U.S. NUCLEAR REGULATORY COMMISSION

REGION II

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Report No:	50-261/97-08
Licensee:	Carolina Power & Light (CP&L)
Facility:	H. B. Robinson Unit 2
Location:	3581 West Entrance Road Hartsville, SC 29550
Dates:	June 8 through July 19, 1997
Inspectors:	B. Desai, Senior Resident Inspector J. Zeiler, Resident Inspector J. Coley, Region II Inspector F. Jape, Senior Project Manager
Approved by:	M. Shymlock, Chief, Projects Branch 4 Division of Reactor Projects

EXECUTIVE SUMMARY

H. B. Robinson Power Plant, Unit 2
NRC Inspection Report 50-261/97-08

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 inspections by a Region II based reactor safety inspector and a project engineer.

Operations

- The conduct of operations was professional and operators exhibited a safety-conscious attitude. Operator performance during a downpower to repair a service water leak on a heater drain pump oil cooler, as well as the identification of the leak by an auxiliary operator was good (Section O1.1).

- Operator identification of an error in the Over-Power Delta-Temperature (OPDT) setpoints was an example of good operator questioning attitude. The root cause of the incorrect setting of the OPDT low limit setpoint was determined to be an inadequate calibration procedure. The safety significance of this error was determined to be low since the OPDT trip functions were redundant to the Over-Temperature Delta-Temperature (OTDT) and high flux reactor trip functions and the OPDT trip was not taken credit for in the accident analysis. This issue was identified as an Unresolved Item (URI) until the NRC completes their review of this issue (Section O1.2).

- The licensee's actions in response to increased reactor containment building average air temperatures were adequate. Good engineering support was provided with the implementation of several innovative engineering initiatives to help maintain temperatures below design limits. While the licensee's root cause for the increased temperature was not completed, preliminary indications were that a modification to the containment cooling system may have adversely impacted the containment cooling efficiency. A URI was identified to review the licensee's root cause evaluation. In addition, the inspector noted that the Updated Final Safety Analysis Report (UFSAR) indicated that only three-out-of-four containment cooling fans were necessary to maintain containment air temperature below its design limit, when, four fans are normally operated and appear necessary to maintain temperature below its design limit during reactor operation (Section O1.3).

- The Plant Nuclear Safety Committee (PNSC) and the Nuclear Assessment Section (NAS) provided strong oversight and safety focus of licensee activities (Section O7.1).

Maintenance

- Observed maintenance activities were appropriately coordinated and conducted (Section M1.1).

- The Plant Manager demonstrated good leadership ability in daily morning operation/production turnover meetings and routinely was observed in the maintenance building and plant assessing plant operations, maintenance, and equipment condition (Section M2.2).

- Maintenance work observed on the "B" Charging Pump was performed in accordance with procedure by knowledgeable and skillful technicians (Section M1.2).

- Processes such as radiography, welding, and chemical injection for microbiological induced corrosion were performed well, and the condition of plant equipment was determined to be excellent (Section M2.1).

Engineering

- The licensee's identification that two Containment Isolation Valves were not adequately leak rate tested due to procedure inadequacies was indicative of a thorough engineering self-assessment. The failure to adequately test the valves in accordance with Technical Specification (TS) 4.4.2 was identified as a Non-Cited Violation (NCV) (Section E2.1).

  The licensee's engineering staff appropriately evaluated and resolved the issue related to secondary calculation of reactor power (Section E2.2).

- The "B" and "C" Safety Injection (SI) pumps were determined to be inoperable from approximately 1988 to 1997. This caused the plant to be outside design basis. Corrective actions were aggressive in rectifying the problem. The enforcement aspect of this issue will be dispositioned following issuance of the NRC's Architectural Engineering (A/E) Team inspection report (Section E2.3).

Plant Support

- Two incidents related to personnel entering the Radiation Control Area (RCA) without required Electronic Dosimetry monitoring were classified as an NCV. Continued licensee emphasis on corrective actions to prevent these types of incidents was warranted. The inspector also assessed that the Condition Report process continues to surface low-significant incidents (Section R1.2).

Report Details

Summary of Plant Status

Power was reduced from 100 percent to 55 percent on June 7-8, 1997, to conduct turbine valve testing. The unit operated at full power until July 17, 1997, when power was reduced to 60 percent to replace a heater drain pump oil cooler, which had developed a leak. During this downpower, a feedwater heater drain bypass valve to the condenser experienced problems. This resulted in several feedwater heater relief valves lifting. Following maintenance on this valve the unit was returned to full power on July 18, 1997 and remained at full power at end of 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 attended daily operation turnovers, management review, 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 (TSs). 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.

In general, the conduct of operations was professional and operators exhibited a safety-conscious attitude. Good plant equipment material conditions and housekeeping were noted throughout the report period. The performance by the non-licensed and licensed operators during the power changes, as well as the identification of the leak on the heater drain pump oil cooler was good. Other specific events and noteworthy observations are detailed in the sections below.

