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

Report No.: 50-261/92-02

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: January 14, 1992 - February 7, 1992

Lead Inspector: H. O. Christensen for 2/25/92
L. W. Garner, Senior Resident Inspector Date Signed

Other Inspector: M. D. Hunt, Reactor Engineer
K. R. Jury, Resident Inspector

Approved by: [signature] 2/25/92
for 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, surveillance observation, maintenance observation, self-assessment, and follow-up.

Results:

A violation was identified for failure to adequately perform stroke timing of two containment isolation valves in the manner required by the test procedure (paragraph 3).

A non-cited violation was identified involving the failure of existing procedures to contain adequate instructions to completely test the motor driven auxiliary feedwater subsystem initiation circuitry as required by Technical Specification 4.8.5 and Table 4.8-1 (paragraph 3).

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Preliminary Individual Plant Examination results identified two scenarios as contributing approximately 71 percent of the plant's 1.7E-03 total core damage frequency. The scenarios were a reactor coolant pump seal loss of coolant accident (LOCA) induced by the loss of all component cooling water pumps and an inter-system LOCA initiated by leakage from the primary system into the residual heat removal system (paragraph 2).

Two lower pistons were replaced in the emergency diesel generator A's engine after the vendor notified the licensee via a 10 CFR Part 21 notification that the pistons could contain a manufacturing flaw (paragraph 2).

A Nuclear Assessment Department audit of the Environmental and Radiation Control Unit identified poor radiological work practices and an ineffective self-audit program as potential issues (paragraph 2).

In 1991 the site experienced the lowest person-rem site exposure, smallest quantity of radwaste shipped, and smallest contaminated area since these items have been trended (paragraph 2).

Design Basis Documentation discrepancies were being adequately evaluated and appropriately addressed (paragraph 2).

Addressing corrective actions to preclude recurrence on non-significant Nuclear Engineering Department (NED) Adverse Condition Reports (ACR) was a good practice. However, an 1991 third quarter NED ACR trend report indicated that procedures and methods may not establish conditions that effectively prevent the occurrence of an adverse condition (paragraph 5).

REPORT DETAILS

1. Persons Contacted

*R. Barnett, Manager, Outages and Modifications
*D. Bauer, Regulatory Compliance Coordinator, Regulatory Compliance
*R. Chambers, Plant General Manager, Robinson Nuclear Project
D. Crook, Senior Specialist, Regulatory Compliance
J. Curley, Manager, HBR Engineering Support Section, Nuclear Engineering Department
*C. Dietz, Vice President, Robinson Nuclear Project
*J. Dobbs, Manager, Nuclear Assessment Department Site Unit
R. Femal, Shift Supervisor, Operations
*W. Flanagan, Jr., Manager, Operations
*W. Gainey, Manager, Plant Support
*J. Kloosterman, Manager, Regulatory Compliance
D. Knight, Shift Supervisor, Operations
A. McCauley, Manager - Electrical Systems, Technical Support
R. Moore, Shift Supervisor, Operations
*M. Olinger, Senior Engineer, Nuclear Engineering Department Site Unit
R. Oliver, Manager - Risk Assessment, Nuclear Engineering Department
A. Padgett, Manager, Environmental and Radiation Control
M. Page, Manager, Technical Support
D. Seagle, Shift Supervisor, Operations
M. Scott, Manager - Support Systems, Technical Support
*R. Smith, Manager, Maintenance
W. Stover, Shift Supervisor, Operations
D. Winters, Shift Supervisor, Operations

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

*Attended exit interview on February 12, 1992.

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 except as discussed below, 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.

Preliminary IPE Results

On January 13, 1992, the licensee informed Region II of preliminary IPE results which indicated a total CDF of 1.7E-03 per reactor year. On January 14, the licensee reviewed these preliminary results in more detail with NRR and Region II. Two scenarios, transient-induced LOCA and interfacing system LOCA, account for 76.4 percent of the total CDF.

