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

Report No.: 50-302/94-14

Licensee: Florida Power Corporation
3201 34th Street, South
St. Petersburg, FL 33733

Docket No.: 50-302 License No.: DPR-72

Facility Name: Crystal River 3

Inspection Conducted: May 7 through June 10, 1994

Inspector: R. Jenkins for 7/5/94
R. Butcher, Senior Resident Inspector Date Signed

Inspector: R. Jenkins for 7/5/94
T. Cooper, Resident Inspector Date Signed

Approved by: K. Landis 6/30/94
K. Landis, Section Chief Date Signed
Division of Reactor Projects

SUMMARY

Scope:

This routine inspection was conducted by the resident inspectors in the areas of plant operations, radiological controls, security, surveillance observations, maintenance observations, licensee event reports, self assessment, refueling activities, on site engineering, fire protection, and licensee action on previous inspection items. Numerous facility tours were conducted and facility operations observed. Backshift inspections were conducted on May 15, 24, 25, 26, 28, 29, 30, 31, June 1, and 2.

Results:

Within the scope of this inspection, the inspectors determined that the licensee continued to demonstrate satisfactory performance to ensure safe plant operations. In addition, the licensee, through self assessment, took prompt action to correct the following non-cited violation:

Non-cited Violation 50-302/94-14-01, Failure to Perform an Adequate Design Review Resulting in Potentially Inoperable Motor Operated Valves Under Degraded Voltage Conditions. (paragraph 12.b)

During this inspection period, the inspectors had comments in the following Systematic Assessment of Licensee Performance functional areas:

Plant Operations:

The Engineered Safeguards Features response testing was well planned and executed. Pre-job briefings were good, and effective communications were maintained. (paragraph 4.a)

The failure to keep the NRC informed as to the revised status of the auxiliary feedwater system was considered a weakness. (paragraph 3.e)

Maintenance:

The clean-up and inspection of the Reactor Building sump was thorough and well handled. However, the failure to maintain the sump area free of foreign material was a weakness. The use of clear plastic in the Reactor Building was a hazard due to the inability to see clear plastic if submerged in water. (paragraph 3.c)

Engineering:

Systems Engineering, with the vendor's assistance, appears to have resolved a long term emergency diesel generator bearing problem. (paragraph 8.a)

Failure to prevent a procedure revision incorporating system modifications from being issued and used before the modifications were performed was considered a weakness in the Modification Approval Record implementation process. (paragraph 4.c)

Plant Support: (Radiation Controls, Emergency Preparedness, Security, Chemistry, Fire Protection, Fitness for Duty, and Housekeeping Controls)

Allowing a fire watch to remain in a designated high radiation area while no work requiring a fire watch was being accomplished was considered a weakness. (paragraph 6.a)

The following general comments were also noted:

- The large number of recommendations in audit report 94-02-OPS reflects the need for a thorough licensee review to enhance the conduct of operations activities reflected in the audit. (paragraph 7.a)

- The use of highly qualified/experienced personnel from other utilities or industry organizations for Quality Assurance Audits was considered a strength. (paragraph 7.a)

- The NRC has reviewed the issue of the potential impact of non-safety related equipment (units 1 and 2 smoke stacks) upon important equipment during an external event and has concurred with the licensee's conclusion that the chimneys are not a significant structural risk to safety related structures at Crystal River 3. (paragraph 12.a)

The inspectors reviewed the following outstanding items:

Item Number	Status	Description and Reference
LER 50-302/93-008 and LER 50-302/93-008-01	Closed	Due to Lack of Engineering Review, Motor Operated Valves With Brakes Could Fail to Perform Their Safety Function Under Degraded Voltage Conditions. (paragraph 10.a)
URI 50-302/94-09-01	Closed	Insufficient Voltage to Operate Main Feedwater Isolation Valve FWV-28. (paragraph 12.b)
10 CFR Part 21	Closed	Air Start Distributor Cam Problems in Fairbanks-Morse Diesel Generators (paragraph 10.b)

REPORT DETAILS

1. Persons Contacted

Licensee Employees

*W. Bandhauer, Nuclear Shift Manager
*P. Beard, Senior Vice President Nuclear Operations
*G. Boldt, Vice President Nuclear Production
*W. Brewer, Supervisor, Nuclear Plant Technical Support
J. Campbell, Nuclear Shift Manager
*R. Davis, Manager, Nuclear Plant Maintenance
*F. Fusick, Manager, Site Nuclear Engineering Services
*S. Garry, Corporate Health Physicist
*G. Halnon, Manager, Nuclear Plant Operations
*B. Hickle, Director, Nuclear Plant Operations
*G. Longhouser, Nuclear Security Superintendent
W. Marshall, Nuclear Shift Manager
*P. McKee, Director, Quality Programs
*R. McLaughlin, Nuclear Regulatory Specialist
*A. Miller, Senior Nuclear Scheduling Coordinator
B. Moore, Manager, Nuclear Integrated Scheduling
W. Neuman, Supervisor, Inservice Inspection
*J. Roberts, Assistant Nuclear Chemical & Radiation Protection Supervisor
*S. Robinson, Manager, Nuclear Quality Assessments
W. Rossfeld, Manager, Site Nuclear Services
W. Stephenson, Nuclear Shift Manager
F. Sullivan, Nuclear Shift Manager
*J. Terry, Manager, Nuclear Plant System Engineering
*R. Widell, Director, Nuclear Operations Site Support
G. Wilson, Nuclear Shift Manager
K. Wilson, Manager, Nuclear Licensing

Other licensee employees contacted included office, operations, engineering, maintenance, chemistry/radiation, and corporate personnel.

