 14.0 psid at 930 gpm on October 24, 1989. The licensee, in an effort to determine root cause, performed an inspection of HVH-4. This inspection revealed an approximate nine feet section of 1/4 inch purge hose (origin is still being investigated) and an accumulation of numerous black "chunks" of an unknown material inside the unit's water box. No significant biofouling was observed and appeared to have not increased subsequent to the last inspection in April, 1989.

A sample of the black chunks was sent to the Harris E & E Center for analysis. The analysis determined the substance as being a bituminous or asphaltic material (e.g. coal tar). Sections of the SW system (SW booster pump suction and discharge piping) were identified as being coal tar lined carbon steel pipe. The coal tar lining is utilized to prevent corrosion of the piping. The piping which contained this lining was then opened and inspected. A piece of the coal tar lining was sent to the Harris E & E center for analysis, with confirmatory agreement with the material removed from HVH-4. The visual inspection of the piping revealed "drooping" and melted sections of the pipe liner. Modification

M-960 had been performed during the 1988 RO on the SW booster pump piping to weld stiffening saddles on the six-inch lines going to the four cooling units, and to weld pipe restraint lugs on the twelve-inch booster pump suction piping. Failure of the coal tar lining was attributed to the welding process and associated heat treatments performed during this modification. The effect of welding on this piping was apparently anticipated during the actual welding; however, the associated heat treatments appeared to have caused this phenomenon (melted liner) which was not anticipated. A more thorough engineering analysis could have precluded this problem from occurring, as it appears that the design engineer did not consider the effects of the welding heat on the lining, and therefore did not include necessary precautions in the modification's installation requirements. Failure to adequately establish suitability of the welding processes in M-960 is identified as a VIO: Failure To Adequately Establish Measures For Suitability Of Processes Essential To Safety-related Functions As Required By 10 CFR 50 Appendix B Criterion III, 89-23-04.

As a result of the foreign materials presence in HVH-4, the other three cooling unit's fan and motor coolers were inspected. These inspections revealed a similar amount of the same foreign material (black chunks) in each of the three other units. No significant biofouling was identified in any of these cooling units. Immediate and long-term corrective action were initiated to improve cleanliness controls and are detailed as follows:

Immediate Corrective Actions

1. The HVH units were cleaned and their respective tubing was cleared with pressurized air.

2. As much "loose" coal tar lining was mechanically removed from the affected piping as possible.

3. All affected piping (from the suction of the SW booster pumps to the outlet piping of the HVH units) was flushed in accordance with Special Procedure SP-898.

4. Ultrasonic pipe thickness measurements were performed at all locations affected by M-960, as well as other portions of the coal tar-lined piping.

5. An engineering evaluation, EE 89-113, was prepared to support JCO 89-11 for operation with the SW piping in the current condition for the next twelve months.

Long-Term Corrective Actions

1. Continue monthly monitoring of HVH-4 differential pressure via PLP-006, the monthly trending of HVH-4 performance data and via EST-102. Prior to the AFW system outage, no anomalies were identified in the data for collected from either of these monitoring systems for HVH-4.

2. Replace the piping that has been damaged during the 1990 RO.

The inspectors monitored the immediate corrective actions as they were implemented and will review the associated long term corrective actions as they occur. The involved EE 89-113 and JCO 89-11 were reviewed and appeared to be complete and comprehensive for short-term acceptability.

In addition to the situation described above, a problem was identified with foreign material in the SW System piping. Specifically, after repairs were performed on the B SW booster pump suction elbow, a 9" by 2" by 4" piece of wood was found at the booster pump impeller during an inspection. The pump was being inspected becaused of a failure to meet the acceptance criteria of the normal surveillance procedure which was being used to return the pump to service. An SCR (89-075) was initiated to determine root cause and the wood's origin. At the end of the report period, no root cause nor origin had been established; however, interim measures have been established to preclude recurrence. These included cleanliness inspections to be performed by QC prior to reassembly of any safety-related, Q-list components. After discussions with the Maintenance Manager and QA/QC Manager, it appears that cleanliness controls were being sporadically implemented. Failure to establish and/or consistently implement cleanliness controls on safety-related components and systems is a violation of 10 CFR 50 Appendix B Criterion V. This violation meets the criteria specified in Section V of the NRC Enforcement Policy for not issuing a Notice of Violation and is not cited. The establishment and effectiveness of long term corrective actions is identified as an IFI: Review Implementation And Effectiveness Of Cleanliness Controls, 89-23-05.