01.2 Overpower-Delta Temperature (OPDT) Setpoint Error

a. Inspection Scope (71707)

On June 10, 1997, with reactor power at 100 percent power, the licensee declared all three OPDT channels inoperable due to setpoint calibration errors. In accordance with 10 CFR 50.72 (b)(1)(ii)(B), the NRC was notified of a condition outside design basis (i.e., inoperability of three OPDT channels) and CR 97-01323 was initiated to investigate the problem. The inspector reviewed CR 97-01323 and the circumstances related to the calibration errors. LER 50-261/97-07-00 related to this issue was also submitted by the licensee to the NRC on July 9, 1997.

b. Observations and Findings

Control room operators noted that during a planned downpower on June 8, 1997, (for turbine control valve testing), the OPDT reactor trip setpoint had gone up by 1-2 degrees Fahrenheit (°F). However, no change in the OPDT reactor trip setpoint was expected for reactor power below 100 percent. The licensee's investigation of the issue identified that a circuit (M module) in the Hagan control system was not properly controlling the output voltage. This output voltage from the "M" module is fed to another circuit within the Hagan control system (G module) which calculates the OPDT reactor trip setpoint. Since the input to the setpoint calculation circuit (G module) was not properly controlled, the OPDT reactor trip setpoint was calculated higher (non-conservative) than required, for reactor power below 100 percent. This non-conservative setpoint was applicable to all three OPDT channels.

Reactor Engineering declared all three OPDT channels inoperable at 11:01 a.m. on June 10, 1997. The Limiting Condition for Operation (LCO) for TS, Table 3.5-2 item 6 requires a minimum of two channels out of the three channels operable when the reactor is critical. With all three channels inoperable, TS 3.0 was entered, which required the licensee to be in hot shutdown within eight hours. Further, the licensee notified the NRC of this condition which was outside of plant design basis.

Upon completion of the investigation, corrective actions were initiated. Procedures and calibration data sheet changes were implemented specifying the correct output voltage, including the low limit values for the "M" module. The OPDT channels were subsequently re-calibrated and satisfactorily tested. Channels 1 and 2 were restored to operability at 3:16 p.m, on June 10, 1997, allowing exit from TS 3.0. Channel 3 was restored to operability at 3:22 p.m. All three channels were restored to operable status within the allowed TS time limits and a plant shutdown was not initiated.

A review of calibration data sheets revealed that the low limit values for the "M" module were properly set from initial plant operation through 1976. However, there was no notation of this setting on calibration data sheets after that time. Review of temperature recording strip charts indicated that the OPDT setpoint had been performing properly up to Refueling Outage 6 (RFO-6) conducted in 1979. An out of tolerance condition had been noted on the strip charts during power transients after that outage. During RFO-6, reactor power was upgraded from 2200 MWT to 2300 MWT and calibration of the OPDT channels was completed. Administrative controls at that time permitted acceptance data to be annotated on the data sheet, and the sheet copied for use during subsequent calibration. The licensee's investigation concluded that the most probable cause for this condition was the failure to correctly transcribe calibration data from the previous calibration sheet to a new data sheet in the current calibration procedure. This resulted in the low limit value for the "M" modules not being properly adjusted.

c. Conclusions

The safety significance of this error in calibration is low. The OPDT trip function is required by TS, but is not utilized in FSAR Chapter 15 Safety Analysis and the OPDT reactor trip is not required to protect against Fuel Center Line Melt for an uncontrolled increase in reactor power. Incorrect setting of the low limit value did not affect the OPDT reactor trip setpoint for reactor powers at or above 100 percent power. The non-conservative OPDT setpoint was also evaluated by Seimens Corp. (fuel supplier and core designer) which reached a similar conclusion. The Overtemperature-Delta Temperature (OTDT) and the high flux reactor trip setpoints remained operable affording protection against power density in the core from exceeding the design limit of 118 percent following a postulated accident.

The inspector concluded that the licensee adequately evaluated, reported, and promptly corrected the issue and restored the system to operable condition. However, the failure to correctly calibrate the "M" module resulted in a condition outside the design basis of the plant. Pending further review by the NRC, this item will be considered as an URI. This item was documented as URI 50-261/97-08-01: Failure to Properly Calibrate the Low Limit Value affecting Reactor Trip Setpoint for OPDT.

01.3 Containment Air Temperature Exceeds Design Basis Limits

a. Inspection Scope (71707, 37551)

The inspector reviewed the circumstances involving the unexpected increase in reactor containment building air temperature experienced during the report period. The inspector monitored the licensee's actions when temperature increased above the containment design limit specified in the UFSAR, reviewed the licensee's corrective actions to address the overall temperature problem, and reviewed the licensee's progress toward determining the root cause of the problem.

b. Observations and Findings

In mid June, as outside atmospheric temperature increased, the licensee observed that containment average air temperature was increasing close to its UFSAR design limit of 120°F. While TS 4.4 indicates in the basis that the maximum temperature was 120°F with the reactor operating, there were no TS action requirements provided. In lieu of a TS action statement, the licensee implemented administrative requirements via an operation's Night Order requiring that a plant shutdown be commenced if temperature increased above 120°F and could not be restored below this value within 8 hours.