NRC Resident Inspectors

*R. Butcher, Senior Resident Inspector
*T. Cooper, Resident Inspector

*Attended exit interview

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

2. Plant Status and Activities

a. Plant Status

At the beginning of this report period, the plant was in a defueled condition. Fuel off-loading had been completed on April 26 at 12:00 p.m. The following evolutions occurred during this report period:

- Fuel loading operations commenced on May 11, 1994, at 4:50 a.m. and the unit entered Mode 6 at that time.
- Fuel reload operations completed on May 14, 1994, at 6:20 p.m.
- Reactor vessel core load verification was completed on May 15, 1994.
- The reactor vessel head was set in place on May 16, 1994, at 10:21 p.m.
- On May 18, 1994, at 3:00 p.m. the unit entered Mode 5.
- On May 29, 1994, at 7:00 a.m. the unit entered Mode 4.
- On May 29, 1994, at 11:00 p.m. the unit entered Mode 3.
- On May 30, 1994, at 11:20 p.m. the RCS reached NOP and NOT. (approximately 2155 psig and 532 degrees F)
- On June 1, 1994, at 8:40 p.m. the unit entered Mode 2.
- On June 1, 1994, at 10:39 p.m. the unit was made critical. (2329 ppm boron and Group 7 at 85%)
- On June 2, 1994, at 6:23 p.m. the unit entered Mode 1.
- On June 3, 1994, at 3:33 a.m. the output breakers were closed and the unit placed on line.
- On June 6, 1994, at 9:50 p.m. the unit reached 100% power.

b. NRC Activity

On May 10 and 11, 1994, L. Raghavan, Project Manager, NRC Office of Nuclear Reactor Regulation, visited the site. His activities included attending an NGRC meeting and discussing outstanding licensing issues with the licensee and resident inspectors.

On May 16-20, 1994, Fred Wright, Senior Radiation Specialist, and Bryan Parker, Radiation Specialist, NRC Region II, were on site to evaluate the licensee's implementation of the revised 10 CFR

Part 20. The results of that inspection are documented in IR 50-302/94-13.

On May 16-20, 1994, Steven Rudisail, Reactor Inspector, NRC Region II, was on site to inspect the status of the cable separation issues. The results of that inspection are documented in IR 50-302/94-12.

3. Plant Operations (71707 & 93702)

Throughout the inspection period, facility tours were conducted to observe operations and maintenance activities in progress. The tours included entries into the protected areas and the radiologically controlled areas of the plant. During these inspections, discussions were held with operators, health physics and instrument and controls technicians, mechanics, security personnel, engineers, supervisors, and plant management. Some operations and maintenance activity observations were conducted during backshifts. Licensee meetings were attended by the inspector to observe planning and management activities. The inspections confirmed FPC's compliance with 10 CFR, Technical Specifications, License Conditions, and Administrative Procedures.

a. The licensee received industry reports that both Three Mile Island and Oconee Nuclear Plants were having problems meeting their control rod drop time testing requirements. The slow down in the drop times was attributed by BWNS to ball check valves stuck in the closed position. As a result, the licensee pulled one CRDM during the refueling outage to inspect for crud build-up on the ball check valves.

The CRDMs convert the rotary motion of the nut assembly to linear travel of the lead screw and control rod. The CRDM positions the control rod within the reactor core and indicates the location of the control rod with respect to the reactor core. The speed at which the control rod is inserted or withdrawn from the core is fixed and is consistent with design reactivity change requirements during reactor operation. For conditions that require a rapid shutdown of the reactor, the drive mechanism releases the CRA which drops by gravity into the core. The reactivity is reduced during such a rod insertion at a rate sufficient to control the core under any operating transient or analyzed accident condition.

The thermal barrier is a flow restriction device which acts to insulate the CRDM from the reactor vessel. The thermal barrier is in the lower flange of the motor tube. During operation the close tolerance of the thermal barrier around the lead screw restricts water passage from the RCS into the rotor and motor tube area. The heat load on the stator cooling system is thereby reduced. Sufficient clearance is provided to accommodate the volume exchange required for normal rod movement. During a rod trip a rapid volume exchange is required to accommodate the speed at which the lead screw drops. This increased flow rate is provided for by four ball check valves. During normal operation the ports are maintained sealed by the weight of the balls. During a rod drop, the differential pressure generated is sufficient to lift the balls and open the ports.

The licensee's inspection revealed that three of the four ball check valves on the CRDM being examined were obstructed by crud build-up. One of the ball check valves was only slightly impeded, and was easily freed by lightly tapping the ball. Therefore, the licensee and BWNS representatives are of the opinion that this valve would have opened during a reactor trip condition. The other two ball check valves were harder to free and it was doubtful that they would have been freed during a reactor trip condition. BWNS has performed analysis on the phenomenon observed at two other licensees and has concluded that for a CRDM to insert unimpeded during a rod drop, only one of the ball check valves is required to be functional. A method to verify that at least one ball check valve is functioning is to trend rod drop times. As long as one valve is functioning, the rod drop times should be unaffected. The licensee has verified that none of the CRDMs in the core have had any degradation in rod drop times, up until the last performance of the rod drop test.

During start-up from the current outage, rod drop testing was conducted to meet TS SR 3.1.4.3 requirements. TS SR 3.1.4.1 requires, prior to reactor criticality after each removal of the RV head, that rod drop times be verified for each control rod from the fully withdrawn position. The inspectors observed rod drop testing and no significant decrease in drop times were observed. The licensee is working with the other affected utilities and B&W to develop a corrective/preventive action plan. (See paragraph 4.b also.) The inspectors will follow the licensee's actions in this area.

b. The improved TS for CR-3 were implemented on March 12, 1994. TS 3.3.9, Source Range Neutron Flux, requires in Modes 2 (at specified power levels), 3, 4, and 5 that two source range neutron flux channels be operable. The TS basis for TS 3.3.9 states that although not normally relied upon to perform the source range neutron flux monitoring function, the Gamma-Metrics post-accident monitoring instrumentation wide range neutron flux (NI-14 and NI-15) have been shown to be functionally equivalent to NI-1 and NI-2 and may be used to comply with this LCO. During the current refueling outage the licensee incorporated a MAR that installed additional neutron indication on the MCB derived from NI-14 and NI-15 which incorporates the equivalent range of the NI-1 and NI-2 indications (from 0.1 to 10E6 cps). The existing NI-14 and NI-15 neutron flux indicators on the main control board ranged from 10E-8 to 100% reactor power and was originally installed to provide for post accident monitoring. Neutron detectors NI-1 and NI-15 are located adjacent to each other at approximately 180 degrees from NI-2 and NI-14. Not specifically addressed by the TS or the

TS basis was the question of which source range detectors could be used in combination to satisfy the TS requirement for two operable source range neutron flux channels.

NRC stated that the two operable source range detectors referenced in the TS must be on opposite sides of the reactor core to ensure proper monitoring of the reactor core during low power operation. Discussions with the licensee revealed that they had already addressed this issue and their procedures complied with the NRC position.

