U.S. NUCLEAR REGULATORY COMMISSION

REGION II

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License No:	DPR-23
Report No:	50-261/98-08
Licensee:	Carolina Power & Light (CP&L)
Facility:	H. B. Robinson Unit 2
Location:	3581 West Entrance Road Hartsville, SC 29550
Dates:	August 2 - September 12, 1998
Inspectors:	B. Desai, Senior Resident Inspector A. Hutto, Resident Inspector E. Girard, Reactor Inspector (Sections M1.2, E8.1) E. Guthrie, Resident Inspector (Section M2.1) W. West, Project Engineer (Section R1.1)
Approved by:	Brian R. Bonser, Chief Reactor Projects Branch 4 Division of Reactor Projects

Enclosure

EXECUTIVE SUMMARY

H. B. Robinson Power Plant, Unit 2
NRC Integrated Inspection Report 50-261/98-08

This integrated inspection included aspects of licensee operations, maintenance, engineering, and plant support. The report covers a six-week period of resident inspection; in addition, it includes the results of announced inspections by two regional inspectors.

Operations

- The conduct of operations was professional, risk informed, and safety-conscious (Section O1.1).
- Two components were found to be out of their normal position. Appropriate corrective actions were taken to restore these components to their normal configuration. The safety significance for both these events was minimal (Section O2.1).
- The licensee took appropriate measures to prepare the site for adverse weather conditions in anticipation of Hurricane Bonnie (Section O2.2).
- A non-cited violation (NCV) was identified involving a failure to perform Technical Specification (TS) required iodine sampling following a power reduction. Operators did not fully understand the requirements for sampling dose equivalent iodine following a rapid power change (Section O4.1).
- The onsite review functions of the Plant Nuclear Safety Committee (PNSC) were conducted in accordance with TSs. During the PNSC meetings topics were thoroughly discussed and evaluated (Section O7.1).

Maintenance

- The inspectors concluded that the licensee had demonstrated that valve CC-749A was not degraded. The increased stroke time previously experienced was apparently the result of a measurement error. The retesting of valve CC-749A could have been performed in a more timely manner (Section M1.2).
- Surveillance tests observed were performed adequately. Operations and maintenance personnel exhibited knowledge of the tasks and the results met the acceptance criteria (Section M2.1).

Engineering

- A review of an Engineering Service Request that changed the Isolation Valve Seal Water System tank operating pressure and low pressure alarm concluded that the plant change was performed in accordance with engineering modification package requirements (Section E2.1).

- A licensee operability determination for an Emergency Diesel Generator (EDG) Service Water piping configuration found to be different than the original design was thorough. The piping maintained structural integrity and EDG operability was not affected (Section E2.2).

- A review of the Auxiliary Feedwater (AFW) system determined that the system was capable of performing its safety function. The system engineer was knowledgeable of system status, maintenance rule data, outstanding system work orders, and planned/proposed system modifications (Section E2.3).

<u>Plant Support</u>

- The inspectors concluded that radiation control and security practices were proper (Section R1.1 and S1.1).

- Radiological controls utilized during a Chemical and Volume Control System filter replacement were effective in minimizing personnel exposure and contamination (Section R1.2).

Report Details

Summary of Plant Status

On August 1, power was reduced to 65 percent from 100 percent for approximately 14 hours to conduct main turbine valve testing. The unit was returned to 100 percent on August 2, and operated at 100 percent through the report period.

I. Operations

01 Conduct of Operations

01.1 General Comments (71707)

The inspectors conducted frequent control room tours to verify proper staffing, operator attentiveness and communications, and adherence to approved procedures. The inspectors routinely attended operation turnovers, management reviews, and plan-of-the-day meetings to maintain awareness of overall plant operations. Operator logs were reviewed to verify operational safety and compliance with Technical Specifications (TS). Instrumentation, computer indications, and safety system lineups were periodically reviewed from the control room to assess operability. Frequent plant tours were conducted to observe equipment status and housekeeping. Condition Reports (CRs) were routinely reviewed to assure that potential safety concerns and equipment problems were reported and resolved. Good plant equipment material conditions and housekeeping continued to be observed throughout the report period. In general, the inspectors concluded that the conduct of operations was risk informed, professional, and safety conscious.