The major contributor to the transient-induced LOCA CDF was a common mode failure of the CCW pumps which cascades into a loss of RCP seals and a loss of ECCS. A loss of CCW would result in the direct loss of cooling to the RCP thermal barrier coolers and loss of cooling to the charging pump coupling oil coolers. The failure of the charging pumps would result in the loss of RCP seal injection cooling water, i. e., a loss of all cooling to the RCP seals resulting in RCP seal failures. Simultaneously CCW cooling to the SI and RHR seal coolers would also be interrupted causing failure of the ECCS pumps. This slow moving daisy chain of events could be stopped by manually supplying cooling water to the charging pump coupling oil cooler. Temporary instructions were issued on January 15, 1992, to provide procedural steps for connecting fire water from a nearby hose station to the existing couplings on the charging pump coupling oil coolers. The inspectors verified that these temporary instructions were adequate, had been disseminated to the operating shifts, and all the equipment necessary for the temporary cooling connection had been pre-staged.

The primary contributor to the interfacing system LOCA was leakage through the normally closed RHR-750 and 751 valves, the RHR system suction valves from the RCS hot leg, which initiates an unisolable LOCA in the low pressure RHR system. This coolant leakage could also steam bind the ECCS pumps thus preventing the them from performing their safety function. The relative large contributor of this scenario to the CDF resulted from the probability of the valves' double disk gate failures due to not periodically leak testing these valves. Leak testing every refueling interval would result in the calculated interfacing system LOCA contribution to the total CDF being reduced by approximately two orders of magnitude. The licensee plans to leak test these valves during RO 14 which is scheduled to begin on March 27, 1992. The inspectors verified that at the end of the report period the outage schedule was being modified to include RHR-750 and 751 valve leak testing.

Implementation of the above described procedure change and leak testing of the RHR-750 and 751 valves will reduce the total CDF from 1.7E-03 to 4.9E-04 per reactor year. The inspectors will continue to follow-up on the proposed actions as part of the routine inspection program.

B Inverter Malfunction

On January 23, 1992, the B inverter which supplies power to protection and control circuits on instrument busses #3 and #8 was observed to have an erratic output voltage, i. e. indicated voltage swings of approximately 15 volts. Operations declared the inverter inoperable and placed the instrument busses on their alternate supply. The inverter was repaired by replacing the capacitors on one of the inverter's circuit cards. The inverter was then placed back in operation and observed for approximately one hour prior to returning it to service. The inspectors observed the inverter's removal from service, troubleshooting activities, and its return to service. The inspectors observed that operating personnel took prudent precautionary measures such as placing the feedwater control system in manual during the time the instrumentation busses were being transferred.

Part 21 Involving Pistons With Manufacturing Flaws

On January 29, 1992, Coltec Industries Inc. notified the licensee that a Part 21 report was being issued concerning the potential for piston cracking due to a manufacturing defect. The Part 21 Report was apparently only applicable to HBR since vendor records indicated that HBR is the only nuclear plant which has a Fairbanks Morse EDG with rotating pistons. The vendor recently determined that circumferential cracking of diesel generator pistons in and immediately below the piston ring grooves was traceable to one foundry which had supplied castings to Coltec Industries Inc.. The foundry has now corrected the manufacturing process problem. Review of records revealed that four lower pistons supplied for HBR's EDGs in 1989 and 1991 may have come from the suspect castings. The licensee determined that two of the suspect pistons were stored in the warehouse; however, the other two pistons had been installed during the last refueling outage in cylinders #2 and #8 of the A EDG . Further review of records was unable to determine if these pistons had been manufactured from the suspect castings. Thus, on February 3, 1992, the licensee determined that it was prudent to replace the two pistons in the A EDG. The inspectors observed the piston replacement on that day. Visual inspection of the removed pistons, as well as, the two pistons in the warehouse revealed no indications of cracking. The licensee plans to return all four potentially suspect pistons to the vendor for testing.