Due to this condition, an extra Service Water Pump was started to provide additional flow to the Containment fan coolers. The second Control Rod Drive Mechanism cooling and Containment Air Iodine Removal Exhaust fans were started to aid in containment air mixing. In

addition, a volumetrically weighted containment average air temperature model was developed and implemented under the modification process to provide a more accurate measure of the true bulk average air temperature in containment. This modification had previously been initiated in 1996 to support the addition of Improved TS surveillance requirements for measuring containment temperature. The inspector reviewed this model and concluded that it was a reasonable approach for more accurately measuring containment bulk average air temperature.

On July 5, 1997, at approximately 4:00 p.m., the weighted average calculation of containment air temperature increased to 120.51°F. In accordance with 10 CFR 50.72 (b)(1)(ii)(B), a one-hour notification was made to inform the NRC of a condition outside design basis. At 8:00 p.m., temperature decreased below 120°F averting the need for a plant shutdown. As a result of cooler prevailing weather conditions and the initiation of a containment purge to the outside environment, containment temperatures remained below the design limit.

During the week of July 6, the licensee implemented several innovative initiatives to help maintain containment temperature below 120°F. These initiatives included the following:

- Containment Dome Cooling: A fire hose was installed to the top of the containment outside structure which sprayed fire protection water from a fire hydrant near containment continuously on the outside containment dome area, this action did not adversely affect the overall fire protection system and,

- Transfer of cooler lake water to Service Water Intake: An engine driven water pump and associated piping was installed to transfer cooler water from the deep area of the lake to the service water intake. The Containment Air Recirculating Cooling System (CARCS) fans use service water as their cooling medium, therefore, providing cooler service water to the intake aids in the efficiency of these fans. The licensee planned to utilize this cooling mechanism only when temperatures appeared to be increasing to 120°F.

At the end of the report period, the licensee's initiatives were effective in preventing containment air temperature from increasing above design limits. The inspector noted that the licensee was closely monitoring temperature for any unexpected increase.

The licensee had not completed their evaluation of the cause of the unexpected increase in containment air temperature. However, preliminary investigations indicated that a modification to the CARCS, implemented during the last refueling outage, most likely was the greatest contributor. Modification Engineering Service Request (ESR) 95-00783 changed the position of CARCS normal and emergency intake dampers resulting in air flow distribution changes that may have adversely affected containment air temperature. The inspector determined that further review of this modification and its impact on

containment air temperature was necessary to determine the exact cause of the unexpected increase in containment air temperature.

The inspector reviewed the UFSAR description of the Reactor Containment Ventilation System Section 9.4.3.1 (Amendment 12) which indicated that one of the design functions of the system was to remove heat lost from equipment and piping in containment during normal plant operation and maintain a temperature of 120°F or less inside the containment, with 95°F cooling water and three-out-of-four CARCS cooling fans operating. The inspector noted that four fans are normally operated during power operation. Based on a preliminary review of containment temperature data, service water intake temperature, and CARCS performance data, the inspector was concerned whether the CARCS could meet its original design function with only three CARCS fans operating, even before the CARCS damper modification was implemented during the last refueling outage. The inspector discussed this concern with the licensee and determined that further review was necessary to resolve this issue and the cause of the increased temperature. Pending completion of this review, this will be identified as Unresolved Item (URI) 50-261/97-08-02: Review Root Cause of Increased Containment Air Temperature.

c. Conclusions

The licensee's actions in response to increased reactor containment building average air temperature were adequate. Good engineering support was provided with the implementation of several innovative engineering initiatives to help maintain temperature below design limits. While the licensee's root cause for the increased temperature was not complete, preliminary indications were that a modification to the containment cooling system may have adversely impacted the containment cooler's efficiency. A URI was identified to review the licensee's root cause evaluation. In addition, the inspector noted that the UFSAR indicated that only three-out-of-four CARCS cooling fans were necessary to maintain containment air temperature below its design limit, however, four fans are normally operated and appear necessary to maintain containment air temperature below its design limit during reactor operation. This will be reviewed as part of the URI.

07 Quality Assurance In Operations

07.1 Plant Nuclear Safety Committee and Nuclear Assessment Section Oversight

a. Inspection Scope (40500)

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

b. Observations and Findings

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

c. Conclusions

The inspector concluded that the onsite review functions of the PNSC were conducted in accordance with TSs. The PNSC meetings attended by the inspector were well coordinated and meetings topics were thoroughly discussed and evaluated. NAS continued to provide strong oversight of licensee activities.

08 Miscellaneous Operations Issues (92901, 92702)

08.1 (Closed) Violation (VIO) 50-261/96-011-01, Failure to Follow or Inadequate Procedures - Three Examples: The corrective actions presented in the licensee's responses, dated November 22, 1996 and January 23, 1997, were reviewed and verified by the inspector. The NRC accepted the response by letter dated December 19, 1996.