02 Operational Status of Facilities and Equipment

02.1 Equipment Out-of-Position

a. Inspection Scope (71707)

The inspectors reviewed and assessed the circumstances surrounding the discovery of the dedicated shutdown (DS) uninterruptible power supply (UPS) and Service Water (SW) valve SW-143 out of their normal position.

b. Observations and Findings

On July 2, a plant operator performing a plant tour, found the DS UPS aligned to its alternate power supply. In this configuration the DS UPS battery would not automatically keep the UPS energized should loss-of-off-site-power occur. The outside auxiliary operator had reported that the UPS was aligned to its normal power supply earlier in the shift. No instrument changes were noted that could have resulted in a shift of the power supply. The DS UPS was returned to its normal power supply in accordance with plant procedures with no abnormalities noted. To date, the licensee has been unable to determine the actual cause of the power supply shift. The safety significance was minimal as the DS UPS is not relied upon for accident mitigation and the power supply can be manually shifted to the battery supply in the event of loss-of-off-site-power.

On August 27, the outside auxiliary operator found SW-143, the outlet valve for the generator hydrogen cooler temperature control valve to be 25 percent open. Operating procedures specify this valve to be fully open. Hydrogen temperature was determined to be controlling satisfactorily at the time. It was determined that vibrations had caused the valve to drift. This problem had been documented before in 1991, prior to some system modifications. The corrective action was to chain the valve in the open position. The safety significance of the mispositioned valve was minimal as the SW flow to the generator hydrogen cooler was found to be maintaining normal temperatures and the system is not safety related.

The inspectors verified that these two instances of mispositioned equipment did not constitute an operability concern. The inspectors determined that the licensee evaluated the cause of the mispositioning issues and dispositioned the two cases in the corrective action program.

c. Conclusions

Two components were found to be out of their normal position. Appropriate corrective actions were taken to restore these components to their normal configuration. The safety significance for both of these events was minimal.

O2.2 Hurricane Preparedness

a. Inspection Scope (71707)

The inspectors monitored licensee activities related to Hurricane Bonnie.

b. Observations and Findings

The licensee initiated actions in accordance with procedure OMM-21, "Operation During Adverse Weather Conditions", Revision 17. The actions included organizing personnel for emergency response, securing plant equipment and material, and installing lifelines for high wind conditions. Hurricane force winds did not reach the plant and there were minimal consequences as a result of the hurricane.

c. Conclusions

The licensee took appropriate measures to prepare the site for adverse weather conditions in anticipation of Hurricane Bonnie.

04 Operator Knowledge and Performance

04.1 Missed TS Surveillance for Iodine Sampling Following Power Reduction

a. Inspection Scope (71707)

The inspectors reviewed the failure to perform a TS required iodine sampling surveillance following a thermal power change of 15 percent or greater in any one hour period.

b. Observations and Findings

On August 2, thermal power was reduced from 100 percent to 65 percent to support turbine valve testing. This power reduction occurred over a period of one hour and 42 minutes. TS surveillance requirements require verifying Reactor Coolant System (RCS) dose equivalent iodine, I-131, specific activity to be less than or equal to one microcurie per gram between two and six hours after a thermal power change of greater than or equal to 15 percent within a one hour period. The purpose of the test is to detect fuel failure, and sampling times are specified to coincide with peak iodine levels following potential fuel failure. The above power change was conducted in two discrete intervals. Unit load reduction from 100 percent power began at 10:04 p.m. and 85 percent was reached at 10:43 p.m. Chemistry was notified of the 15 percent power change at 10:43 p.m. and the RCS was sampled at 12:52 a.m. Reactor power was further reduced to 65 percent at 11:46 p.m., resulting in a second power change of greater than 15 percent in less than one hour. The operating crew failed to recognize the requirement to obtain a second RCS iodine sample. At shift change, the oncoming crew indicated that a second RCS iodine sample was needed and chemistry was instructed to obtain a second sample. At this point, TS SR 3.0.3 was entered and the reactor coolant verified to contain less than or equal to one microcurie per gram I-131, after which SR 3.0.3 was exited.

A significant CR (98-01665) was written to document the missed surveillance. A similar missed surveillance occurred in November, 1997, following a power increase. In the previous case, the operators failed to realize an I-131 sample was required (NCV 50-261/97-12-02). In April, 1998, another event based TS surveillance was missed for containment air lock leak rate testing (VIO 50-261/98-05-02). A number of corrective actions were implemented following these events to aid the operators in identifying and tracking event based TS required surveillances. These included a revision to the Equipment Inoperable Record (EIR) procedure to include an EIR tracking form for event based surveillances. In the August 2 case the operators were aware of the surveillance requirement prior to the power reduction. They failed to recognize that the second power reduction required a separate iodine sample.