Improperly Installed Instrument Manifold Valves

On November 3, 1989, while observing re-torquing of Patel seals, the inspectors observed that some S/G level and steam flow instrument manifold valves and associated tubing were not supported in a manner consistent with the original configuration. Subsequent walkdown of instrument racks by the licensee resulted in a list of 22 manifold valves (16 inside the CV and 6 outside) which were not installed as originally designed. A seismic analysis was performed on the as-found configurations. The as-found conditions were determined to be acceptable. The licensee elected to re-install the manifold valves and tubing in accordance with the original design. The work was accomplished by WR's 89-AKKF1 and 89-AKKZ1. Review of work requests revealed that the majority of the observed discrepancies were caused by a replacement of discontinued manifold valves with a new style manifold valve. For example, the S/G steam line flow transmitters FT-474, 475, 484, 485, 494, and 495 had their respective manifold valves replaced in December 1988. All six flow transmitters did not have their respective manifold valves secured to the existing supports. The as-found configurations constituted an authorized modification to the facility in that no evaluation had been performed to justify not attaching the new manifold valves to the existing supports, Failure to provide adequate procedures for proper installation of instrument manifold valves is a VIO: Failure To Establish And Implement Procedures As Required By 10 CFR 50 Appendix B Criterion V, 89-23-06.

RHR Pump Discharge Check Valve Degradation

During hurricane preparation testing of the RHR system, the B RHR pump discharge check valve, RHR-753B, was identified as not fully seating. Only a 10 psig difference between the A and B RHR pump discharge pressures existed when the A RHR pump was operating. Inspection of the valve internals revealed that the disc stem had experienced wear such that the disc would become misaligned with the seat during closing. A new disc assembly was installed per WR/JO 89-AISA1. The valve was successfully tested and returned to service. Inspection of the check valve on the A RHR loop did not reveal similar of wear. These are the only two valves of this type, size, and manufacturer installed in safety-related applications.

Two violations were identified.

5. Onsite Followup of Events at Operating Power Reactors (93702)

Hurricane Hugo

On September 21, 1989, while plotting the path and forces of Hurricane Hugo, the licensee implemented OMM-21, Operation During Adverse Weather Conditions, revision 5. This procedure delineates the specific instructions for the operation of H. B. Robinson during hurricanes or tornadoes. At the time of procedure implementation the plant was in cold shutdown conditions, thus precluding any anticipated mode changes (i.e. placing the plant in hot shutdown two hours prior to the anticipated arrival of hurricane force winds at the site). The inspectors verified that various procedural actions were performed, including: securing of loose equipment, placement of personnel safety hand lines, review of inoperable equipment and equipment status, EDG testing as necessary, and the testing of safeguards equipment (pumps, HVH units, etc.) which was not currently operating. All of the above actions were performed satisfactorily; however, during the testing of RHR pump A, a discrepancy was identified with operation of valve RHR-753B, the B pump discharge line check valve. This situation is discussed in detail in paragraph 4. The inspectors considered the actions taken to prepare for potential adverse impacts of Hurricane Hugo to be both timely and prudent.

Hurricane Hugo passed through the area in the late evening of September 21 and early hours of September 22, 1989. During this time, county power supplying some buildings was lost. The EOF/TSC building and security system was transferred to the backup EOF/TSC/PAP diesel generator. Offsite power to the safety-related facilities were never lost. At 1:15 a.m. on September 22, 1989, the onsite meteorological data system was lost due to the loss of the county power system. Normal telephone service was also interrupted. However, the licensee intra-company phone system remained operable and meteorological data was obtained through the corporate office. The red phone also remained in service. The NRC contacted the site six times via the red phone for status updates during the passage of the hurricane. The site incurred only minor wind damage. A corner of the old administration building had the roof damaged, thereby, allowing water damage to some offices. Item I.4 of Appendix B of IR 89-11 addresses a concern involving not declaring a NOUE during the hurricane. See IR 89-11 for additional information.

AFW NPSH Issue

Inspection Reports 89-17, 89-18, and 89-20 discussed the circumstances leading up to and including the shutdown on August 22, 1989, due to degraded AFW performance. Report 89-17 also addressed proposed modifications to the AFW suction piping to address the inadequate NPSH concern. The following is a description of the actual modification, M-1018, which was installed and subsequent tested to allow restart of the unit.