Containment Isolation Valve Leakage

On January 13, 1992, primary sample valve PS-956F, a normally closed outboard CIV from RCS loops 2 and 3, failed to meet its stroke time test requirement of 10 seconds. This test was being conducted following maintenance on the valve's position indicating lights. The penetration was apparently isolated as required by TS 3.6.3 by closing and removing power to the PS-956E valve, which is the inboard CIV. During repair efforts on January 14, it was determined that valve PS-956E had not fully closed and was leaking at the rate of approximately 830 cc/min with RCS system pressure at 2235 psig. It was also determined that valve PS-956F was leaking and while not measured, its leakage rate appeared to be less than the 830 cc/min through valve PS-956E. These two valves have IVSW injected between them to enhance their containment isolation ability. Since the valve acceptance leak rate per EST-004, Isolation Valve Seal Water, is at 46 psig, the determination of whether or not the valve was exceeding its allowable leak rate based upon the 2235 psig leak rate could not be determined by the SS. As a result the SS initiated an Expert Operability Determination request at 5:00 p.m. on January 14.

On January 15, the actuator spring for PS-956E was adjusted in an effort to reduce the leak rate to allow work on PS-956F. This evolution ultimately reduced the leak rate to 13 cc/min at 2235 psig at 11:00 a.m.. Subsequent attempts to adjust the spring actuator on PS-956F was unsuccessful in further reducing the leak rate. On January 16 calculation 92-C-0001 was generated by Technical Support to correlate the leakage at 2235 psig to 46 psig to determine the significance of the leak and penetration operability. The calculation demonstrated that the 13 cc/min leakage rate was low enough to meet operability requirements; however, it concluded that from 5:00 p.m. on January 14 through 11:00 a.m. on January 15, "the observed leakage was such that the penetration could not be considered operable...". This condition was reported as required by 10 CFR 50.73 (a) (z) (ii) in LER 92-001.

While reviewing the history associated with the valve failures, the inspectors determined that problems with primary sample valve stroking had occurred in December 1991. At that time, it was identified that valves PS-956F and 956G could not be successfully timed from the local panel due to dual indication while closing and the loss of closed indication, respectively. The operators then locally timed the valves' stem travel as remote timing was not possible. Discussions with the SS which managed the evolution revealed that Operations felt local timing of valve stem travel was adequate to comply with their procedural requirements. However, OST-701, Inservice Inspection Valve Test, section 6.3, required that the valve stroke times be measured from control switch actuation (or other actuating signal) to the time the valve reaches the required position as determined by the valve position indicating lights. This methodology is utilized by the licensee to meet ASME Section XI stroke testing requirements and to demonstrate valve operability. By only timing valve stem travel, the licensee did not verify necessary valve disk movement via an approved or accepted methodology. Failure to follow the requirements of OST-701 in testing these sample valves is a violation: Failure To Test Primary Sample Valves In Accordance With Procedures, 92-02-01.

Several concerns resulted from this evolution. The first deals with the SS and operators being unaware what effect the December 1991 testing methodology utilized had on assuring valve operability. The second concern is that while indication of inadequate CIV operation was available (i. e., dual indication for PS-956F), operations did not take the proper actions to verify that the valve could perform its containment isolation function nor was analysis performed to determine the cause of the dual indication (WR initiation was the only action taken). Additionally, after the concern with the testing methodology was identified on ACR 92-009, the timing methodology to be utilized during future valve stroke tests and what actions to take if a test anomaly occurs, was not formally nor uniformly disseminated to all operating shifts. These concerns were discussed with the Operations Manager.

The licensee intends to leave the penetration in the configuration of PS-956E and 956F closed with power removed until RO 14, at which time repairs can be performed. Other sample points are being utilized for monitoring purposes. The inspectors reviewed the licensee's actions, Operability Determination, and LER, and do not have significant technical concerns with the actions taken nor the penetration's current configuration.