The licensee plans to implement several corrective actions as a result of this event. Training is planned for operations and chemistry personnel concerning fast power changes, chemistry notification, and preparation of procedural guidance for I-131 sampling. Procedural changes are planned for OMM-001-13 "Plant Chemistry", Revision 4, clarifying I-131 sampling requirements, with revisions to GP-005 "Power Operation", Revision 57; GP-006 "Normal Plant Shutdown From Power Operation to Hot Shutdown", Revision 33; and OP-105 "Maneuvering The Plant When Greater Than 25% Power", Revision 12; to reference this guidance. The inspectors reviewed TS, operator logs, sample results and interviewed licensee personnel regarding the activities associated with the downpower and the TS required iodine sampling.

The I-131 dose equivalent activity in the reactor coolant prior to the power decrease was a factor of 1000 below the limit of one microcurie per gram. The I-131 dose equivalent activity actually decreased slightly after the power change. Therefore, the missed surveillance had minimal safety significance. This non-repetitive, licensee identified and corrected violation is being treated as an NCV, consistent with Section VII.B.1 of the NRC enforcement policy. This issue is documented as NCV 50-261/98-08-01: Missed TS SR 3.4.16.2, Reactor Coolant I-131 Dose Equivalent Verification.

c. Conclusions

An NCV was identified involving a failure to perform TS required iodine sampling following a power reduction. Operators did not fully understand the requirements for sampling dose equivalent iodine following a rapid power change.

07 Quality Assurance In Operations

07.1 Plant Nuclear Safety Committee and Nuclear Assessment Section Oversight

a. Inspection Scope (71707)

The inspectors evaluated Plant Nuclear Safety Committee (PNSC) and Nuclear Assessment Section (NAS) activities to determine whether the onsite review functions were conducted in accordance with TS requirements.

b. Observations and Findings

The inspectors periodically attended PNSC meetings during the inspection report period. The presentations to the committee were thorough and the presenters readily responded to all questions. The committee members asked probing questions and were well prepared. The committee members displayed an understanding of the issues. The inspectors also reviewed NAS audits and concluded that they were appropriately focused to identify and enhance safety.

c. Conclusions

The onsite review functions of the PNSC were conducted in accordance with TSs. During the PNSC meetings topics were thoroughly discussed and evaluated.

O8 Miscellaneous Operations Issues and Open and Closed Items (92901)

O8.1 (Closed) Violation (VIO) 50-261/98-05-02: failure to perform personnel air lock testing. The inspector verified licensee corrective action resulting from the missed surveillance. Licensee corrective action included revision of procedure OMM-007, "Equipment Inoperability Record", Revision 45, which now includes a method to document, track, and flag conditional surveillance requirements. The corrective actions were further detailed in CR(98-00890). Based on a review of the CR and the upgraded procedure, this item is considered closed.

II. Maintenance

M1 Conduct of Maintenance

M1.1 Evaluation of Valve CC-749A Stroke Time

a. Inspection Scope (62707)

The inspectors reviewed the licensee's actions to address an unexpectedly high opening stroke time on Component Cooling Water (CCW) valve CC-749A. Valve CC-749A was the Residual Heat Removal Heat Exchanger CCW discharge isolation valve.

b. Observations and Findings

During surveillance testing on August 21, the licensee measured an opening stroke time on valve CC-749A of 184 seconds. This was an increase of about 32 seconds over previous testing. This stroke time was within the licensee's procedural acceptance limit of 200 seconds. However, this large a stroke time increase could have been indicative of a test error or serious degradation (such that the valve might not be capable of performing its design function).

On August 31,the inspectors were informed that the licensee had not retested or otherwise investigated the stroke time increase. The licensee had initiated a Work Request/Job Order (98-AFII1) to address the increased stroke time on August 25, 1998, but no action had been taken. The inspectors were informed that the licensee had delayed action to provide for contingencies, such as availability of replacement parts for any necessary repair. Licensee MOV personnel stated that they did not believe CC-749A was degraded but that the increased stroke time was due to a measurement error. They stated they would have expected a thermal overload trip or a failure of the valve to close after testing if the stroke time increase was caused by degradation.