A common suction line supplying all three AFW pumps consists of two 6-inch outlet lines from the CST which join into a 12-inch line. From the SDAFW-MDAFW branch, a 6-inch line was run to join the existing 6-inch line just upstream of the common suction isolation valve for the MDAFW pumps. From the SDAFW-MDAFW branch, a short section of 12-inch pipe continues toward the SDAFW pump and then necks down to an 8-inch line which connects to the pump suction nozzle. The 8-inch section of pipe includes a 6-inch check valve and 8-inch suction isolation valve. The new piping is stainless steel. On October 19, 1989, acceptance test procedure AT-2 contained in field revision no. 19 to M-1018, was performed on the A and B MDAFW pumps to validate the analytical model used to show that available NPSH would exceed required NPSH when all three AFW pumps operate at design flow rates. The test was observed by the inspectors. Results of the test demonstrated that the measured pressure drops in a section of suction piping was less than predicted by the analytical model. Thus, it is anticipated that the actual pressure losses with all three pumps operating will be less than predicted, e.g.

the available NPSH will be greater than predicted. Since the SDAFW pump cannot be tested until steam is available, i.e. after startup, this validation provided reasonable confidence that all AFW pumps are operable. Additional flow testing of the SDAFW pump will be performed after startup to further validate the analytical model. The inspectors plan to witness this testing when performed. During performance of the test, the test engineer was concerned that air was trapped in the 12-inch section of the common suction pipe. The piping configuration was such that both the 6-inch pipe going to the MDAFW pumps and the 8-inch pipe going to the SDAFW come off the bottom side of the 12-inch header. Since the 12-inch header is horizontal with no vent, it is not possible to completely vent this section of pipe. This had been brought to the licensee's attention by the inspectors during construction activities, but was not considered as a potential problem. The piping was modified to add a vent just upstream of the 8-inch pipe connection. The inspectors observed that after the normal fill and venting activity was performed, opening of the new vent did result in air blowing out of the header. Because of the concern that air in the line might have adversely effected the test results, the test was again performed on October 22-23, 1989. Results of the second test did not deviate significantly from that obtained during the first test.

During review of the test data taken on October 19, 1989, the inspectors observed that attachments 8.10, 8.11, and 8.12 of AT-2 did not correct for the difference between the pipe center line of the 12-inch pipe and that of the 4-inch suction lines to the MDAFW pumps. Further review by the licensee indicated that the velocity term of Bernoulli's equation had also not been applied. Thus, the values determined in the attachments to be compared with the analytical results of calculation RNP-MN/MECH-1055 revision 0, would not result in a valid comparison. These deficiencies were corrected by field revision 22 to M-1018. The above described results were based upon the methodology provided in revision 22. Failure to incorporate in revision 19 a valid methodology for determining the acceptability of an analytical model for NPSH determination of the AFW System is a violation. This is considered as a second example of violation, 89-23-04.

Failure of SDAFW Flow Control Valve - Unanalyzed Condition

On October 6, 1989, the licensee reported per 10 CFR 50(b)(2)(i) that a failure of the SDAFW pump discharge flow control valve, FCV-6416, during a main steam line break accident could result in AFW flow to a faulted S/G in excess of the quantity assumed in the accident analysis. During the report period the licensee has developed a modification, M-1025, to limit the flow from the SDAFW pump to approximately 300 gpm at normal operating pressures. This would limit the flow to a depressurized S/G during a steam line break to approximately 625 gpm, while still supplying a minimum of 240 gpm to the S/Gs under other FSAR Chapter 15, accident/transient conditions. Limiting the SDAFW pump flow to 625 gpm would, when added to the 650 gpm flow from both MDAFW pumps, result in a value of less than 1300 gpm as assumed in the steam line break analysis. The licensee has also performed a safety analysis which indicated that the 1300 gpm could be increased to 1325 gpm with no significant reduction in the margin of safety. Limiting of the flow will be accomplished by adjusting a mechanical stop on FCV-6416. The mechanical stop has been adjusted to a pre-determined position based on a hydraulic model of the system. However, final positioning of the mechanical stop will be accomplished during startup of the unit. When sufficient steam is available to operate the SDAFW, flow measurements will be taken and compared with the anticipated flow rate at the pre-determined position. The mechanical stop will be adjusted as necessary to make the measured flow rate agree with the desired calculated flow rate at the actual S/G pressure. The inspectors plan to observe this evolution when performed.