The inspectors reviewed OP-202, Plant Heatup, Revision 90, and verified that the procedure required NI-1 or NI-15 and NI-2 or NI-14 to satisfy the requirement for two operable source range neutron monitors. The inspectors and the licensee monitored the performance of NI-14 and NI-15 during the start-up and could not correlate their indications to that observed on NI-1 and NI-2. Therefore, the licensee relied on NI-1 and NI-2 during the startup. The licensee has initiated an investigation into the problems identified with NI-14 and NI-15 indications. The inspectors will follow the licensee's actions.

c. On April 11, 1994, shortly after reactor shutdown and opening of the RB, the inspectors visually examined the RB sump and the sump screen assembly. This inspection was documented in IR 50-302/94-09. In preparation for closing out the RB for this refueling outage, on May 21, 1994, the licensee pumped down the RB sump in order to clean the sump and verify the material condition of the sump and screen was satisfactory for RB closeout. This inspection was accomplished per procedure SP-175, Containment Sump Level and Flood Monitoring System Calibration, Section 4.5, Containment Emergency Sump Inspection. On May 23, 1994, the inspectors again conducted a tour of the reactor building to determine the condition of the reactor building sump. The licensee had cleaned the sump on May 20, 1994, in preparation for the end of the outage. The inspectors noted debris floating in the sump. The debris appeared to be tags, which had fallen off of bags of material stored on top of the grating. Along with the bags, other materials, such as clear plastic, tools, ropes, and protective clothing were also stacked on top of the reactor building sump grating.

The inspectors reported this material condition to the shift manager and the reactor building coordinator. When questioned as to what other materials might have fallen in the sump, the licensee scheduled to have the sump drained and recleaned. Prior to this being accomplished, licensee management was touring the reactor building and noted material being dropped on the grating from work in progress above. The concerns of both the inspectors and licensee management resulted in the licensee declaring the reactor building sump area a foreign material exclusion zone.

The inspectors toured the reactor building on May 27, 1994, to perform a final pre-startup cleanliness inspection. At that time, it was noted that the sump was clean, with no signs of debris and in good condition.

During this reactor building inspection, it was noted that the lids on the TSP containers were loose. Additionally, large paint chips were noted on the floor beneath the RCDT. The licensee installed bands on the TSP container lids and removed the paint chips from beneath the RCDT.

The inspection and cleanup of the RB sump was thorough and well handled. However, the failure to maintain the RB sump area as a clean area was considered a weakness. Also, the use of clear plastic material in sensitive areas such as the RB sump is a hazard since it is almost impossible to detect if submerged in water.

d. On April 26, 1994, the NRC issued a CAL regarding the planned inspection program for the OTSG tubes. The CAL documented the planned inspection details regarding criteria for addressing low signal to noise ratio indications that are not addressed under current TS criteria. NRC Region II inspectors examined the licensee's test program and results and this inspection is documented in IR 50-302/94-11. By letter dated May 25, 1994, the licensee submitted the summary of the results of the refuel 9 OTSG tube inspections and repairs. By letter dated May 27, 1994, the NRC advised the licensee that they had satisfied the CAL for entry into Mode 4 and higher.

e. On May 31, 1994, with the unit in Mode 3 and preparations underway for entry into Mode 2 the following day, the inspectors requested to see the licensee's documentation where they had verified the operability of FWP-7. No testing had been accomplished to verify FWP-7 was operable and the status of FWP-7 was unknown at that time. Additionally, no procedure was available to periodically test FWP-7.

By letter dated April 29, 1988, the NRC had notified the licensee that based upon previous NRC/licensee meetings and correspondence where the licensee had committed to install an additional means of secondary side decay heat removal, that upon implementation of that commitment, the CR-3 AFW system and secondary side decay heat removal capability would meet the SRP criterion. By letter dated May 31, 1990, the licensee submitted the conceptual design information for the AFW addition (FWP-7 and related piping) for NRC review. This letter also committed that the installation of FWP-7 would be completed during Refuel 8. Attachment 1 of that letter also stated that provisions would be made for periodic testing of the AFW pump.

By letter dated September 19, 1990, the NRC notified the licensee that the NRC had completed its review of the conceptual design information submitted in the letter dated May 31, 1990. However, the NRC requested that it be kept informed of significant design features, status, and other developments of interest as the modification progressed toward the licensee's full implementation during Refuel 8.

By letter dated June 25, 1992, the licensee notified the NRC that the installation of FWP-7 had been completed. However, due to excessive motor vibrations, additional corrective actions and functional testing was required and the final complete flush of the system would occur during the next scheduled outage (mid-cycle outage 9M that occurred during the time frame of February -April, 1993). Based on the information submitted, the licensee requested that GSI-124, Auxiliary Feedwater System Reliability, be closed and stated that the NRC staff would be kept informed regarding the schedule for motor reinstallation and the systems availability for use.

On June 1, 1994, in response to the inspectors query as to the status of the FWP-7, the licensee performed a test of FWP-7 per OP-605, Feedwater System, Section 4.20, Operation of FWP-7 in Recirculation to CDT-1. This test was conducted to demonstrate that the FWP-7 was functional in the recirculation mode and to obtain base line vibration data. The original MAR for the installation of FWP-7 (MAR 88-07-05-01, Auxiliary Feedwater Pump FWP-7) has never been closed out due to the high vibration levels experienced by FWP-7 during MAR functional testing and Engineering recommended not operating FWP-7 except in an emergency or to obtain test data. FCN 17 to MAR 88-07-05-01 has been prepared to modify the FWP-7 base in order to reduce vibration levels. On June 6, 1994, the inspectors witnessed the operation of FWP-7 to validate the new surveillance procedure SP-348A, Auxiliary Feedwater Pump (FWP-7) Testing. Performed as a part of SP-348A was the no flow stroke test of FWV-216 and FWV-217, FWP-7 flow control valves to the OTSG A and OTSG B respectively. During this test, FWV-217 failed to stroke. The licensee initiated a WR (320081) to correct this problem. No date had been set to incorporate the FCN as of the time of this inspection. By letter dated June 6, 1994, the NRC requested the licensee provide a schedule for completing the planned modification to FWP-7, functional testing, and final system turnover for operation.

The failure to keep the NRC informed as to the revised status of FWP-7, as was previously requested in NRC/licensee correspondence, was considered a weakness.

f. The inspectors monitored the reactor startup on June 1, 1994. The startup was accomplished using procedures OP-210, Reactor Startup, and PT-110, Controlling Procedure for Zero Power Physics Testing. The approach to criticality was performed in a controlled and professional manner with good communications between involved personnel. The Operations Manager was present in the control room to provide Management oversight of the startup. Reactor criticality was successfully accomplished at 10:40 p.m. on June 1, 1994.