The inspectors reviewed data from diagnostic tests performed on CC-749A and on the similar B-train CC-749B valve after their modification during RO 18 (March 1998) and found no indication of an unsatisfactory condition. This data was documented in the following records:

- Valve CC-749A, "Analysis of MOV Diagnostic Data," Static Test #1/3-24-98
- Valve CC-749B, "Analysis of MOV Diagnostic Data," Static Test #2/3-24-98

The inspectors agreed that the increased stroke time of CC-749A was likely due to a test error. However, they questioned the licensee's delay in retesting the valve in order to assure availability of replacement parts. The plant was operating and determining if there was serious valve degradation could have been performed in a more timely manner. On September 1, 1998, the licensee retested CC-749A while monitoring motor current with diagnostics. The inspectors observed the test. The stroke time was at the expected value and the current measurement plot did not reveal any degradation.

c. Conclusions

The inspectors concluded that the licensee had demonstrated that valve CC-749A was not degraded. The increased stroke time previously experienced was apparently the result of a measurement error. The retesting of valve CC-749A could have been performed in a more timely manner (Section M1.2).

M2 Maintenance and Material Condition of Facilities and Equipment

M2.1 Surveillance Observation

a. Inspection Scope (61726)

The inspectors observed all or portions of the following surveillance tests:

- MST-023 Safeguards Relay Rack Train "B", Revision 12
- OST-201-2 MDAFW System Component Test Train "B", Revision 8
- SP-1438 SDAFW Pump Emergency Cooling System Flow Test
- OST-101-2 CVCS Component Test Charging Pump "B", Revision 12

b. Observations and Findings

The surveillances were performed in a professional manner by qualified personnel. Procedures for the tests were present and in use during the testing. Pre-job briefings were held prior to doing the work. Upon completion of the testing, restoration and alignment of plant components was completed and verified. All surveillances were completed satisfactorily and met the acceptance criteria. Several observations were noted by the inspectors during the conduct of OST-101-2,"CVCS Component Test Charging Pump B."

During the conduct of the B Charging Pump surveillance on August 12, the inspectors noted an inconsistency in data gathering and an omission during the pre-job briefing. The pre-job briefing did not cover the test acceptance criteria which was a management expectation. The inspectors also observed discrepancies in the pump data gathering. The operator discussed the issue with the shift superintendent who then decided to reperform the test and obtain another set of data on charging pump B. The inspectors observed the retesting and reviewed the test paperwork and verified that the surveillance met the acceptance criteria.

c. Conclusions

Surveillance tests observed were performed adequately. Operations and maintenance personnel exhibited knowledge of the tasks and the results met the acceptance criteria.

III. Engineering

E2 Engineering Support of Facilities and Equipment

E2.1 Isolation Valve Seal Water (IVSW) Modification

a. Inspection Scope (37551)

The inspectors reviewed Engineering Service Request (ESR) 9700326, related to the IVSW.

b. Observations and Findings

Technical Specification 3.6.8, IVSW, requires that the IVSW tank pressure be maintained greater than or equal to 44 psig and verified every 12 hours. Prior to implementing the ESR, the IVSW tank pressure was maintained and verified at approximately 47 psig. The low pressure alarm set point was set at 42 psig. This did not give adequate warning to the operator of a potential tank pressure below the TS limit. The ESR raised the IVSW pressure to approximately 52 psig and changed the low pressure alarm set point to 48 psig. This change would give operators sufficient warning of a TS minimum limit being approached.

The inspectors reviewed and verified the ESR package including the 10 CFR 50.59 evaluation, the ESR installation procedure, and the proposed revisions to the alarm calibration procedure (PIC-025), the revised annunciator response procedure (APP-007), the revised operating procedure (OP-911), the revised surveillance procedure (OST-933), and the Updated Final Safety Analysis Report (UFSAR) section.

c. Conclusions

A review of an Engineering Service Request that changed the IVSW tank operating pressure and low pressure alarm concluded that the plant change was performed in accordance with engineering modification package requirements.

E2.2 Operability Determination for SW Piping to Emergency Diesel Generator (EDG) Heat Exchangers

a. Inspection Scope (37551)

The inspectors observed/reviewed licensee activities regarding a degraded EDG SW piping condition.

b. Observations and Findings

During a walkdown of the SW system supporting the EDGs, the licensee identified that a portion of the SW piping serving the skid mounted EDG heat exchangers utilized a rubber expansion joint at the connection lines and on the SW return lines. The calculation of record used to seismically qualify the SW piping (SW-12-7112) assumed a rigid anchor versus a flexible connection. This condition was applicable to both the EDGs.