A general revision to Plant Program Procedure PLP-047, Foreign Material Exclusion Area Program, was completed as part of the resolution for example one of the violation. The revision was issued on January 30, 1997.

A revision to the training program for contract personnel has been initiated for use during the next refueling outage as part of the corrective action for example 2 of the violation.

Corrective action for example 3 of the violation required a revision to procedure EPRAD-03, Dose Projections. The revision provided instructions for assessing the computer program to calculate off-site dose projections and removed misleading statements. The revision, dated August 26, 1996 was validated by control room operators.

This violation is closed.

08.2 (Closed) VIO 50-261/96-011-02, Inadequate Corrective Action to Prevent Expired Fire Brigade Medical Physicals: The corrective actions presented in the licensee's response, dated November 22, 1996 and accepted by the NRC on December 19, 1996, were reviewed and verified as completed by the inspector.

Operations personnel, for each on-coming shift, are required to verify their qualification status prior to standing watch. The processes used

by corporate personnel for tracking medical examinations has been revised and were implemented at the site since March 3, 1997.

This violation is closed.

08.3 (Closed) Licensee Event Report (LER) 50-261/96-002-00, Condition Prohibited by TS Due to Inoperable Boric Acid Heat Trace: A clearance error resulted in removal of power from the B Boric Acid Transfer Pump heat trace circuits. This error resulted in a condition prohibited by TS. Immediately upon discovery, the error was corrected. An operations night order was issued to inform operators of actions to be taken when a discrepancy is detected while implementing a clearance request. This issue was also discussed in NRC Inspection Report 50-261/96-02 and was identified as an NCV.

The licensee conducted an investigation and completed corrective actions described within the LER. The inspector reviewed and verified the completion of the corrective actions. The action included real-time training, improved labeling of heat trace circuits and counseling of those persons involved.

This LER is closed.

08.4 (Closed) LER 50-261/96-005-00, Potential for Clogging Containment Spray Nozzle Due to Degraded ECCS Sump Screens: On September 11, 1996, the Emergency Core Cooling System (ECCS) sump screens was discovered to be degraded. Alterations and previous repairs failed to maintain the design requirements and configuration control standards. The sump was examined and cleaned of debris. The ECCS screens were restored to an acceptable functional and structural condition within the licensing basis of the system. Upon completion of restoration, the sump was inspected and found in a satisfactory condition. NRC inspection activities related to the sump repairs and restoration were reported in NRC Inspection Report 50-261/96-12, paragraph E8.5 as URI 96-12-08. The URI remains open pending further review by the NRC.

The licensee established procedural controls to prevent alteration of the screens without proper design control. The system was declared operable prior to restart from refueling outage 17.

This LER is closed.

08.5 (Closed) LER 50-261/96-006-00, Condition Prohibited by TS Due to Failure to Maintain Containment Integrity during Refueling Operations: During reload activities on October 4, 1996, containment integrity was temporarily lost due to maintenance activities on a penetration bellows. A vent line was opened in error. Immediate actions were taken to restore integrity. NRC inspections regarding this matter were reported in NRC Inspection Report 50-261/96-12, Section 01.2 and was identified as an NCV.

Corrective actions described in the LER were reviewed and verified by the inspector. The licensee reviewed this event with operations, maintenance and engineering personnel.

In addition, Operation personnel will review applicable procedures to determine if improvements are needed prior to their use during the next scheduled refueling outage.

This LER is closed.

II. Maintenance

M1 Conduct of Maintenance

M1.1 General Comments

a. Inspection Scope (61726 and 62707)

The inspector reviewed/observed all or portions of the following maintenance related Work Request/Job Orders (WRs/JOs) and/or surveillances and reviewed the associated documentation:

- WR/JO AKNV-02 "A" Station Battery Cell Temperature and Specific Gravity Measurements
- OST 704 Inservice Testing Containment Purge Valve Testing
- SP-1405 Containment Isolation Valve Leakage Test for RC-553

b. Observations and Findings

The inspector observed that these activities were performed by personnel who were experienced and knowledgeable of their assigned tasks. Work procedures were present at the work location and were utilized by the workers. Procedures provided sufficient detail and guidance for the intended activities. Activities were properly authorized and coordinated with operations prior to start. Test equipment in use was calibrated, procedure prerequisites were met, and system restoration was completed. Particularly noteworthy was the good work coordination and use of self check by Instrumentation and Electrical technicians measuring the specific gravity of the "A" station battery cells.

c. Conclusions

The inspectors concluded that routine maintenance activities were performed satisfactorily.

M1.2 Charging Pump Corrective Maintenance Implementation

a. Inspection Scope (62700)

The chemical and volume control system (CVCS) was designed to regulate the primary coolant chemistry, to provide both reactivity and corrosion control, and to maintain primary system inventory during normal plant operations.