E&RC Performance

In January 1992, NAD completed an E&RC assessment. The draft report identified several potential findings, the most significant of which involved field observations of poor radiological work practices and an ineffective self-audit program. At the end of the report period, site management was in the process of determining the scope of the potential issues such that corrective actions could be implemented. Inspection of the licensee's response to the potential issues will be conducted by both the residents and regional specialists as part of the routine inspection program.

Review of E&RC goals for 1991 revealed a continuing improving trend in certain key performance indicators. Of specific note were three items which indicated the best performance on record. These were: 1) the lowest site exposure, 193 person-rem; 2) the smallest quantity of radwaste shipped, 2289 cubic feet; and 3) the smallest amount of contaminated floor space excluding the CV, 1340 square feet.

DBD Status

The inspectors reviewed the status of the DBD effort. The licensee plans to complete the initial DBD writing effort by the spring of this year. The inspectors reviewed DBD discrepancies to verify that they had been evaluated for operability concerns and completed and proposed resolutions were adequate. The inspectors verified that there were not operability concerns in the open DBD discrepancies and that documentation was generally sufficient to verify that the discrepancy was being satisfactorily resolved. In those instances in which the documentation was not sufficient, the inspectors were supplied additional information to demonstrate that the item was being satisfactorily addressed. All DBD discrepancies should be resolved by the end of the year; however, implementation of these resolutions may take several years. As they were identified, the most significant resolutions were being scheduled. The licensee had no plans to incorporate references to the resolutions in their revised DBDs. Since these documents contain valuable information, the inspectors noted that including resolution references would be beneficial. The licensee indicated that they would consider the benefits of providing this information in the DBDs.

M-1087 10 CFR 50.59 Review

The inspectors review the 10 CFR 50.59 safety review package for M-1087, RHR Minimum Flow Recirculation. The inspectors observed that the modification package, which was still in approval routing, indicated that no unreviewed safety question existed. However, on page C42, the safety review package stated that "Avoiding an Unreviewed Safety Question is contingent upon a successful Large Break LOCA calculation by our fuel vendor that accommodates the change in delivered LHSI." The inspectors discussed with the NED HBR Engineering Support Manager the appropriateness of signing the modification approval sheet as no unreviewed safety question existed when the safety analysis contained a contingency. Further discussion with the author of the safety analysis and cognizant engineer revealed that sufficient preliminary results had been obtained from their Fuel Section to justify that an unreviewed safety question did not exist and that the wording in the analysis had been poor. The engineering manager indicated that it was not their practice to sign-off unreviewed safety question determinations which contained contingencies. The inspectors had no further concerns regarding the reviewed documentation.

One violation was 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, 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. Specifically, the inspectors witnessed/reviewed portions of the following test activities:

OST-207 Motor Driven Auxiliary Feedwater Pump Flow Test

OST-401 Emergency Diesels (Slow Speed Start)

Inadequate AFW Test Procedures

On January 28, 1992, the licensee identified that existing surveillance test procedures did not completely verify that MDAFW pump actuation circuitry would function properly. Specifically, testing one set of normally closed contacts and associated interconnecting wiring in each of the MDAFW pump actuation circuits had not been incorporated in the test procedures. These contacts must remain closed and the wiring integrity must be maintained for the MDAFW pumps to automatically start when low-low S/G water level or tripping of the main feedwater pumps is detected. TS 4.8.5 and TS Table 4.8-1 items a and e. requires periodic testing to verify that these conditions will automatically start the MDAFW pumps. In addition, the untested portion of the circuitry also included the ATWS AFW actuation feature; however, testing of this feature is not required by TS. This testing deficiency resulted from inadequate overlap of tests which were performed during plant shutdowns when portions of the actuation circuitry is normally bypassed. Review of plant records revealed that both MDAFW pumps had successfully started on August 30, 1991 when a low-low S/G water level condition had occurred after a condensate pump shaft failure resulted in reduced feedwater flow (see IR 91-19). This event was sufficient to demonstrate within the required TS frequency (refueling interval) that the untested portion of the circuitry would function properly. The licensee plans to implement revised test procedures during RO 14 to fully test the MDAFW actuation circuitry. Similar deficiencies in TS required surveillance activities have been previously identified. As discussed in their responses dated July 9, 1990, and April 3, 1991, to Notice of Violations, a review of the programmatic and procedural adequacy of the TS surveillance program will be completed during 1992. This violation will not be subject to enforcement action because the licensee's efforts in identifying and correcting the violation meet the criteria specified in Section V.G. of the Enforcement Policy. This item is identified as NCV: Failure To Provide Adequate Procedures For Performing TS Required AFW Surveillance Activities, 92-02-02.