The licensee initiated an operability evaluation in accordance with PLP-102, "Operability Determinations", Revision 1, to evaluate the actual SW configuration to determine if it met the Short Term Structural Integrity (STSI) requirement of licensee procedure EGR-NGGC-0320, Revision 2. The STSI of the SW lines and their intended support to the EDGs was confirmed by re-performing the calculations assuming as built configuration with the rubber expansion joints installed. The licensee performed a safety evaluation and concluded that an Unreviewed Safety Question did not exist as a result of this condition.

The licensee intends to permanently resolve the condition through the corrective action program. The licensee is projecting that an analysis to determine the method to fully (long term) qualify the SW line would be completed prior to the next refueling outage.

The root cause for the discrepant condition between the stress calculation piping analysis and the field condition was attributed by the licensee to be inadequate field verification during the implementation of NRC Bulletin (IEB) 79-14, "Seismic Analysis for As The Built Safety Related Piping Systems."

The inspectors reviewed the licensee's safety evaluation, root cause evaluation, plant drawings, calculations and performed walkdowns of the EDG piping. The inspectors concluded that the operability evaluation was adequate and agreed with the conclusion that the EDG SW piping maintained structural integrity.

c. Conclusions

A licensee operability determination for an EDG SW piping configuration found to be different than the original design was thorough. The piping maintained structural integrity and EDG operability was not affected

E2.3 Auxiliary Feedwater (AFW) Walkdown

a. Inspection Scope (37551)

The inspectors performed a walkdown and review of the AFW system.

b. Observations and Findings

The inspectors walked down portions of the AFW system, reviewed open WRs, reviewed the AFW system engineer's note book, reviewed system maintenance rule data, and attended the engineering system status meeting.

c. Conclusions

A review of the AFW system determined that the system was capable of performing its safety function. The system engineer was knowledgeable of system status, maintenance rule data, outstanding system work orders, and planned/proposed system modifications.

E8 Miscellaneous Engineering Issues (37551, 92903)

E8.1 (Closed) Inspector Followup Item (IFI) 50-261/98-01-01: Generic Letter (GL) 89-10 commitments. This followup item was opened pending the licensee's completion of three commitment actions described in a February 20, 1998, letter to the NRC. These actions were developed to resolve issues raised during an NRC inspection of the licensee's implementation of GL 89-10, "Safety-Related Motor-Operated Valve Testing and Surveillance." In a letter to the NRC dated June 25, 1998, the licensee informed the NRC that the commitment actions had been completed and summarized important details of the associated test results and analyses.

The inspectors reviewed licensee documentation to verify completion of the commitment actions and to verify completion of several other actions the licensee had planned for improvement of the Robinson MOV Program (described in NRC Integrated Inspection Report No. 50-261/98-01). The inspectors confirmed that the actions were satisfactorily completed and that the associated document revisions, tests, analyses, and hardware changes were thorough and technically sound. The reviews conducted by the inspectors and their findings are discussed below:

Commitment - MOV Stem Coefficient of Friction Assumptions

The licensee committed to perform testing during RO 18 to provide more precise static and dynamic stem factors to support the stem coefficient of friction (COF) assumptions contained in Engineering Service Request ESR-9700331, "Determination of MOV Stem Factors," Revision 0. The inspectors verified completion of this commitment action by reviewing the current ESR-9700331 (Revision 1) and confirming that it documented and analyzed stem factors obtained from tests performed during RO 18. In addition, the inspectors reviewed the following RO 18 static and dynamic (DP) test records for consistency with ESR-9700331 data:

- Valve RHR-744A, "Analysis of MOV Diagnostic Data," DP Test #2/3-27-98
- Valve RHR-744A, "Analysis of MOV Diagnostic Data," Static Test #1/3-24-98
- Valve CC-749A, "Analysis of MOV Diagnostic Data," Static Test #1/3-24-98