On June 18, 1997, the inspector reviewed documentation and observed work activities for the "B" CVCS pump which consisted of the replacement of the position actuator, the air regulator, associated tubing, and calibration of the speed controller. This work was performed because operations personnel had observed fluctuations in charging flow from the "B" CVCS Pump which would decrease from 32 to 27 gpm and then increase back to 32 gpm. This level of cycling would repeat in about four minute intervals, when the pump was placed in manual operation the cycling stopped. This risk significant activity was examined to verify that corrective maintenance activities were conducted in a manner which would result in the reliable and safe operation of the plant.

b. Observations and Findings

Maintenance work on the "B" charging pump was conducted in accordance with Work Request Nos. 97-ABKJ1 and 97-ADUI1. The inspector verified that maintenance personnel assigned were qualified and understood the scope of the task, TS were met, test equipment was calibrated, engineering and supervisory over-sight was adequate and post maintenance testing was delineated.

The inspector noted that post-maintenance testing adequately demonstrated acceptable pump performance and the fluctuations in flow had been corrected. However, during this testing, the operators observed that the pump was controlling at a higher demand signal versus the charging flow output than before. Engineering personnel determined that this difference was the result of the speed controller linkage not being set properly. This problem would not impact pump performance and response. A work request was initiated to adjust the linkage during the next scheduled preventive maintenance on the pump. In addition, the positioner calibration procedure was to be revised to ensure that the speed controller linkage was properly set during future speed controller maintenance. The inspector determined that the licensee had adequately resolved this minor procedure weakness.

c. Conclusions

Maintenance work observed on the "B" Charging Pump was performed in accordance with procedure by knowledgeable and skillful technicians.

From June 17 through June 20, 1997, the inspector attended the morning operation/production meeting to determine the status of work performed. The inspector noted that the plant manager was knowledgeable of the

ongoing work activities, the impact of the work on plant equipment, and was active in ensuring proper work priorities. The plant manager was also observed in the maintenance building and other areas of the plant assessing plant operations, maintenance, and equipment condition.

M2 Maintenance and Material Condition of Facilities and Equipment

M2.1 Examination of Processes and Completed Maintenance

a. Inspection Scope (62700 and 57090)

The inspector reviewed documentation and observed completed work activities consisting of the review of radiographic film for safety related welds, review of the microbiological induced corrosion (MIC) monitoring program, conducted a walkdown inspection of the chlorination facility, reviewed completed work requests for the dedicated diesel, observed completed welding on a support for the Refueling Water Storage Tank (RWST) fill line, and observed the general condition of pumps, diesels, and batteries in order to evaluate equipment and material condition of the Robinson facility.

b. Observations and findings

The inspector reviewed radiographs for safety related pipe weld No. SV1-4C-W-1 and electrical penetration weld No. C-3 which had been welded during Refueling Outage 17. These welds were radiographed in accordance with the 1996 edition of Section V, to the American Society of Mechanical Engineers (ASME) Code, and Carolina Power and Light Procedure No. NDE-101, Revision 15. The radiographic techniques used and quality of the radiographs were good. The quality of the welds radiographed was also very good.

The inspector held discussions with the service water system engineer, reviewed procedures, drawings of replaced piping, and chemical monitoring documentation, and conducted a walkdown inspection of the chlorination facility to determine the status of MIC at the Robinson facility. In the mid 80's MIC had been a significant problem at this plant. However, the inspector found that with chemical injection of chloride and improved piping materials, the licensee has reduced MIC to the point that no failure has occurred since 1989. This was judged as very good performance.

The dedicated diesel is a critical component for the Robinson facility; therefore, the inspector reviewed the preventive maintenance performed on this equipment. In addition, completed work requests since June 1, 1996 were reviewed to determine the scope of problems experienced. This review revealed that this equipment has performed satisfactory and is well maintained with only minimum problems reported.

The inspector observed completed welding on a pipe support for the fill line on the RWST. This activity was judged as a high quality weld of appropriate size for installation of the support.

The inspector also conducted inspections of various plant equipment on several occasions to determine the condition of plant equipment such as pumps, diesels, and batteries. The condition of plant equipment was excellent.

c. Conclusions

Processes such as radiography, welding, and chemical injection for MIC were performed well, and the condition of plant equipment was also judged to be excellent.

M8 Miscellaneous Maintenance Issues (92702, 92902)

M8.1 (Closed) VIO 50-261/97-004-02, Failure to Follow Procedure SPP-002 for Properly Monitoring and Logging Freeze Pack Temperatures: The corrective actions described in the licensee's response, dated May 16, 1997, and accepted by the NRC on June 11, 1997, were reviewed and verified by the inspector as completed. Immediately upon notification of the temperature logging requirement stated in procedure SPP-002, Rev. 10, the maintenance personnel initiated recording of the temperature. The work crew supervisor discussed this event with those involved to re-emphasize the need to follow instructions. The Maintenance supervisor also reviewed his expectations of the craft personnel. Further, the manager of maintenance counseled the supervisor regarding good work practices and use of procedures.

This violation is closed.