During a PNSC on August 18, 1989, the licensee discussed that a single failure could cause FCV-6416 to fail full open and that steps needed to be taken to compensate for increased flow due to reduced S/G pressure during a steam line break event. At that time the licensee failed to consider that anticipated flows in excess of 1300 gpm constituted an unanalyzed condition which per 10 CFR 50.72 required reporting of the condition to the NRC. Actual performance of the AFW system with all three pumps running during two reactor trips had indicated that the actual flow was limited to approximately 900 gpm. However, this was close to the normal operating pressure, not a depressurized system. On August 22, the licensee had identified that the low flow rates was mainly attributed to a higher than anticipated friction loss in the suction piping. Subsequent inspection of the suction piping indicated that the pipe was rougher inside than assumed in the design calculations due to degradation of the pipe surface. Thus at that point, by August 30, 1989, the licensee should have known that prior to degradation of the pipe and with normal operating levels in the condensate storage tank, the 1300 gpm flow rate to a depressurized S/G would have have been exceeded. Failure to make the report within four hours of the determination of a reportable condition is a violation: Failure To Make A 10 CFR 50.72 Report Within Four Hours As Required, 89-23-07.

LTOPP Actuation

On October 16, 1989, while the unit was in cold shutdown, operator error resulted in actuation of the LTOPP system. Prior to the actuation, the RCS was water-solid at 160 degrees F and 350 psig. The A RCP and the A RHR pump were in service. An operator had been given instructions to raise the RCS temperature to between 180 and 185 degrees F. To accomplish this task the operator closed the only RHR Hx discharge valve which was open. This action isolated the primary system pressure control valve, PCV-145, and resulted in an almost instantaneous increase in RCS pressure to the LTOPP actuation setpoint. The system performed as expected and limited the pressure increase to 418 psig. On November 7, 1989, the licensee issued a special report concerning this event in accordance with TS 3.1.2.1.e..

EDG Exhaust Line Seismic Qualification

During a review of issues that could have affected Unit start-up from the AFW outage, the licensee identified a concern regarding EDG exhaust piping seismic acceptability. On October 20, 1989, the licensee reported to the NRC that as a result of seismic re-analysis, both EDGs could be rendered inoperable during a seismic event. This analysis indicated that the two expansion bellows on the A EDG could fail during a DBE. The resulting release of gases into both EDG rooms could cause either or both EDGs to fail due to mechanical problems caused by exhaust back pressure or high temperature conditions.

LER 89-012 was issued on November 20, 1989, and concluded that the cause for this condition existing was due to improper assumptions made during previous design reviews. The improper assumptions were that the EDG exhaust piping was not Seismic Class I piping (based on existing design information) and that the Chart Method analysis could be utilized for this piping. The Chart Method has only been utilized at HBR for small bore (2 1/2 inches and under), non-thermal piping. Upon identification of this situation, the licensee took expedient corrective actions by installing a DCN to Modification M-955, Emergency Diesel Generators A and B Upgrade. DCN 955-16 entailed the addition of three Class I supports for the A EDG exhaust line and one support for the B exhaust line. This modification ensures that each EDG's respective exhaust line meet Seismic Class I criteria.

Pending further review by the NRC of the circumstances surrounding the discovery and corrective actions taken to preclude reoccurrence, this item is considered as a URI: EDG Exhaust Lines Not Seismic Qualified, 89-23-08.

Patel Conduit Seals

During the week of October 22, 1989, an inspection of open EQ issues was conducted by a Region II inspector. Subsequent in-office review of documentation associated with Patel conduit seal qualification resulted in a telephone conference call on November 3, 1989.

In response to a concern involving potential relaxation of the torque applied to Patel seals, the licensee committed to verify that Patel seals were properly torqued. The inspectors observed field re-torquing of approximately 30 Patel seals associated with Rosemont transmitters. During this verification, pressure transmitter PT-921's seal was observed to exhibit movement as the torque was being verified. The seal rotated approximately 30 degrees while being torqued to the required 50 Ft.-lbs. This is approximately the amount of rotation required to loosen a seal so it can be removed by hand. Approximately half of the other seals moved slightly, 10 degrees turn or less. The licensee reported that the re-torquing of the remainder of the seals on November 3 and 4, resulted in either no movement or slight movement as described above. The licensee is reviewing this phenomenon to determine if periodic re-torquing of the seals is necessary to prevent torque relaxation.

Prior to completion of the torque verification, the licensee determined that some seals may not have contained the proper size grommet for the wire(s) penetrating the grommet hole(s). This determination resulted from the evaluation of a Region II question concerning minimum wire sizes utilized for different grommet sizes. Grommets are sized in accordance with the hole size required for different AWG wire sizes, e.g. an AWG #16 wire required a #16 grommet. However, the AWG number referred to the conductor diameters and did not consider variations in wire insulation thickness. As a result, the licensee determined AWG #16 wires could require a smaller grommet size, and that 69 of the total 98 EQ Patel seals would require field measurement of wire diameter for potential replacement or other corrective actions. Twenty-nine Patel seals were excluded based upon review of purchase specifications. The specifications indicated that the wire diameters including insulation thickness were within the use range of the installed grommets.

