U.S. NUCLEAR REGULATORY COMMISSION

REGION II

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License No:	DPR-23
Report No:	50-261/98-02
Licensee:	Carolina Power & Light (CP&L)
Facility:	H. B. Robinson Unit 2
Location:	3581 West Entrance Road Hartsville, SC 29550
Dates:	February 15 - March 28, 1998
Inspectors:	B. Desai, Senior Resident Inspector N. Economos, Region II Inspector (Sections M1.3, M1.4, M1.5, M1.6) J. Lenahan, Region II Inspector (Section E2.1) F. Jape, Region II Inspector (Section E1.2, E1.3)
Accompanying Personnel:	J. Shea, NRR Project Manager
Approved by:	M. Shymlock, Chief, Projects Branch 4 Division of Reactor Projects

Enclosure

EXECUTIVE SUMMARY

H. B. Robinson Power Plant, Unit 2
NRC Integrated Inspection Report 50-261/98-02

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 Region II based reactor safety inspectors.

Operations

- The conduct of operations was professional, risk informed, and safety-conscious. An issue related to logging of Technical Specification action statement identified by the inspector was promptly corrected (Section O1.1).

- Overall, activities related to the shutdown of Unit 2 for refueling outage 18 were conducted in a safe and deliberate manner (Section O1.2).

- Reactor coolant system draindown activities were appropriately conducted. A non-cited violation for not having both residual heat removal trains operable in Mode 6 was identified (Section O1.3).

- The Reactor Coolant System draindown and overflow of approximately 200 gallons of water onto the containment floor exhibited a weakness on part of operations in the planning of a surveillance test. Further, the normal level of attention to detail was not exhibited by operations that contributed to this event (Section O1.4).

- Nuclear Assessment Section and Plant Nuclear Safety Committee continued to provide strong oversight, including during refueling outage 18 (Section O7.1).

Maintenance

- Maintenance and surveillance activities were performed satisfactorily. The inspector noted good controls of housekeeping and good supervisor oversight of work activities (Section M1.1).

- Licensee response to the Emergency Diesel Generator problems resulting form a raw water pressure switch during the 24 hour surveillance test were appropriate. However, a more detailed and thorough investigation after the first problem (reverse power trip) could have potentially identified the failure of the stopping relay (Section M1.2).

- Inservice inspection (ISI) activities observed were performed following approved procedures and applicable code requirements. Technicians were well trained and qualified to perform the assigned ISI examinations. The licensee's evaluations of unacceptable field conditions were evaluated and dispositioned with conservatism (Section M1.3).

- The eddy current examination of SG tubes during this outage was well managed and executed following approved plans, procedures and industry

guidelines. Technical personnel were well qualified to perform their assigned tasks. The eddy current inspection plan and the results attained met code and industry standards (Section M1.4).

- The piping modification to increase the net positive suction head (NPSH) margin available to "B" and "C" safety injection (SI) pumps was implemented following design specifications and applicable code requirements. Fabrication of acceptable welds was accomplished with some degree of difficulty as evidenced by numerous weld rejections. Radiographs of the new welds exhibited film artifacts and the sensitivity on a number of films showed a need for improving the shooting technique and film developing practices. These observations reflected a weakness in the areas of welding and radiography (Section M1.5)

- Records of Flow Accelerated Corrosion Testing that were reviewed were found to be complete and accurate. Isolated areas with below code minimum thickness were identified in Steam Generator (SG) "A" Feedwater nozzle to pipe reducer. These areas were properly evaluated by engineering and accepted for an additional fuel cycle (Section M1.6).

Engineering

- Licensee appropriately evaluated the condition involving missing reactor coolant pump diffuser cap screws that was identified during the outage. The evaluation included the potential impact of the loose parts that were created by virtue of the missing cap screws on the reactor coolant system pressure boundary as well as fuel integrity as well as operating the reactor coolant pump with less than 16 diffuser cap screws (Section E1.1).

- Modifications to enhance motor operated valve performance were well planned and executed (Section E1.2).

- Activities for correcting the instrumentation sensing line slope were performed in a technically and administratively acceptable manner (Section E1.3).

- The generator rotor retaining ring modification, including the lifting of the rotor, which constituted a "heavy load" was appropriately accomplished (Section E1.4).

- The licensee's program for inspection of the containment vessel liner was effectively implemented. The results of the inspections showed that the liner met design requirements (Section E2.1).

- Licensee was noted to be appropriately tracking and addressing the partial control rod cracking issue that was observed at another site. Long term actions will be developed through efforts led by Westinghouse Owners Group (Section E8.1).

Plant Support

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

Report Details

Summary of Plant Status

Robinson Unit 2 operated at full power until March 1, 1998, at which time a coastdown was initiated to refueling outage (RFO)-18. On March 7, 1998 the unit was shutdown and subsequently cooled down. The core was off-loaded to the spent fuel pool between March 14-17, 1998. Core reload was commenced on March 27, and as of March 28, core reload activities were in progress.

I. Operations

O1 Conduct of Operations

O1.1 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 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 (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. Good plant equipment material conditions and housekeeping continued to be observed throughout the report period.