Violations or deviations were not identified.

4. Surveillance Observations (61726)

The inspectors observed TS required surveillance testing and verified that the test procedures conformed to the requirements of the TSs; testing was performed in accordance with adequate procedures; test instrumentation was calibrated; limiting conditions for operation were met; test results met acceptance criteria requirements and were reviewed by personnel other than the individual directing the test; deficiencies were identified, as appropriate, and were properly reviewed and resolved by management personnel; and system restoration was adequate. For completed tests, the inspectors verified testing frequencies were met and tests were performed by qualified individuals.

The inspectors witnessed/reviewed portions of the following test activities:

- SP-102, Control Rod Drop Time Tests;
- SP-121, ATWS System Calibration;
- SP-348A, Auxiliary Feedwater Pump (FWP-7) Test;
- SP-354A, Monthly Functional Test of the Emergency Diesel Generator EGDG-1A;
- SP-417, Refueling Interval Integrated Plant Response to an Engineered Safeguards Actuation;
- SP-435, Valve Testing During Cold Shutdown; and
- 93-05-11-01, TP:1 and TP:2, Revise MFWI Logic (Post MAR Testing).

The following items were considered noteworthy:

a. The inspectors witnessed the performance of the ESF response test for both the A and B trains, per SP-417. The test was coordinated by the Operations department, with assistance from systems engineering and electrical shop personnel.

   The same personnel were used to perform testing on both trains. Pre-job briefings were held prior to each train being performed to clarify expectations and task assignments. Continuous communications were established by way of the hand held radios, with non-job related communications on that channel prohibited.

The inspectors noted that the tests were well planned and executed. Each train test was performed without problems and with the expected results.

b. The inspectors witnessed the performance of the control rod drop time tests per SP-102. This test was performed to satisfy TS SR 3.1.4.3 requirements and to determine if sticking of the ball check valves was occurring. All groups of CRDMs, with the exception of Group 7, showed a slight increase in rod drop times (approximately 0.03 seconds), even though all rods were still below the TS requirement of 1.66 seconds to 75% insertion. The increase may or may not be significant; analysis is still being performed. The licensee plans to develop an action plan after BWNS develops recommendations based on analysis of CRDM behavior at all of the licensees with type A CRDMs.

   The inspectors will review the action plan, after it is developed. No problems were noted with the test. See paragraph 3.a for further discussion of this issue.

c. While observing the performance of SP-121, ATWS System Calibration, Revision 4, the inspector noted that the procedure no longer performed as expected when the technician was performing step 4.2.9. The technician stopped the procedure, verified all of the test connections and attempted the step again. The step did not perform as expected a second time. The technician halted the procedure and notified his supervision. The supervisors came to the job site and verified the steps of the procedures up to the point where the procedure quit performing as expected. Once again, the technician unsuccessfully attempted to perform the procedure. The supervisors decided to stop the performance of the procedure until the discrepancy was resolved.

   Investigation by the licensee determined that Revision 3 to SP-121 could have been performed successfully. Revision 4 to the procedure incorporated changes installed by MAR 93-06-16-01, on NI-14 and NI-15 and AMSAC setpoints. The MAR was scheduled to be installed during the ongoing refueling outage, but had not yet been installed. The procedure revision was issued, with the changes, but nothing was present to require the I&C department to verify installation of the MAR prior to attempting to perform the procedure, other than a note on the scheduling work sheet that there was an outstanding MAR on the system.

   When questioned, the licensee stated that there was a race between installing the MAR and issuing the procedure revision and the procedure revision was issued first. Failure to prevent a procedure revision incorporating system modifications from being issued and used before the modifications were performed was considered a weakness in the Modification Approval Record implementation process.

The inspectors determined that the above testing activities were performed in a satisfactory manner and met the requirements of the TSs.

Violations or deviations were not identified.

5. Maintenance Observations (62703)

Station maintenance activities of safety-related systems and components were observed and reviewed to ascertain they were conducted in accordance with approved procedures, regulatory guides, industry codes and standards, and in conformance with the TSs.

The following items were considered during this review, as appropriate: LCOs were met while components or systems were removed from service; approvals were obtained prior to initiating work; activities were accomplished using approved procedures and were inspected as applicable; procedures used were adequate to control the activity; troubleshooting activities were controlled and repair records accurately reflected the maintenance performed; functional testing and/or calibrations were performed prior to returning components or systems to service; QC records were maintained; activities were accomplished by qualified personnel; parts and materials used were properly certified; radiological controls were properly implemented; QC hold points were established and observed where required; fire prevention controls were implemented; outside contractor force activities were controlled in accordance with the approved QA program; and housekeeping was actively pursued.

The inspectors witnessed/reviewed portions of the following maintenance activities in progress:

- WR 0312314, Replace B 480V ES Transformer per MAR 90-12-04-05;
- WR 0312930, Remove Firedoors to Allow Replacement of B ES Transformer;
- WR 0318610, Perform ECAD Testing of NI-5 Cables;
- WR 0319381, Replace Generator Bearing and Inner Closure on B EGDG; and
- WR 0319451, Troubleshoot and Repair NI-1 Start-up Range Neutron Monitor.

The following item was considered noteworthy:

Following the five year maintenance on EGDG-1A, during the current refueling outage, the diesel was tested and declared operable on May 23, 1994. On May 24, 1994, an ANO noted a slow leak of water on the jacket cooling water pump, DJP-1. Engineering evaluated and determined that the leak did not affect the operability of the A EGDG and scheduled seal replacement during a system outage scheduled for July, 1994. Later that day, however, the leakage increased and the decision was made to replace the seals. When mechanical maintenance disassembled the pump to inspect the seals, it was noted that the impeller was cracked and that indications existed on the shaft. Instead of just replacing the seals, the pump was replaced with a rebuilt pump.

On May 27, 1994, following the maintenance on the pump, SP-354A, Monthly Functional Test of the Emergency Diesel Generator EGDG-1A, was started as the functional test for the work. The test was run for a short period of time and then halted due to the continuing leak from DJP-1. Maintenance personnel replaced the seal on the pump, but it continued to leak. Upon closer examination, it was determined that the fixed face and the rotating face were not aligned properly during the pump rebuild. Aligning the two faces stopped the leak.

After the seal was replaced on DJP-1, the decision was made to defer the EGDG-1A system outage from July 1994 until later in the year, since seal replacement was the major job for the outage. This seal replacement has been listed as part of the corrective action for LER 92-002 since 1992. The licensee waited until a leak developed to schedule replacement.