One NCV was 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. In particular, the inspectors observed/reviewed the following maintenance activities:

PIC-002	D/P Electronic Transmitter (4-20ma)
CM-610	Emergency Diesel Piston, Ring Main Bearing And Crankshaft Overhaul
CM-615	Emergency Diesel Generator A And B Injector Nozzle And Injector Nozzle Adapter
WR/JO 92-ABBZ1	B Inverter Repair
WR/JO 92-ABJI1	EDG A Cylinder #2 And #8 Lower Piston Replacement

During performance of CM-610 the inspectors observed that the lower piston retainer plate to grove clearance was recorded as 0.00015, whereas the measured value was 0.0015. The correct value was subsequently entered into the procedure. The measured value met the 0.002 maximum allowed clearance.

During performance of CM-615, the inspectors noted that the procedure which had been revised on January 15, 1992 as part of the maintenance procedure upgrade program contained a human factor weakness. The weakness involved placement of acceptance criteria in caution notes without repeating the information in the subsequent procedure steps or on data sheets. Examples included the caution notes on page 16 of the procedure which provided a +/- 25 degree acceptance criteria for positioning the injector nozzle adapter collar studs and a requirement that the injector nozzle holder be positioned with the word TOP in the up position. This item was discussed with the appropriate maintenance personnel.

No violations or deviations were identified.

5. Self-Assessment (45000)

The inspectors reviewed the corporate NED corrective action program as it pertained to HBR. The inspection consisted of a review of all the 1991 ACRs associated with the HBR engineering project. The inspectors verified that the ACRs had been properly classified as significant or non-significant, were being addressed in a timely manner, and corrective actions appeared to be appropriate. The inspectors noted that most non-significant ACRs, through not required by procedures, had corrective actions specified to preclude recurrence. This was considered to be a good practice. However, the 1991 Third Quarter Trend Report For Adverse Condition Reports Generated By NED, the latest report available involving all three nuclear sites, identified a management attention item, i. e., "Evaluation of causal factors indicates that NED procedures and methods may not establish conditions that effectively prevent the occurrence of an adverse condition." In response to this potential item, a committee had been formed to perform a detailed review of the ACRs' casual factors and determine what corrective actions, if any, are required. The inspectors will follow-up on this item as part of the routine inspection program.

No violations or deviations were identified.

6. Follow-up (92700, 92701, 92702)

(Open) URI 50-261/89-26-02, Cable Submergence Qualification. The licensee discovered in 1989 that several EQ cables would become submerged, but felt that if the cables met the LOCA submergence testing they were qualified. The NRC staff did not accept this submergence testing as meeting the intent of NUREG-0588, paragraph 2.2(5). The licensee then chose to await further industry testing. In May of 1991, NUREG/CR-5655 was published which documented the testing of 12 different cable products.

The licensee's list of cables that would be submerged contained 3 brand/types that were not identical or similar to those tested. These were identified during this inspection. Later, discussions with the licensee were held and the licensee is to submit additional data to the NRC for acceptability studies. This item remains open.