On November 6, 1989, the inspectors requested the licensee to evaluate the impact of the torque verification and proper wire/grommet combination verification on the connecting threaded joints between the Patel seal and the sealed device. In some instances these evolutions involved torquing of the Patel seal union nut to 50-55 Ft-lbs. while attempting to hold a round stainless steel pipe nipple with a pair of channel lock pliers. In some instances, the inspectors had observed twisting of the connecting joints(s). Even though the torquing would tighten the joints, it could also break the cured Loctite sealant on the threaded sections. The licensee determined that only Patel seals associated with Rosemont transmitters, required leak testing. The method chosen was similar to that recommended by Rosemont to verify integrity of a joint after resealing it. The test involved installation of electronic housing covers with airline tubing connections and pressurization of the assembly to 50 psig. Leak testing of 40 transmitters revealed 3 transmitters with leaking threaded joints. Two of these involved only minor leakage, i.e. individual bubble formation. However, FT-494, C S/G steam line flow instrument exhibited a sustained blowing at the joints associated with the pipe nipple between the Patel seal and the transmitter. Visual inspection of the joints revealed that a thread lubricant had been utilized on the joint rather than a thread sealant. Review of WR/JO 88-ADHN1 revealed that this transmitter had been replaced on March 10, 1988, and that the work performed was accomplished in accordance with

CM-310, Installation of Patel Conduit Seals. At that time, Step 7.16 of CM-310 specified assembly of the pipe nipple to the low pressure seal housing and to the instrument housing using an approved thread sealant. The WR/JO parts list did not identify a thread sealant as having been used; however, the WR/JO indicated that lubricant N-5000 had been issued. Failure to use a thread sealant as specified by CM-310, revision 1, is a violation. This is considered as a second example of violation, 89-23-06.

During pressure testing of the first transmitter, LT-484, B S/G fluid level instrument, the inspectors observed an unexpected phenomena. When using a hand pump, pressure could only be increased slightly more than 20 psig. However, when a regulated air supply was attached, minimal external leakage was detected. On the next transmitter tested, LT-485, both methods successfully pressurized the transmitter and no leakage was detected. The licensee elected to use the regulated air supply on subsequent tests since the purpose of the leak test was to detect leakage out of the threaded connections. This method of pressurization was preferred because it was quicker and easier.

On November 7, 1989, the inspectors discussed with the cognizant engineer the observed phenomena. The inspectors expressed the concern that there must be leakage through the seal internals, e.g. by or through the grommet area. The next evening, LT-484 was again pressurized. However, this time, the conduit to the Patel seal had been loosened without disturbing the seal to allow detection of leakage around the grommet. The soap solution continuously foamed at pressures as low as 15 psig. Testing was not conducted at pressures less than 15 psig. Measurement of the total wire diameter resulted in an average thickness of .108 inches, i.e. less than the .115 inches minimal value recommended for the installed #16 size grommet. Level transmitter LT-484 had been one of the 29 Patel seals which had been excluded from inspection based upon the purchase order review. Subsequent investigation by the licensee revealed that the insulation diameter listed in the manufactor's specification included a braided cover which must be removed before placing the wire through the grommet. The licensee then included the remaining 29 Patel seals in the inspection program.

On November 4, 1989, the inspectors observed during re-assembly of the Patel seal for LT-928, C accumulator fluid level instrument, that the insulation above the seal was cut thereby exposing the conductor. The damaged cable was replaced WR/JO 89-AKBH1 by splicing a new cable section to the existing cable.

Field verification of the as-built Patel seal configurations resulted in the following actions; A total of 58 #16 size grommets required replacement with #18 size grommets. These were associated with transmitters, solenoid valves, limit switches and temperature elements. Ten seals were either found to be within desired tolerances or evaluated as acceptable. A total of 29 seals associated with ASCO solenoids, had the conduit leading to the Patel seals drilled such that liquids or condensation would be directed away from the solenoids. According to the licensee, this more closely represented the test configuration used by ASCO during EQ testing. One ASCO solenoid was found with a hole already in the conduit. The licensee has stated that all 30 of these solenoids are not required to be energized or re-energized during long term recovery operations. This position was reviewed by Region II personnel with responsibility for the EQ inspection program and found to be acceptable.