The inspectors reviewed licensee procedure MP-499, Emergency Diesel Generator Engine Inspection/Maintenance and verified that seal replacement on the ancillary pumps had been included on a routine basis. Mechanical seal replacement for the jacket cooling water pumps has been included in the 10 year maintenance schedule. The seal for DJP-2, the jacket cooling water pump on EGDG-1B, was replaced under WR 295446 in March, 1992. The evaluation performed as one of the corrective actions for the LER, on the other pumps, noted when the seals had actually been replaced or that they were due. The jacket cooling water pump for EGDG-1A, DJP-1, was evaluated as due at that time. The pumps that were evaluated as due have not been scheduled for seal replacement. However, they are being worked as needed, and then are being included in the replacement schedule, per MP-499. During the current refueling outage, even though major maintenance was scheduled on the 1A EGDG, the mechanical seals were not scheduled to be changed. After the maintenance was complete, a leak developed on DJP-1 and the pump and seal were replaced.

Other ancillary systems have been evaluated and their seals have been replaced or the seal replacement was labeled as due in 1992. All of these pumps have been included in MP-499 for routine replacement, but none of these pumps have been scheduled to have their seals replaced. This issue was addressed as corrective action for LER 92-002 which is still open. The inspectors will follow-up this issue under LER 92-002.

The licensee's engineering judgement indicated that the 1A EGDG could have performed its design function of operating for seven days with the leaking seal on DJP-1. Also, the licensee's engineering judgement indicated that an EGDG could perform its design function with a jacket water cooling pump seal or any of the ancillary system seals leaking (if they should begin to leak) due to age/wear. The inspectors concluded that there was no EGDG operability concern with the licensee leaving the seals installed until they start to leak.

For those maintenance activities observed, the inspectors determined that the activities were conducted in a satisfactory manner and that the work was properly performed in accordance with approved maintenance work orders.

Violations or deviations were not identified.

6. Plant Support (71750)

a. Health Physics Program

Radiation protection control activities were observed to verify that these activities were in conformance with the facility policies and procedures, and in compliance with regulatory requirements. These observations included:

- Entry to and exit from contaminated areas, including step-off pad conditions and disposal of contaminated clothing;

- Area postings and controls;

- Work activity within radiation, high radiation, and contaminated areas;

- RCA exiting practices;

- Proper wearing of personnel monitoring equipment, protective clothing, and respiratory equipment; and

- NRC form 3 and NOVs involving radiological working conditions were posted in accordance with 10 CFR 19.11.

Effluent and environmental monitoring was observed to determine that radiation and meteorological recorders and indicators were operable with no unexplained abnormal traces evident. Other observations verified that control room toxic monitors were operable and that plant chemistry was within TS and procedural limits.

During a reactor building tour on May 23, 1994, the inspector observed a person in a sitting position overhead, leaning up against a vertical run of piping, approximately ten feet from where maintenance was taking place on MUV-55. Upon leaving the reactor building, the inspector notified the health physics technician at the entrance to containment. The HP notified the roving HP technician in the building who went and relocated the person from his position on the piping, which was in a designated high radiation area.

Maintenance management was notified of the situation and they performed a prompt investigation. The person observed in the reactor building was the fire watch for the maintenance being performed. At the time the inspector observed him, no grinding or welding was taking place on the job, so an active fire watch was not necessary. However, since welding and grinding had been taking place and since more was expected to be done prior to the completion of the job, the fire watch was needed to be available.

The inspector observed that the fire watch could have been available and still have been outside of the high radiation area. The fire watch agreed and stated that he knew that a high radiation area was not the proper place to be when no work was being performed. Maintenance management counseled the employee as to the proper procedures to be followed to meet ALARA criteria in the work environment.

This incident was considered a weakness in the application of ALARA principles during a maintenance activity.

b. Security Control

In the course of the monthly activities, the inspector included a review of the licensee's physical security program. The performance of various shifts of the security force was observed in the conduct of daily activities to include: protected and vital areas access controls; searching of personnel, packages, and vehicles; badge issuance and retrieval; escorting of visitors; patrols; and compensatory posts. In addition, the inspector observed the operational status of protected area lighting, protected and vital areas barrier integrity, and the security organization interface with operations and maintenance. No performance discrepancies were identified by the inspectors.

c. Fire Protection

Fire protection activities, staffing, and equipment were observed to verify that fire brigade staffing was appropriate and that fire alarms, extinguishing equipment, actuating controls, fire fighting equipment, emergency equipment, and fire barriers were operable. See paragraph 13 also.

Violations or deviations were not identified.

7. Self Assessment (40500)

a. The licensee routinely performs Quality Program audits of plant activities as required under its QA program or as requested by management. To assess the effectiveness of these licensee audits, the inspectors examined the status, scope, findings and recommendations of the following audit reports:

REPORT NO.	TITLE	NO. OF FINDINGS	NO. OF RECOMMENDATIONS
94-02-OPS	Nuclear Plant Operations	0	66
94-03-SSUP	Nuclear Operations Site Support	0	10

Since there were no audit findings, no additional NRC follow-up is planned. However, the large number of recommendations in audit 94-02-OPS reflects the need for the licensee to review and enhance the conduct of activities that support plant operations or in plant operations itself. A noted strength in audit 94-02-OPS was the use of two experienced B&W plant SROs, obtained from other facilities, for a two week period during this audit. Also, an experienced/qualified fire protection engineer, obtained from another facility, reviewed the CR-3 Fire Protection Plan during audit 94-03-SSUP.

b. The inspector attended an NGRC meeting on May 11, 1994. The agenda included a plant status report by the Director, Nuclear Plant Operations and NGRC subcommittee reports for Operations and Maintenance; Radiation, Chemistry and Environment; Quality and Regulatory Verification; and Engineering and Technical Support. Each report was comprehensive. A proposed revision to the NGRC charter was presented. This revision reflected the relocation of the NGRC review and audit requirements from the TS to FSAR Section 12.8.2.8. The NGRC charter revision was approved. The inspector determined that the NGRC was operating in a manner that met the FSAR commitments and promoted plant safety.

c. The inspectors reviewed the Nuclear Plant Technical Support Groups first quarter report for 1994 (dated May 3, 1994). The report consisted of an administrative overview, system performance evaluation reports, program status, and major component/project status. Also, the Rotating Equipment condition Monitoring Program quarterly report (dated April 25, 1994) was reviewed. This report summarized the rotating equipment condition monitoring program status. These reports provided a comprehensive summary of significant work activities and system/component status. Known equipment problems were discussed with proposed corrective actions or action plans presented.