III. Engineering

E2 Engineering Support of Facilities and Equipment

E2.1 Inadequate Local Leak Rate Test of Two Containment Isolation Valves

a. Inspection Scope (37551, 40500, 61726 and 62707)

During an engineering self-assessment of the 10 CFR 50, Appendix J, Local Leak Rate Test program, the licensee identified that two Containment Isolation Valves (CIVs) had not been properly leak rate tested in accordance with TS 4.4.3.a during the last refueling outage conducted during September 1996. The licensee initiated CR 97-01458 to address the problem. The inspector reviewed the licensee's immediate corrective actions upon identification of the discrepancy and subsequent corrective actions to test the valves and evaluate the root cause of the problem.

b. Observations and Findings

Between July 2-4, 1997, the licensee identified that CIVs RC-553 and WD-1794 had not been adequately leak rate tested as a result of inadequate test vent paths specified in Operations Surveillance Test (OST)-933, Containment Isolation Valves Leakage Test, Revision 3. Check valve RC-

553 is the inboard CIV from the Pressurizer Relief Tank to the Gas Analyzer and check valve WD-1794 is the inboard CIV from the Reactor Coolant Drain Tank sample line to the Gas Analyzer. The test vent path specified in OST-933 aligned the valve leakage flowpath through a pressure regulator valve and then to their respective tanks which were vented to atmosphere. However, the pressure regulator was not designed to allow flow in the direction that would be required should the CIVs leak by. Without an open vent path, the leak rate testing was considered inadequate.

The inspector verified that appropriate actions were implemented upon identification of the test discrepancies. In accordance with TS 3.6.3 within four hours of declaring RC-553 and WD-1794 inoperable, the outboard CIVs of both penetrations were closed and instrument air to the valve actuators isolated.

On July 11, 1997, RC-553 and WD-1794 were successfully leak rate tested in accordance with special procedures (SP)-1405 and SP-1406, respectively. The inspector reviewed the procedures and determined that they provided good detail and controls for ensuring that a proper leakage test was performed. Leakage test results for each valve were measured to be well within acceptable limits.

The licensee determined that the inadequate test vent path alignment for these valves had existed since 1987 when OST-933 was first developed. At that time, it was assumed that the pressure regulator valve would allow flow in either direction. However, the self-assessment questioned this alignment and determined that the pressure regulator would not pass flow in the test vent direction. The licensee performed a complete review of OST-933 to determine if similar problems existed; no further problems were identified.

c. Conclusions

The inspector concluded that the licensee's identification of this issue was indicative of a thorough engineering self-assessment of their Appendix J program implementation. The failure to adequately leak rate test CIVs RC-553 and WD-1794 in accordance with the requirements of TS 4.4.2.a was identified as a violation. This licensee-identified and corrected violation is being treated as a NCV, consistent with Section VII.B.1 of the NRC Enforcement Policy. This issue was documented as NCV 50-261/97-08-03: Inadequate Procedure for Leak Rate Testing Containment Isolation Valves RC-553 and WD-1794.

E2.2 Secondary Calorimetrics Power Calculation

a. Inspection Scope (37551)

The inspector reviewed licensee methodology involving secondary calorimetric based power calculation.

b. Observations and Findings

During review of redundant indications of reactor power, the licensee noted that the feedwater flow calorimetric was indicating a small increase in reactor power during the period March-May 1997, while the steam flow calorimetric indicated no change in reactor power. A further review of the data indicated that the turbine first stage impulse pressure and feedwater temperature indicated similar trends.

A primary heat balance power calculation indicated an increase of 0.13 percent over the same period and similarly, the steam flow calorimetric power increased by 0.02 percent and the feedwater flow based calorimetric indicated a reactor power increase of 0.25 percent over the same period.

Due to the difference in indicated power, the licensee initiated a CR and conservatively limited power to that indicated on the higher feedwater flow based calorimetric.

An ESR evaluated the condition and concluded that the conflicting trends between the feedwater flow and steam flow calorimetric were attributed to several factors. These included: ambient temperature changes, containment temperature changes, and power reductions. The licensee verified that the process instruments were calibrated for the expected variances in the factors discussed above. Based on this, the licensee concluded that the feedwater and main steam flow based calorimetric were both accurate and the variances were within the instrument accuracies.

The licensee subsequently chose to utilizing the main steam flow based calorimetric.

c. Conclusions

The inspector concluded that the licensee staff appropriately evaluated and resolved the issue related to secondary calculation of reactor power.