(Closed) IFI 90-19-01, Evaluate Licensee's Inspection Program For Openings In Reinforced Concrete Walls With Respect To Appendix "R" and IEB 80-11 Reinforcements. The inspector reviewed Operations Surveillance Test Procedure OST-623, Fire Barrier Penetration Seal Inspection (Refueling) which was completed January 26, 1991. This inspection was performed over a period of time from April until December of 1991, and identified 50 fire barrier penetration which by procedure were declared inoperable. The inspector reviewed the Final Report for Supplemental Response to IE Bulletin 80-11, dated March 1991 which documented the inspections and verification or justification for the fill-in materials for the block out configuration of the 50 questionable fire barrier penetrations originally identified by OST-623. The fill-in material for these penetration was determined and evaluated. External loading affects were considered and evaluated for the walls containing these penetrations. Additional rework was required for five HVAC penetration which contained only one row of brick and twenty-four attachments. After correction of these 5 penetrations the analysis and evaluation confirmed that existing brick in-fill penetration conform to the requirement of IEB 80-11.

Engineering evaluations 90-104, Generic Evaluation of HVAC Fire Damper and Fire Door Installation Discrepancies and 90-129, Evaluation of Fire Barrier Penetration Seals in Fire Zone 27 were reviewed by the inspector. The purpose of these evaluations was to determine the adequacy of the existing fire dampers and doors and the adequacy of the fire barrier penetrations seals in fire zone 27. There were no concerns identified by the inspector. This item is closed.

No violations or deviations were identified.

7. Exit Interview (30703)

The inspection scope and findings were summarized on February 12, 1992, 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. Dissenting comments were not received from the licensee. The licensee did not identify as proprietary any of the materials provided to or reviewed by the inspectors during this inspection.

Item Number	Description/Reference Paragraph
92-02-01	VIO - Failure To Test Primary Sample Valves In Accordance With Procedures (paragraph 2).
92-02-02	NCV - Failure To Provide Adequate Procedures For Performing TS Required AFW Surveillance Activities (paragraph 3).

8. List of Acronyms and Initialisms

a.m.	Ante Meridiem
ACR	Adverse Condition Report
AFW	Auxiliary Feedwater
ASME	American Society of Mechanical Engineers
ATWS	Anticipated Transient Without Scram
cc/min	cubic centimeters/minute
CCW	Component Cooling Water
CDF	Core Damage Frequency
CFR	Code of Federal Regulations
CIV	Containment Isolation Valve
CM	Corrective Maintenance

CV	Containment Vessel
DBD	Design Basis Documentation
D/P	Differential Pressure
E&RC	Environmental and Radiation Control
ECCS	Emergency Core Cooling System
EDG	Emergency Diesel Generator
EQ	Environmental Qualifications
EST	Engineering Surveillance Test
HBR	H. B. Robinson
HVAC	Heating Ventilation Air Conditioning
i.e.	That is
IE	Inspection and Enforcement
IEB	Inspection and Enforcement Bulletin
IFI	Inspector Follow-up Item
IR	Inspection Report
IPE	Individual Plant Evaluation
IVSW	Isolation Valve Seal Water
LCO	Limiting Condition for Operation
LER	Licensee Event Report
LOCA	Loss of Coolant Accident
LHSI	Low Head Safety Injection
M	Modification
MDAFW	Motor Driven Auxiliary Feed Water
NAD	Nuclear Assessment Department
NCV	Non-cited Violation
NED	Nuclear Engineering Department
NRR	Nuclear Reactor Regulation
OST	Operations Surveillance Test
p.m.	Post Meridiem
PIC	Process Instrument Calibration
PS	Primary Sample
Psig	Pounds per square inch - gage
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
rem	Roentgen Equivalent Man
RHR	Residual Heat Removal
RO	Refueling Outage
S/G	Steam Generator
SI	Safety Injection
SS	Shift Supervisor
TS	Technical Specification
URI	Unresolved Item*
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.