Violations or deviations were not identified.

8. Onsite Engineering Evaluation (37551)

The inspectors performed an assessment of the onsite engineering function to determine the effectiveness of the onsite engineering staff. This includes onsite design engineers, system engineers, component engineers, shop engineers and any onsite staff providing engineering support to enhance the plant performance. A limited assessment of the engineering processes was performed to determine the adequacy of support to the plant.

a. An issue involving bearing wear and alignment problems on the B EGDG was evaluated to determine how effective the licensee was in determining the cause of the problem, dispositioning any operability issues, implementing corrective actions, and expanding the scope of the corrective actions to include applicable related systems, equipment, procedures, and personnel actions.

   The licensee noted that thrust strain could not be set within tolerances and that thrust bearings were exhibiting a tendency to flash and degrade. Investigation revealed that thrust strain had been left outside of the vendor recommended tolerances during the previous adjustment. Systems engineering attempted to perform tests to determine the cause but were unable to come to conclusions. A vendor representative was brought on site and worked with the systems engineer to evaluate possible causes of the problems. Two major problems were identified: a yaw existed in the thrust bearing, preventing correct alignment of the bearing to the shaft, and the insulation ring in the generator bearing had degraded, allowing eddy current to be transmitted to the bearings. The generator bearing was replaced and a specially made thrust bearing, designed to compensate for the yaw in the mounting brackets, was installed. The licensee was able to adjust the thrust strain within tolerances. Post maintenance testing was conducted successfully and the EGDG was returned to operation.

   Systems engineering coordinated testing on the EGDG, with the help of the vendor representative, to determine the cause of the problem. The probable cause was identified and corrected. The task was well planned and scheduled and coordination with maintenance was conducted well, resulting in no major impact to the outage schedule.

b. A second problem, involving the replacement of the thermal sleeve for MUV-37, HPI injection valve, was evaluated by the inspectors. During the refueling outage, the licensee performed radiography on the HPI thermal sleeves, to check for gap formation. MUV-37 had indications of a crack having formed.

   The old thermal sleeve was removed and preliminary engineering analysis determined that the shape of the thermal sleeve contributed to failure of the component, with sharp bends causing flow perturbations. As a result, it was decided to install a previously redesigned thermal sleeve that has a different shape to preclude recurrence.

The original replacement thermal sleeve was inserted for fit-up, but became wedged in place. Attempts to remove the thermal sleeve resulted in damage to the thermal sleeve and it had to be replaced. The second sleeve was inserted in place, but had to be machined to a shorter length, to enable it to fit properly. Engineering supplied support and the necessary calculations and guidance to successfully complete the job.

The removed thermal sleeve will be sent for metallurgical analysis to determine the exact failure mechanism. Throughout this evaluation, engineering provided the guidance and support necessary to successfully complete the assigned tasks. See IR 50-302/94-11 for further details.

Violations or deviations were not identified.

9. Refueling Activities (60710)

The inspectors observed refueling operations during the reload activities. Fuel movement in the vessel and in the spent fuel pool area was observed. The inspectors witnessed the coordination effort conducted from the main control room area for the fuel movements. The licensed operator moving fuel in the spent fuel pool area observed bowed fuel rods in assembly N5PE, which was scheduled to be installed in core position H-8, the center of the core. The operator notified the reactor engineer in the control room, who directed the operator to place the assembly in a holding location while it was inspected. The fuel movement sheets were revised by the reactor engineer to continue with the assembly loading, while leaving that location empty.

The assembly was video taped and sent to BWNS, who concurred with the licensee that it was likely that a soft stop was broken in the assembly. A different assembly, N48W, was identified that had similar burn-up histories and exposures as N5PE. This assembly was substituted in location H-8 and a new loading map was generated. Assembly N5PE will be inspected, by the licensee, at a later date.

Core reload was completed at 6:36 p.m. on May 14, 1994, and core verification was completed at 3:00 a.m. on May 15, 1994. Refueling activities were well coordinated and good communications were maintained.

Violations or deviations were not identified.

10. Onsite Follow-up and In-Office Review of Written Reports of Non-routine Events and 10 CFR Part 21 Reviews (90712/90713/92700)

The Licensee Event Reports and/or 10 CFR Part 21 Reports discussed below were reviewed. The inspectors verified that reporting requirements had been met, root cause analysis was performed, corrective actions appeared appropriate, and generic applicability had been considered. Additionally, the inspectors verified the licensee had reviewed each event, corrective actions were implemented, responsibility for corrective actions not fully completed was clearly assigned, safety questions had been evaluated and resolved, and violations of regulations or TS conditions had been identified. When applicable, the criteria of 10 CFR Part 2, Appendix C, were applied.

a. (Closed) LER 50-302/93-008, and LER 50-302/93-008-01: Due to Lack of Engineering Review, Motor Operated Valves With Brakes Could Fail to Perform Their Safety Function Under Degraded Voltage Conditions.

   As documented in paragraph 11.b, the licensee has modified the affected MOVs in order to eliminate the conditions which affected operability. The inspectors will follow the licensee's actions regarding the verification of the calculations due by the end of July, 1994. This LER is closed.

b. On January 10, 1994, COLTEC Industries, Fairbanks Morse Engine Division of Beloit, Wisconsin, made a 10 CFR Part 21 notification regarding a problem with the air start distributor cam used in the EGDGs. Among the affected sites referenced in the Part 21 report was the Crystal River Nuclear Plant. This notification was previously documented in IR 50-302/94-03, paragraph 7.c.

   During the current refueling outage (9R) both the A and B EGDGs were inspected for the air start distributor cam problems noted in the Part 21 report. Procedure MP-499, Emergency Diesel Generator Engine Inspection/Maintenance, Revision 1, paragraph 4.1, Refueling Interval Inspections, requires the disassembly, cleaning and refurbishment of the air start valves. Enclosure 21 of MP-499 is used for cleaning and inspection of the air start distributor. The inspectors examined the disassembled components of the B EGDG air start distributor and no adverse conditions were noted. The existing air start distributor cams in both the A and B EGDGs appeared to be in good condition. However, the information from COLTEC Industries recommended replacement of the cams if the scribe line on the cam was from a chisel. Since both existing cams had scribe lines that apparently were made by a chisel, they were replaced. The remaining components all appeared to be in good condition with slight amounts of debris present.