ESR-9700331, Revision 1, reported and statistically analyzed static test results from 29 valves and dynamic test results from 13 of the 29 valves. The ESR and the test records indicated that "Quick Stem Sensors" were used for the RO 18 tests to provide more precise measurements than with the licensee's previous torque wrench method. ESR-9700331 determined and justified new stem COFs, a COF allowance for stem degradation, and an assumed difference between static and dynamic COFs. The values were consistent with values given in the licensee's June 25, 1998, letter to the NRC. Based on a review of the logic used in the licensee's analysis and on hand calculations, the inspectors found that the licensee's analysis satisfactorily supported the new stem COFs, COF degradation allowance, and the assumed difference between static and dynamic COFs. The inspectors also found that the licensee had employed the new values in evaluating the impact of recently revised actuator guidance on MOV capabilities in ESR 9800408, "Limitorque Tech. Update 98-01 impact evaluation," Revision 0. The inspectors concluded that the commitment had been met.

Commitment - Evaluate the Hydrodynamic Torque Requirements for Butterfly Valves

The licensee committed to perform calculations, tests, and/or inspections to evaluate the hydrodynamic torque requirements for 16-inch Allis Chalmers butterfly valves V6-16A, B, and C. The licensee had previously assumed that seating torque predominated over hydrodynamic torque. The inspectors verified completion of this commitment action by reviewing the following record of tests which the licensee had performed on valve V6-16B:

- Valve V6-16B, "Analysis of MOV Diagnostic Data," Static and DP Tests 2, 3, 4, and 7; Date 4/7-8/98

The inspectors found that the analysis of test results contained in the above record evaluated the hydrodynamic torque and demonstrated that seating torque was predominant. This confirmed that the licensee assumption that seating torque predominated was correct. The inspectors concluded that the commitment had been met.

In their review, the inspectors found that the licensee had exceeded the torque rating of the Limitorque HBC actuator on valve V6-16B. Licensee personnel noted that this had occurred after having manually seated the valve without apparent undue force. The inspectors verified that the licensee had performed detailed visual inspections of the actuator components and determined that no damage was present following the overtorque. The inspections were documented on Work Request/Job Order 98-ACQZ1.

<u>Commitment - Revise Site Procedures to Require Diagnostic Verification of Switch Settings</u>

The licensee committed to revise site procedures to require diagnostic verification of close limit switch and torque switch bypass settings for valves that are position-controlled for accident scenarios, if the valves are capable of being diagnostically tested. The inspectors verified completion of this commitment action by confirming that Technical Management Procedure TMM-035, "Analysis and Trending of MOV Performance," had been revised (Revision 13) to require the stated diagnostic verification. In addition, the inspectors reviewed the following test records for position controlled valves and verified that the commitment had been implemented:

- Valve RC-535, "Analysis of MOV Diagnostic Data," Static Test #2/3-19-98
- Valve AFW-V2-14B, "Analysis of MOV Diagnostic Data," Static Test #1/3-14-98

The inspectors concluded that the commitment had been met.

<u>Improvement Action - Valve Factor (VF) for 1500 pound Copes-Vulcan 14-inch Parallel Double-Disc Gate Valves</u>

The licensee had not been able to test two 1500 pound Copes-Vulcan 14-inch parallel double-disc gate valves (RHR-750 and 751) to provide data to support the VF that was assumed in thrust calculations. The licensee indicated that subsequent efforts would be made to obtain supporting data. In the current inspection, the inspectors were informed that the licensee had obtained the results of an Electric Power Research Institute (EPRI) Performance Prediction Methodology (PPM) calculation performed by Consolidated Edison which supported the VF employed by Robinson. The inspectors reviewed ESR 97-00330, "Determination of MOV Valve Factors," Revision 2, which documented the licensee's evaluation of the PPM calculation results. The PPM calculation was not strictly applicable to the Copes-Vulcan valve design but was based on a similar Anchor Darling design. The PPM calculation results supported a 0.63 VF whereas the licensee specified a more conservative 0.65 VF. The inspectors considered this adequate for these valves.

Improvement Action - Comparison of Pressurizer Power Operated Relief Valve (PORV) Block Valves

Robinson's PORV block valves (RC-535 and RC-536) were 1500 pound Westinghouse 3-inch flex-wedge gate valves. The licensee obtained the results of tests performed by Comanche Peak on Westinghouse block valves and used these results to establish VFs for Robinson's block valves. The NRC inspectors expressed concern that the licensee had not compared the internals of the Robinson and Comanche Peak valves to confirm that the valves were of similar design, such that use of the Comanche Peak data would be appropriate. In response, the licensee indicated that this comparison would be made or that an EPRI PPM calculation would be used to confirm that the Robinson VFs were satisfactory. In the current inspection, the inspectors found that the licensee had obtained EPRI PPM calculation results and valve design information from Consolidated Edison which supported the assumed VFs. The inspectors reviewed the information obtained by the licensee, incorporated in ESR 97-00330, and determined that it resolved the previous concern.