On November 6, 1989, at 9:43 a.m., the licensee reported per 10 CFR 50.72 (b)(2)(iii)(D) that a potential deficiency existed in the EQ of both trains of instrumentation required to mitigate the consequences of an accident. The licensee reported that this condition had been identified at approximately 1:00 p.m. on November 3, 1989. On November 4, 1989, the licensee had confirmed that this condition did actually exist on both trains associated with the accumulator fluid level instrumentation. Failure to make the report within four hours of the determination of the condition is considered as a second example of a violation, 89-23-07.

The preceding documentation is a current update of issues associated with IR 89-26. Additional enforcement action concerning application of Patel conduit seal outside of their grommet size use range, if any, will be addressed in IR 89-26.

One violation was identified with two examples. An additional example of each of the two previously identified violations was identified.

6. 10 CFR 50.59 Safety Review

A review was conducted of the licensee's procedure to perform safety reviews for plant modifications. Plant procedure, MOD-013 governed the performance of 50.59 reviews. The stated purpose of this procedure was to provide the guidelines for performing a nuclear SR as required by 10 CFR 50.59, 10 CFR 72.48, TS Sections 6.5.1.1 and 6.5.1.2, and the ISFS Special Nuclear Materials License. The procedure defined the need for a SR, specified the Qualified Nuclear Safety Reviewers List for the performance of a SR, defined acceptable SR methods, outlined a SR procedure, and included a checklist/questionnaire for the Safety Review Report.

Procedure, MOD-013 (revison 4), was reviewed against the current industry guidance for performing 50.59 reviews, NSAC-125, "Guidelines for 10 CFR 50.59 Safety Evaluations," dated June, 1989. The NSAC-125 guidelines generally contain NRC review comments and are considered the best consensus guidance currently available. The procedure was found to be inadequate in providing the guidance to perform the Safety Evaluation. Examples of some of these inadequacies are listed below:

* No guidance on performing analysis which would determine if there is an USQ, except a listing of the three criteria.

* No guidance on the performance of a safety evaluation if the modification is not directly associated with a SSC described in the SAR. NSAC-125, Section 4.1.1 outlines the need to evaluate those modifications.

* No procedure for Safety Evaluation Disposition. MOD-013, revision 4, provides no guidance on changes which are deemed to involve a USQ. For example, there was no provision in MOD-013 to facilitate a plant modification which would result in an overall safety improvement and which also involves a USQ.

In addition to the review of the MOD-013, a review was conducted of recent EEs associated with plant modifications. Each modification reviewed is discussed below:

* M-934 - Auxiliary Building Ventilation System Upgrade

  This modification involved changes in ventilation equipment and ventilation flow-paths (by installation of air transfer ducts, automatic release mechanism on fire doors, duct-work in pipe alleys and relocation of ventilation vents) to correct inadequate ventilation in the RAB, in part as a result of Appendix R related changes. This modification also involved changes to the Evaporative Air Coolers by replacement of a recirculation system of service water with that of a once-through system.

  The SR did not contain information on the design basis of the affected portion of the RAB ventilation system. There was no information on the air flow requirements of the safety equipment serviced by the RAB ventilation system being modified or how the modified system/air flow would continue to satisfy those requirements. There was a statement that the modification would "restore 80 percent of the design flow." There was no information on the adequacy of 80 percent of the design flow nor was there information on the design flow at various affected areas.

  The SR contained a statement that the modification package has been reviewed for USQ and each of the three criteria for no USQ were met. However, there was no supporting basis for meeting the criteria of no USQ determination.

* M-938 - Eliminate Turbine Trip From Loss of Main Feed Pumps

  This modification eliminated the design feature of turbine trip from the loss of both main feedwater pumps. This was accomplished by disconnecting the input circuits which initiated the turbine trip when both main feedwater pump breakers were in the open position.

The SR performed a review of the updated FSAR and determined that as a result of this modification, a change to a figure (Figure 7.2.1-28), "Turbine Trip Signal" would be necessary. The SR also pointed out that the turbine and feedwater flow mismatch trips "are not used in the (FSAR Section 15) transients and accident analysis." With that statement as the only basis, the SR concluded that this modification meets all the 50.59 criteria for no USQ.