   Since the licensee has now made the air start distributor disassembly, cleaning, and refurbishment a refueling interval inspection, this 10 CFR Part 21 issue is closed.

Violations or deviations were not identified.

11. Maintenance Activities Follow-up (92902)

On April 14, 1994, there was a report from workers in the Reactor Building that steam was detected coming from the pressurizer vents. The pressurizer vents were open to support the RCS drain down. Investigation by the licensee indicated that pressurizer heater group 4 indicated approximately 50 amps on the pressurizer heater MCC and 62 kV was indicated on the control board indicator RC-203-J1. All main control board pressurizer heater switches were selected to the off position. PR 94-0095, Inadvertent Pressurizer Heater Activation While Control Switches Were Off, was issued to provide for investigation and corrective action.

The pressurizer heaters are arranged in five banks for control. Banks A, B, and C use SCR control to regulate current through the heater elements. As system pressure decreases, the time that the SCRs are gated on increases, thereby supplying more AC power to the heater elements. The amount of time the heaters are energized is proportional to the error between actual system pressure and normal operating pressure. The pressure signal for control is supplied by the RPS through a selector jack which will allow selection of either loop A or loop B for control.

Heater banks D and E, when in automatic, will be either full on or off at setpoint. These two banks can be selected to be energized at all times (subject to level interlock) by placing the control switch in ON. Any bank of heaters may be selected to remain off by placing the control switch to OFF. The control switches for banks A, B, and C are two position (OFF/AUTO) switches with red and green indicating lights to display when the heaters are either energized or off respectively. The control switches for banks D and E are three position (ON/OFF/AUTO) switches with green, white, and red indicating lights to indicate when the heaters are de-energized (green), heaters are energized (red), and if power is available (white). The 480 VAC pressurizer heater MCCs A and B are fed from the 480 VAC Reactor Aux Bus 3A and 3B respectively. Heaters are powered from MCCs as follows:

Pressurizer Heater MCC A	Pressurizer Heater MCC B
SCR Gp 1, 3, and 4	SCR Gp 2, 5, and 6
Gp 7, 8, and 9	Gp 10, 11, 12, and 13

BANK	GROUP	NO. ELEMENTS	RATING IN kW
A	1	9	126
B	2	9	126
C	3-6	36	504
D	7-9	27	378
E	10-13	36	504

A WR (NU0318653) was originated to perform trouble shooting to determine the cause of the heater turn on with the control switches in the off position. Under the work request trouble shooting, the licensee determined that a GTL in the control cabinet was out of calibration. A new GTU was installed and calibrated and the heaters performed satisfactorily. To ensure pressurizer heaters are de-energized when no longer required to maintain a pressurizer bubble, PR 94-0095 also has a corrective action step to revise OP-209, Plant Cooldown, to provide guidance to open the 480 VAC feeder breakers to remove pressurizer heater power. OP-209 now directs the operator to just turn the pressurizer heater banks off. This procedure change will ensure the pressurizer heaters cannot be energized by out of calibration controllers in the future. This action is scheduled to be completed by August 26, 1994. The inspectors will follow the licensee's actions to verify the noted procedure revision is completed.

Violations or deviations were not identified.

12. Engineering Activities Follow-up (92903)

a. In IR 50-302/93-13, paragraph 3.c, Emergency Preparedness for Hurricanes, the impact of non-safety equipment on important equipment during external events was discussed. The licensee had stated that an evaluation of the potential impact upon CR-3 if the smoke stacks from units 1 and/or 2 were to fall would be performed. In July 1993 the licensee performed a risk analysis of the units 1 and/or 2 stacks falling on the CR-3 site as an addition to the site IPEEE. The licensee determined that the risk associated with the postulated event was negligible. However, as a prudent approach, the licensee has completed construction of a reinforcing sleeve structure around the units 1 and 2 stacks which meet the current American Concrete Institute code for design and construction of concrete chimneys (ACI307-88). The licensee's analysis was forwarded by NRC Region II to NRR for review on September 9, 1993.

The licensee's consultant, Chimney Consultants, Inc. of West Lebanon, New Hampshire, had determined that the controlling failure mode of the stack was circumferential bending, occurring at a location 394 feet from the bottom of the stack. This resulted in the calculated length of stack failure, falling as a rigid body, having a striking distance of only 109 feet. The nearest CR-3 building to the stack is over 300 feet away. Based on the results of their review, NRR concurred with the licensee's conclusion that the Units 1 and 2 stacks did not pose a significant structural risk to CR-3 safety related structures. The structural upgrade also provided an added assurance that the chimney would not be a significant risk to safety related structures at CR-3.

Based on NRR's review, it was concluded that the Units 1 and 2 chimneys are not a significant structural risk to safety-related structures at the CR nuclear facility. Additionally, the Probabilistic Safety Assessment Branch calculated the postulated stack failure to be of negligible safety concern to core damage. This issue is closed.

b. (Closed) URI 50-302/94-09-01, Insufficient Voltage to Operate Main Feedwater Isolation Valve FWV-28.

As previously discussed in IR 50-302/94-09, on April 12, 1994, a four hour notification was made to the HQ duty officer regarding the operability of FWV-28. Previously, in June 1993, by LER 50-302/93-008, Due to Lack of Engineering Review, Motor Operated Valves With Brakes Could Fail to Perform Their Safety Function Under Degraded Voltage Conditions, other valves had been identified as having a discrepancy between the acceptance criteria for the electrical brake minimum operating voltage versus the valve motor minimum voltage acceptance criteria. The previously identified problem valves (MUV-58 and MUV-78, HPI suction valves from the BWST) had their stationary and rotating discs removed which permanently de-coupled the motor from the brake assembly and the electrical power connections to the brake were also removed. This modification was accomplished in July of 1993 under MAR 83-07-01-01. LER 50-302/93-008-01 was issued May 12, 1994, to include FWV-28 as another MOV with brakes that could fail to perform its safety function under degraded voltage conditions.

The current safety function of FWV-28 is to isolate feedwater upon receipt of an EFIC signal generated by a main steam line break. The only time FWV-28 would be required to perform this function is during start-up or shut-down when the plant is operating on one feedwater pump. For example, if the A FWP is running and there is a steam line break on the B OTSG, then upon reaching 600 psig in the B OTSG, EFIC would send a signal to close the B train suction valve, block valves, FWV-28, and trip the B FWP, but not the A FWP or valves. During this scenario with FWV-28 inoperable and a single failure of the block valve, there would be continued feed to the bad generator which could result in an over cooling of the RCS.