E2.3 Safety Injection/Residual Heat Removal Net Positive Suction Head Issues

a. Inspection Scope (37551)

As a result of questions raised by the NRC Architectural Engineering (AE) team inspection that was conducted in April - May, 1997, the licensee concluded that the "B" and "C" Safety Injection (SI) pumps had inadequate net positive suction head to fulfill their safety related function. This issue was documented by the licensee in significant CR 97-01217. The inspector reviewed licensee's actions related to this issue. Further AE report will also discuss this issue.

b. Observations and Findings

Introduction:

During licensee investigation of the basis for the Safety Injection (SI) and Residual Heat Removal (RHR) pump net positive suction head (NPSH) availability as a result of NRC AE team questions in April - May, 1997, it was determined that original calculation assumptions were not consistent with current system response. The original Westinghouse calculations of record dated 1967 through 1971 assumed flow rates and piping resistance values inconsistent with current system operating configuration. These values are used to determine available NPSH and define the RWST water level necessary to provide acceptable SI pump operating conditions. The licensee subsequently hired a contractor (Altran) to develop a new ECCS model based on new calculations. New SI system NPSH calculations dated July 10, 1997 (hereafter referred to as the new NPSH calculation) indicated that available NPSH was inadequate for SI pumps "B" and "C" operation at the original RWST low and low-low (27 and 9 percent) levels for the required emergency operating configurations. Consequently, the licensee performed a modification and raised the RWST level such that all three SI pumps now have adequate NPSH for all postulated ECCS pump configurations.

The modification involved increasing the RWST water level by approximately 11% (3.5 feet) from the original values per ESR 97-00307. The increase was gained by a combination of raising the water level by 9.5% and reducing the instrument uncertainty, which added approximately 1.5%. The RWST level increases were followed by rescaling of the instruments, such that the change was transparent to the operators as well as procedures. With the increased height of water, the new NPSH calculation data indicates acceptable NPSH available for SI pumps "A", "B" and "C".

The licensee also plans to develop a plan for Safety System Functional Inspection and A/E type assessments of other systems which includes an evaluation of system modifications and related design calculations for potential inadequacies.

c. Conclusions

As a result of questions raised by the AE team, the licensee developed a new ECCS flow model. Subsequently, the licensee implemented a modification to raise the RWST level. This ensured adequate NPSH for the SI pumps. The issue will be further discussed in the AE inspection report.

E7 Quality Assurance in Engineering Activities

E7.1 Special UFSAR Review

A recent discovery of a licensee operating their facility in a manner contrary to the UFSAR description highlighted the need for a special

focused review that compares plant practices, procedures and/or parameters to the UFSAR descriptions. While performing the inspection discussed in this report, the inspector reviewed selected portions of the UFSAR that related to the areas inspected. The inspector verified that for the select portions of the UFSAR reviewed, the UFSAR wording was consistent with the observed plant practices, procedures and/or parameters.

E8 Miscellaneous Engineering Issues (92702, 92903)

E8.1 (Closed) VIO 50-261/95-007-01, Failure to Consider the Impact of a Plant Configuration Change on Technical Specification 4.4.1.2 Leakage Limit: The corrective actions presented in the licensee's response dated May 9, 1995, and accepted by the NRC on May 25, 1995 were reviewed and verified by the inspector as completed. The Robinson Engineering Support Section was provided with real time training on this occurrence on March 30, 1995. In addition, the licensee issued revision 14 to Technical Management Manual (TMM)-005 on April 4, 1997 to provide guidance on the required action to be taken when isolating the Penetration Pressure System from containment penetrations.

This violation is closed.

IV. Plant Support

R1 Radiological Protection and Chemistry Controls (71750)

R1.1 Tours of the Radiological Control Area (RCA)

The inspectors periodically toured the 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 conditions. The inspectors concluded that radiation control practices were proper.

R1.2 Personnel Entering RCA Without Electronic Dosimetry

a. Inspection Scope

The inspector reviewed circumstances surrounding two entries that were made by licensee employees without electronic dosimetry (ED).

b. Observations and Findings

On June 16, 1997, an operator entered the RCA without an ED. The operator was in the log-in process at which time he was interrupted to answer a radio call. He left the log-in station without removing the ED from the log-in computer module and entered the RCA. The operator was wearing his Thermo Luminescent Dosimeter (TLD).

The operator spent approximately 15 minutes in the RCA, most of which was spent in the Auxiliary Operator (AO) office. The operator was reminded by another operator that his ED was missing. The operator subsequently contacted Health Physics (HP), and a condition report was initiated. HP assessed that the dose received by the operator during the 15 minutes was minimal, and that this was monitored by the TLD.

As corrective action the licensee installed a swinging gate at the RCA boundary, with a sign reminding people to ensure they had proper monitoring. Additionally, the licensee is considering installation of a turnstile, which will allow entry following positive verification of appropriate monitoring.

On June 23, 1997, a planner entered the RCA to review a work package at the RWST. After approximately five minutes, when returning to log out, he noticed that the ED he was wearing read "pause", as if log-in had not been done. The planner was wearing his TLD. The licensee assessed that the planner most likely removed the ED prematurely during the log-in process. Thus, it was never activated. The planner was counseled with regard to the error.