Improvement Action - Obtain Industry Data to Justify the VF Applied to Fire Protection Valves

Fire Protection valves FP-248, 249, 256, and 258 were 900 pound Anchor/Darling 4-inch flex-wedge gate valves. The licensee applied a 0.80 closing VF to these valves in thrust calculations. There were no in-plant test results available to support this VF. The licensee indicated further efforts would be undertaken to obtain applicable industry data for these valves. In the current inspection, the inspectors found that the licensee had obtained additional information from Commonwealth Edison which supported the 0.80 VF. This was documented in ESR 97-00330.

Improvement Action - Modify Butterfly Valves to Provide Position Control for Closing

The safety function of the licensee's Generic Letter 89-10 butterfly valves was to close and closure was controlled by torque switch settings with the valves torquing closed into stopnuts. The licensee indicated the closing control scheme for these valves would be modified to position control to improve their closing capabilities. In the current inspection, the inspectors reviewed the following work request/job orders and verified that this modification had been completed:

- 97-AEIU1, completed 3/26/98
- 97-AEIW1, completed 3/27/98
- 97-AEIX1, completed 3/27/98

Improvement Action - Modify Gate Valves to Increase Their Capabilities

The licensee had identified several gate valves which had negative margins and were to be modified during RO 18 to improve their capability margins. In the current inspection, the inspectors verified adequate completion of the modifications through a review of the following modification design and post modification diagnostic test documents for three of the valves (RHR-744A, CC-749A, and CC-749B):

- ESR 97-00538, "Improve torque/thrust margin for various MOVs," Revision 0, which re-examined the operating requirements for these valves and developed modifications.
- Valve RHR-744A, "Analysis of MOV Diagnostic Data," DP Test #2/3-27-98
- Valve CC-749A, "Analysis of MOV Diagnostic Data," Static Test #1/3-24-98
- Valve CC-749B, "Analysis of MOV Diagnostic Data," Static Test #2/3-24-98

The inspectors determined that the design modifications, supported by post modification tests, provided positive capability margins for the valves.

E8.2 (Closed) VIO 50-261/98-03-03: failure to report peak cladding temperature (PCT) changes. The inspector reviewed licensee corrective actions resulting from the failure to report a significant change in peak cladding temperature as required by 10 CFR 50.46. The licensee reported the condition to the NRC in a letter dated October 16, 1997. Further, procedure REG-NGGC-0006, "Identification of Changes to (or errors in) LOCA Evaluation Models or Applications according to 10 CFR 50.46," Revision 2, was created to provide a consistent method for identification of changes or errors in the results of Loss-of-Coolant-Accidents (LOCA) Evaluation Models (EM) analysis for each of the CP&L nuclear units. Based on the review of this procedure, this item is considered closed.

E8.3 (Closed) Licensee Event Report (LER) 50-261/90-012-02: potential of inadequate net positive suction head (NPSH) for safety injection pumps. This LER supplement was submitted to the NRC on December 1, 1997. The issues related to inadequate NPSH associated with the Safety Injection (SI) pumps were discussed in NRC Inspection Reports 50-261/97-201 and 50-261/98-03, including those addressed by the LER. These inspections resulted in several enforcement actions, including that related to inadequate NPSH for the SI pumps. Based on the discussion of this issue in those reports, and the ensuing enforcement, this LER is considered closed.

IV. Plant Support

R1 Radiological Protection and Chemistry Controls

R1.1 General Comments (71750)

The inspectors periodically toured the Radiological Control Area (RCA) during the inspection period. Radiological control practices were observed and discussed with radiological control personnel including RCA entry and exit, survey postings, locked high radiation areas, and radiological area material condition. The inspector concluded that radiation control practices were proper. The inspector also toured the radwaste building and found that radwaste storage containers and laundry bags were in good condition and appropriately labeled. In addition, outside radwaste storage areas and structures were properly posted and exhibited correct labeling and effective housekeeping. The inspectors found that housekeeping throughout the plant was effective in maintaining areas free of unnecessary equipment and debris. Relatively few contaminated areas were noted and posted locked high radiation areas were properly secured against unauthorized entry.