The licensee modified the EFIC logic per MAR 93-05-11-01 during the current refueling outage (9R). MAR 93-05-11-01 provided the capability to change the FWP trip and FWP suction valve closure logic when FWV-28 would be open (below 55% reactor power) by providing two key locked switches with a FWV-28 open position (BOTH) and a FWV-28 closed position (ONE). When the key lock switches are placed in the BOTH position (FWV-28 open), if a FWI signal is generated by EFIC, the logic will trip both MFWPs and close both MFWP suction isolation valves. When the key lock switches are placed in the ONE position (FWV-28 closed), the EFIC logic functions as originally designed and isolates the affected OTSG.

The above MAR modified the EFIC logic such that when FWV-28 is open, both MFWPs and their suction isolation valves will isolate on low OTSG pressure, thereby eliminating the requirement for FWV-28 to close. The inspectors verified that the key lock switches per the noted MAR had been installed during the 9R outage. The inspectors also witnessed portions of the post modification testing performed under post MAR test procedures 93-05-11-01, TP:1, and TP:2.

The failure to perform an adequate design review resulting in potentially inoperable safety related MOVs (MUV-58, MUV-78, and FWV-28) under degraded voltage conditions is a violation. This violation is being tracked as non-cited violation 50-302/94-14-01, Failure to perform an adequate design review resulting in potentially inoperable MOVs under degraded voltage conditions. This licensee identified violation is not being cited because the criteria specified in Section VII.B of the NRC Enforcement Policy were satisfied.

One NCV was identified.

13. Fire Protection (64704)

During the previous inspection period, the inspector continued an inspection of fire protection begun in February 1994 (documented in NRC IR 50-302/94-05). The inspector accompanied licensee fire protection personnel on their performance of SP-809, Fire Protection Weekly Inspection. Although this procedure is officially performed once per week, the licensee had been regularly performing daily walkdowns of all accessible plant areas as described by SP-809, and was performing two such walkdowns per day during the outage. The NRC inspector found the licensee's inspection to be thorough and complete, and the individual conducting the inspection was knowledgeable of applicable regulatory requirements. A good familiarity with ongoing work items was also demonstrated, so that cleanliness practices were being effectively monitored during work in progress and any unattended flammable material identified. The licensee was doing a good job of pro-actively identifying potential hazards or deficiencies, bringing them to the attention of appropriate line management, and tracking the identified items to ensure timely resolution.

14. Exit Interview

The inspection scope and findings were summarized on June 10, 1994, with those persons indicated in paragraph 1. The inspectors described the areas inspected and discussed in detail the inspection results listed below. Proprietary information is not contained in this report. Dissenting comments were not received from the licensee.

Item Number	Status	Description and Reference
NCV 50-302/94-14-01	Closed	Failure to Perform an Adequate Design Review Resulting in Potentially Inoperable Motor Operated Valves Under Degraded Voltage Conditions. (paragraph 12.b)
LER 50-302/93-008 and LER 50-302/93-008-01	Closed	Due to Lack of Engineering Review, Motor Operated Valves With Brakes Could Fail to Perform Their Safety Function Under Degraded Voltage Conditions. (paragraph 10.a)
URI 50-302/94-09-01	Closed	Insufficient Voltage to Operate Main Feedwater Isolation Valve FWV-28. (paragraph 12.b)
10 CFR Part 21	Closed	Air Start Distributor Cam Problems in Fairbanks-Morse Diesel Generators (paragraph 10.b)

15. Acronyms and Abbreviations

AC - Alternating Current
ALARA - As Low as Reasonably Achievable
AFW - Auxiliary Feedwater
amps - amperes
AMSAC - ATWS (anticipated transient without scram) mitigating system actuation circuitry
ANO - Auxiliary Nuclear Operator
ATWS - Anticipated Transient Without Scram
B&W - Babcock & Wilcox
BWNS - B&W Nuclear Services
BWST - Borated Water Storage Tank
CAL - Confirmatory Action Letter
CDT - Condensate Storage Tank
CFR - Code of Federal Regulations
cps - count per second
CR-3 - Crystal River Unit 3
CRA - Control Rod Assembly
CRDM - Control Rod Drive Mechanism
DJP - Diesel Jacket Pump
ECAD - Engineering Computer Aided Drawing
EGDG - Emergency Diesel Generators
EFIC - Emergency Feedwater Initiation and Control System
ES - Engineered Safeguards
ESF - Engineered Safeguards Feature
FCN - Field Change Notice
FPC - Florida Power Corporation
FSAR - Final Safety Analysis Report
FWI - Feedwater Isolation

FWP - Feedwater Pump
FWV - Feedwater Valve
GSI - Generic Safety Issue
GTU - Gate Trigger Unit
HP - Health Physics
HPI - High Pressure Injection
I&C - Instrumentation and Control
IPEEE - Individual Plant Examination of Externally Initiated Events
IR - Inspection Report
kV - kilovolt
kW - kilowatt
LCO - Limiting Condition for Operation
LER - Licensee Event Report
MAR - Modification Approval Record
MCB - Main Control Board
MCC - Motor Control Center
MFWI - Main Feedwater Isolation
MFWP - Main Feedwater Pump
MOV - Motor Operated Valve
MP - Maintenance Procedure
MUV - Make-up Valve
NCV - Non-cited Violation
NGRC - Nuclear General Review Committee
NI - Neutron Instrumentation
NOP - Normal Operating Pressure
NOT - Normal Operating Temperature
NOV - Notice of Violation
NRC - Nuclear Regulatory Commission
NRR - Office of Nuclear Reactor Regulation
OP - Operating Procedure
OTSG - Once Through Steam Generator
ppm - parts per million
PR - Problem Report
psig - pounds per square inch gauge
PT - Performance Testing Procedure
QC - Quality Control
QA - Quality Assurance
RB - Reactor Building
RCA - Radiation Control Area
RCDT - Reactor Coolant Drain Tank
RCS - Reactor Coolant System
RPS - Reactor Protection System
RV - Reactor Vessel
SCR - Silicon Control Rectifier
SP - Surveillance Procedure
SR - Surveillance Requirement
SRO - Senior Reactor Operator
SRP - Standard Review Plan
TP - Test Procedure
TS - Technical Specification
TSP - Tri Sodium Phosphate

URI - Unresolved Item
V - Volt
VAC - Volts Alternating Current
WR - Work Request