The inspector reviewed the circumstances surrounding a past event that occurred on May 8, 1997 and discussed in NRC 50-261/97-07. This event involved an engineer entering the RCA without being monitored (i.e. no ED or TLD was worn). However, in the two incidents that are discussed above, the individuals were monitored by the TLD. Notwithstanding, the failure to have an ED as required by the applicable Radiation Work Permit, is considered a violation. This violation will be classified as non-cited consistent with Section VII.B.1 of the NRC Enforcement Policy. Licensee corrective actions were prompt and they included installation of a gate at the entrance to the RCA. The inspector also determined that the licensee's CR process was effective in surfacing low-significant events, that potentially would not be identified. This is allowing the licensee to focus on corrective actions, before the issue becomes more significant.

c. Conclusions

Two incidents related to personnel entering RCA without required ED monitoring were classified as NCV 50-261/97-08-04: Failure to Wear Electronic Dosimeter in RCA. Continued licensee emphasis on preventing these types of incidents is warranted. The inspector also assessed that the CR process continues to surface low-significance incidents.

S1 Conduct of Security and Safeguards Activities (71750)

S1.1 General Comments

During the period, the inspector toured the protected area and noted that the perimeter fence was intact and not compromised by erosion nor 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.

S8 Miscellaneous Plant Support Issues (92904)

S8.1 (Closed) VIO 95-018-01, Failure to Protect Safeguards Information: The corrective actions presented in the licensee's response, dated July 24, 1995, and accepted by the NRC on July 22, 1995, were reviewed and verified by the inspector as being completed.

This violation is closed.

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 July 28, 1997. No proprietary information was identified.

PARTIAL LIST OF PERSONS CONTACTED

Licensee

J. Boska, Manager, Operations
H. Chernoff, Supervisor, Licensing/Regulatory Programs
T. Cleary, Manager, Maintenance
J. Clements, Manager, Site Support Services
D. Crook, Senior Specialist, Licensing/Regulatory Compliance
J. Keenan, Vice President, Robinson Nuclear Plant
G. Miller, Manager, Robinson Engineering Support Services
R. Moore, Manager, Outage Management
J. Moyer, Manager, Robinson Plant
D. Stoddard, Manager, Operating Experience Assessment
R. Warden, Manager, Nuclear Assessment Section
T. Wilkerson, Manager, Regulatory Affairs
D. Young, Director, Site Operations

NRC

B. Desai, Senior Resident Inspector
J. Zeiler, Resident Inspector

INSPECTION PROCEDURES USED

IP 37551:	Onsite Engineering
IP 40500:	Effectiveness of Licensee Controls in Identifying, Resolving, and Preventing Problems
IP 57090:	Pipe Support and Restraint Systems
IP 61726:	Surveillance Observations
IP 62700:	Maintenance Implementation
IP 62707:	Maintenance Observation
IP 71707:	Plant Operations
IP 71750:	Plant Support Activities
IP 92702:	Followup - Corrective Actions for Violations
IP 92901:	Followup - Operations
IP 92902:	Followup - Maintenance
IP 92903:	Followup - Engineering
IP 92904:	Followup - Plant Support

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

Type	Item Number	Status	Description and Reference
URI	50-261/97-08-01	Open	Failure to Properly Calibrate the Low Trip Setpoint for OPDT (Section O1.2)
URI	50-261/97-08-02	Open	Review Root Cause of Increased Containment Air Temperature (Section O1.3)
NCV	50-261/97-08-03	Open	Inadequate Procedure for Leak Rate Testing Containment Isolation Valves RC-553 and WD-1794 (Section E2.1)
NCV	50-261/97-08-04	Open	Failure to were Electronic Dosimeter in RCA (Section R1.2)

Closed

Type	Item Number	Status	Description and Reference
VIO	50-261/96-011-01	Closed	Failure to Follow or Inadequate Procedures - Three Examples (Section O8.1)
VIO	50-261/96-011-02	Closed	Inadequate Corrective Action to Prevent Expired Fire Brigade Medical Physicals (Section O8.2)
LER	50-261/96-002-00	Closed	Condition Prohibited by TS Due to Inoperable Boric Acid Heat Trace (Section O8.3)

LER	50-261/96-005-00	Closed	Potential for Clogging Containment Spray Nozzle Due to Degraded ECCS Sump Screens (Section O8.4)
LER	50-261/96-006-00	Closed	Condition Prohibited by TS Due to Failure to Maintain Containment Integrity during Refueling Operations (Section O8.5)
NCV	50-261/97-08-03	Closed	Inadequate Procedure for Leak Rate Testing Containment Isolation Valves RC-553 and WD-1794 (Section E2.1)
VIO	50-261/97-004-02	Closed	Failure to Follow Procedure SPP-002 for Properly Monitoring and Logging Freeze Pack Temperatures (Section M8.1)
VIO	50-261/95-007-01	Closed	Failure to Consider the Impact of a Plant Configuration Change on Technical Specification 4.4.1.2 Leakage Limit (Section E8.1)
NCV	50-261/97-08-04	Closed	Failure to Wear Electronic Dosimeter in RCA (Section R1.2)
VIO	50-261/95-018-01	Closed	Failure to Protect Safeguards Information (Section S8.1)