It is apparent that the trip feature was part of the SAR description (i.e., Figure 7.2.1-28). Although not explicitly credited in the SAR accident analysis, the trip logic is an anticipatory trip to prevent a loss of heat sink. At power levels greater than 10 percent, the reactor protection logic also trips the reactor when a turbine trip occurs. The SR failed to address the design basis of the trip feature and the safety implication of its elimination. 10 CFR 50.59(b)(1) requires a safety evaluation which provides the basis for determination that the changes in the facility as described in the SAR does not involve a USQ. The guidance for going beyond the Chapter 15, Accident Analysis, is also explicit in NSAC-125 which states, "10 CFR 50.59 is also applicable to other events with which the plant was designed to cope and described in the SAR."

* M-985 - Remove C Solenoid Valve from HVH Damper

In conjunction with the A and B solenoids, the C solenoid valve was a two-way valve which will de-energize and vent an instrument-air line upon receipt of an SI signal to allow the spring-loaded emergency ventilation inlet valves to open. All three solenoids are required to be environmentally qualified. However, the C solenoid cannot meet the 10 CFR 50.49 (EQ) requirement and a qualified replacement did not exist. The purpose of this modification was to remove the C solenoid valve and cap the associated HVH emergency damper actuation air-exhaust line.

The Safety Evaluation for this modification contained a thorough FSAR review on the functional requirements of the solenoid valves. It pointed out the redundant feature of the three valves (such that removal of the C solenoid valve will still meet single failure criteria) and it provides a discussion of a functional test to assure that the emergency dampers will still open within the time period assumed in accident analysis with the removal of the C solenoid valve. With these discussions, the conclusion of the no USQ was supported by adequate basis and the evaluation was in full compliance with the guidance of MOD-013.

As described above ,the present plant procedure to perform 50.59 reviews (MOD-013, revison 4) when compared with NSAC-125, was inadequate in providing guidance to perform the safety evaluations. A review of a few selected modification packages found that the quality and completeness of the 50.59 reviews was inconsistent. This inconsistency was a reflection of the inadequacy of guidance of the present plant procedure. However, the licensee has recognized the need for improvement in this area. A company-wide task force has been established for a revision of the procedure which, when adopted, will be applied to all the licensee's nuclear facilities. The draft revision appeared to be a close model of NSAC-125. It is apparent that when the draft revision is adopted, the quality and consistency of the licensee's 50.59 review should improve. The staff will follow the progress of the implementation of revision 5 to MOD-013 when it becomes effective.

No violations or deviations were identified.

7. Onsite Review Committee (40700)

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 PNSC meetings on the following dates:

Date	Subject
10/18/89	Modification M-1025 Review
10/25/89	Modification M-1018 Results and Conclusions
10/26/89	Prior to 200 Degrees Fahrenheit Issues
11/02/89	Review of SW and Accumulator Level Issues
11/07/89	Plant Start-up

It was ascertained that provisions of the TS dealing with membership, review process, frequency, and qualifications were satisfied. Previous meeting minutes were reviewed to confirm that decisions and recommendations were accurately reflected in the minutes. The inspectors also followed up on selected previously identified PNSC activities to independently confirm that corrective actions were progressing satisfactorily.

No violations or deviations were identified.

8. Onsite Followup of Written Reports of Nonroutine Events at Power Reactor Facilities (92700)

(CLOSED) P2185-02, Seismic Supports Were Declared Inoperable Based On Requirements Of IEB 79-02. Inspection report 85-06 issued a VIO, 85-06-01, Technical Specification Violation Regarding Piping/Restraint Operability. This item was subsequently inspected and closed by the NRC as documented in IR 87-08. The licensee also issued LER 85-06 concerning this issue.

This LER was subsequently inspected and considered closed as stated in IR 86-13. Based upon the previous closure of 85-06-01 and LER 85-06, the inspectors consider this part 21 item as closed.

(CLOSED) P2186-01, Errors In Reactor Vessel Water Level Instrumentation System. Westinghouse modified the design of the 7300 RVLIS equipment to include an additional amplifier. This change was incorporated into the plant RVLIS by DCN no. 96 to M-526.

(CLOSED) P2186-02, Anacon Chlorine Probes Had Blue Tips Verses White And Were Insensitive To Chlorine. The licensee verified that Anacon chlorine probes were neither in stock nor in use at the site. Anacon was not on the approved vendor list. The E&RC support supervisor also stated that to his knowledge the site had never procured or used Anacon chlorine probes.

No violations or deviations were identified.