R1.2 Radiological Protection and Controls

a. Inspection Scope (71750)

The inspectors observed radiological work practices and controls associated with the replacement of the Chemical and Volume Control System (CVCS) filters and the general radiological controls and conditions within the (RCA).

b. Observations and Findings

On August 17, the licensee replaced the CVCS filters. The filter housing is controlled as a high radiation area and is surrounded by a five foot high concrete wall. The filters are transferred to a shielded transport pig and transported to the radwaste bunkers in the radwaste building. Contact radiation levels on the exposed filters are as high as 3-4 rem/hr. Contact readings on the shielded transport pig once the filters were in place were as high as 300 mrem/hr. Radiological controls of the work including pre-job ALARA planning and work area access control were effective. Dose rate monitoring of the three mechanics performing the bulk of the work was performed using dosimeter telemetry. Total dose picked up by personnel during the job was 33 mrem. The licensee's controls to minimize dose during the filter replacement were effective.

The inspectors observed the work in progress and reviewed personnel and job dose records and concluded that the radiological controls for the job were effective in minimizing dose and the spread of contamination.

The inspectors also observed radiological controls within the Auxiliary Building. During the inspection the inspectors identified a "sharpie" type pen in the trough around the "B" charging pump. This area was controlled as a contaminated area. This condition was pointed out to the radiological control technician at the RCA access point. An Operation Experience (OE) item was disseminated during this time period concerning the use of "sharpie" pens on some metal piping. The ink can have a detrimental effect on certain alloys. The inspector did not find any piping in the charging pump room that appeared to have markings attributable to "sharpie" pens.

c. Conclusions

Radiological controls utilized during a Chemical and Volume Control System filter replacement were effective in minimizing personnel exposure and contamination

S1 Conduct of Security and Safeguards Activities

S1.1 General Comments (71750)

During the period, the inspector toured the protected area and noted that the perimeter fence was intact and not compromised by erosion or disrepair. Isolation zones were maintained on both sides of the barrier and were free of objects which could shield or conceal an individual. The inspector periodically observed personnel, packages, and vehicles entering the protected area and verified that necessary searches, visitor escorting, and special purpose detectors were used as applicable prior to entry. Lighting of the perimeter and of the protected area was acceptable and met illumination requirements.

V. Management Meetings

X1 Exit Meeting Summary

The inspectors presented the inspection results to members of licensee management at the conclusion of the inspection on September 18, 1998. The licensee acknowledged the findings presented. No proprietary information was identified.

PARTIAL LIST OF PERSONS CONTACTED

Licensee

T. Cleary, Manager, Operations
H. Chernoff, Supervisor, Licensing/Regulatory Programs
J. Clements, Manager, Site Support Services
R. Duncan, Manager, Robinson Engineering Support Services
J. Fletcher, Manager, Maintenance
J. Moyer, Director, Site Operations
R. Steele, Manager, Outage Management
R. Warden, Manager, Nuclear Assessment Section
T. Wilkerson, Manager, Regulatory Affairs
D. Young, Vice President, Robinson Nuclear Plant

NRC

B. Desai, Senior Resident Inspector
A. Hutto, Resident Inspector

INSPECTION PROCEDURES USED

IP 37551: Onsite Engineering
IP 61726: Surveillance Observations
IP 62707: Maintenance Observation
IP 71707: Plant Operations
IP 71750: Plant Support Activities
IP 92901: Followup - Operations
IP 92903: Followup - Engineering

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

Type	Item Number	Status	Description and Reference
NCV	50-261/98-08-01	Open	Missed TS SR 3.4.16.2, Reactor Coolant I-131 Dose Equivalent Verification (Section 04.1)

Closed

Type	Item Number	Status	Description and Reference
NCV	50-261/98-08-01	Closed	Missed TS SR 3.4.16.2, Reactor Coolant I-131 Dose Equivalent Verification (Section 04.1)
VIO	50-261/98-05-02	Closed	Failure to Perform Personnel Air Lock Testing (Section 08.1)
IFI	50-261/98-01-01	Closed	GL 89-10 Commitments. (Section E8.1)
VIO	50-261/98-03-03	Closed	Failure to Report PCT Changes (Section E8.2)
LER	50-261/90-012-02	Closed	Potential of Inadequate NPSH for Safety Injection Pumps (Section E8.3)