9. Exit Interview (30703)

The inspection scope and findings were summarized on December 5, 1989, with those persons indicated in paragraph 1. The inspectors described the areas inspected and discussed in detail the inspection findings listed below and in the summary. The Director of Regulatory Compliance indicated that the reporting requirement was met for the unanalyzed condition in that the report was made within 4 hours of the actual determination of the condition. However, the inspectors considered that sufficient information was available, if only evaluated, to make this determination on or before August 30, 1989 verses the actual reporting date of October 6, 1989. Proprietary information is not contained in this report.

Item Number	Description/Reference Paragraph
89-23-01	IFI - Review Use Of Shielding For Implemenatation Of The ALARA Program (paragraph 4)
89-23-02	IFI - Review Planned PM Schedule For AFW Components (paragraph 4)
89-23-03	IFI - Verify MDAFW Pump Motor Starting Voltages (paragraph 4)
89-23-04	VIO - Failure To Adequately Establish Measures For Suitability Of Processes Essential To Safety-Related Functions As Required By 10 CFR 50 Appendix B Criterion III (paragraphs 4 and 5)

89-23-05	IFI - Review Implementation And Effectiveness Of Cleanliness Controls (paragraph 4)
89-23-06	VIO - Failure To Establish and Implement Procedures as Required By 10 CFR 50 Appendix B Criterion V (paragraphs 4 and 5)
89-23-07	VIO - Failure To Make A 10 CFR 50.72 Report Within 4 Hours As Required (paragraph 5)
89-23-08	URI - EDG Exhaust Lines Not Seismic Qualified (paragraph 5)

10. List of Acronyms and Initialisms

AFW	Auxiliary Feedwater
ALARA	As Low As Reasonably Achievable
ASCO	American Switch Company
AWG	American Wire Gage
CFR	Code of Federal Regulations
CM	Corrective Maintenance
CP&L	Carolina Power & Light
CST	Condensate Storage Tank
CV	Containment Vessel
DBE	Design Basis Earthquake
DCN	Design Change Notice
DS	Dedicated Shutdown
E&RC	Environmental and Radiation Control
EDG	Emergency Diesel Generator
EE	Engineering Evaluation
EOF/TSC	Emergency Operations Facility/Technical Support Center
EOF/TSC/PAP	EOF/TSC/Personnel Access Portal
EQ	Environmental Qualifications
ESF	Engineered Safety Feature
EST	Engineering Surveillance Test
F	Fahrenheit
FCV	Flow Control Valve
Ft	Feet
FT	Flow Transmitter
FSAR	Final Safety Analysis Report
GPM	Gallons Per Minute
HBR	H. B. Robinson
HP	Health Physicist
HVH	Heating Ventilation Handling
Hx	Heat Exchanger
IEB	Inspection and Enforcement Bulletin
IFI	Inspector Followup Item
IR	Inspection Report
ISFI	Independent Spent Fuel Installation
ISI/IST	In-service Inspection/In-service Testing
JCO	Justification For Continued Operation

lbs	Pounds
LCO	Limiting Condition for Operation
LER	Licensee Event Report
LT	Level Transmitter
LTOPP	Low Temperature Over Pressurization Protection
M	Modification
MDAFW	Motor Driven Auxiliary Feed Water
MIC	Microbiological Induced Corrosion
MOD	Modification and Design Control Procedure
mrem/hr	milliroentgen per hour
MST	Maintenance Surveillance Test
NOUE	Notice of Unusual Event
NOV	Notice of Violation
NPSH	Net Positive Suction Head
NRC	Nuclear Regulatory Commission
NSAC	Nuclear Safety Analysis Center
OMM	Operations Management Manual
OST	Operations Surveillance Test
PCV	Pressure Control Valve
PLP	Plant Program
PM	Preventative Maintenance
PNSC	Plant Nuclear Safety Committee
Psig	Pounds per square inch - gage
Psid	Pounds per square inch - differential
PT	Pressure Transmitter
QC	Quality Control
RAB	Reactor Auxiliary Building
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RHR	Residual Heat Removal
RO	Refueling Outage
RVLIS	Reactor Vessel Fluid Level Indicating System
SAR	Safety Analysis Report
SCR	Significant Condition Report
SDAFW	System Driven Auxiliary Feedwater
S/G	Steam Generator
SI	Safety Injection
SP	Special Procedure
SR	Safety Review
SSC	System, Structure or Component
SW	Service Water
TS	Technical Specification
URI	Unresolved Item*
USQ	Unreviewed Safety Question
VIO	Violation
WR	Work Request
WR/JO	Work Request/Job Order

*Unresolved items are matters about which more information is required to determine whether they are acceptable or may involve violations or deviations.