In general, the conduct of operations was risk informed and professional.

O1.2 Unit 2 Coastdown/Shutdown for Refueling Outage-18

a. Inspection Scope (71707)

The inspector monitored and reviewed licensee activities related to Unit 2 coastdown and shutdown for RFO-18.

b. Observations and Findings

Plant coastdown on Unit 2 was started on March 1, 1998. At 8:00 p.m., on March 6, 1988 shutdown of Unit 2 was initiated from approximately 95 percent power. The unit was taken off-line at 11:22 p.m., and mode 3 was entered at 12:11 a.m, on March 7, 1988. All monitored shutdown activities were performed in accordance with approved procedures and

excellent command, control, and communications were noted. Reactor Coolant System (RCS) cooldown limits were maintained within requirements. Adequate control room staffing and management oversight was noted. All shutdown activities were uneventful with the exception discussed below.

At 1:38 p.m., while in mode 4, a feedwater isolation signal was received. The licensee was in General Procedure (GP)007, Plant Cooldown From Hot Shutdown to Cold Shutdown, which required the cycling of feed header section valves (feedwater block valves) V2-6A, 6B, and 6C, in order to preclude thermal binding. One main feedwater pump(MFP) as well as both condensate pumps were running at the time. When valve V2-6A was cycled, the level in the "A" SG increased, indicative of leakage past the main feedwater control valve, which were closed. An auxiliary operator (AO) was dispatched to locally ensure that valve V2-6A was fully closed. At 70 percent SG level, the running MFP was secured; however, the SG level continued to increase as the condensate pumps continued to run, discharging at a higher pressure (550 psig) than the SG pressure (100 psig). A feedwater isolation signal was generated when the "A" SG level reached 75 percent. Since both MFPs, all feedwater isolation, and bypass valves were already shut, no valve repositioning occurred as a result of the feedwater isolation.

The licensee initiated CR 98-00509 to track the problem and initiate appropriate corrective actions. Current licensee plans are to revise GP-007 to ensure that the MFPs are secured before cycling the feed header section valves. The licensee also reviewed the problem for reportability and concluded that it was not reportable. Even feedwater isolation, initiated by an SI, is required only in modes 1, 2, and 3 per TS 3.3-28. The feedwater isolation signal on March 7 was due to high SG level and is not credited in the safety analysis, and was not initiated by SI.

c. Conclusions

The inspector concluded that overall, Unit 2 shutdown activities for RFO-18 were conducted in an excellent manner.

01.3 RCS Draindown/Reflood and Core Off-Load Activities

a. Inspection Scope (71707)

The inspector monitored licensee activities during RCS draindown, reflood, and core off-load.

b. Observations and Findings

RCS drain down was commenced on March 10 to allow reactor vessel head disassembly. During the draindown, the available level indications including Pressurizer, Reactor Vessel Level Indication System (RVLIS), and the designated level instruments LT-403 and LT-404 tracked well without any significant difficulties. RCS was drained to approximately

-9 inches on LT-403 and LT-404 to accommodate reactor vessel head detensioning. This level corresponds to 9 inches below the lip of the reactor vessel flange. The licensee did not enter reduced inventory (-36 inches) or midloop (-67 inches) operations. The draindown activities were conducted in accordance with procedure GP-008, Draining the Reactor Coolant System. Following head and upper internals removal, the refueling cavity was flooded and core was completely off-loaded to the spent fuel pool.

The inspector reviewed licensee Outage Management Procedures (OMP)-004, Outage Risk Assessment, and OMP-003, Shutdown Safety Function Guidelines. OMP-003 provides the guidelines and requirements for ensuring safety during RCS drain down and reduced inventory conditions. The inspector determined that, with the exception described further in this section, the licensee appropriately conducted activities in accordance with OMP-003, including, assurance of Containment closure capability, RCS Temperature monitoring, RCS level monitoring, and RCS inventory control and makeup capability.

TS 3.9.5, Residual heat Removal (RHR) and Coolant Circulation -Low Water Level, required two RHR trains to be operable and one RHR train to be in operation in Mode 6, with water level less than 23 feet above the top of reactor vessel flange. TS defines a train operable when it is capable of performing its intended function and when all necessary attendant instrumentation, controls, and normal or emergency power are also capable of performing their intended function. Further, Bases for TS 3.7.6 and TS 3.7.7 state that the Component Cooling Water (CCW) and Service Water (SW) system operability requirements in Modes 5 and 6 are determined by the system they support. Since the SW and CCW systems support the RHR system, both trains of the SW as well as CCW systems were also required to be operable.

On March 13, 1998, a unit supervisor (Senior Reactor Operator) questioned the operability of the "A" RHR train. At this time, the "A" RHR pump was in service. However, the "A" and "B" SW pumps were out-of-service and under a clearance. "A" and "B" SW pumps support the "A" RHR train and were inoperable for approximately 53 hours from 4:43 a.m., March 11, 1998 to 11:37 a.m., March 13, 1998. With one train of RHR inoperable, TS 3.9.5 action statements requires actions to be initiated immediately to establish greater than 23 feet of water above the top of the reactor vessel flange. Licensee failure to meet the requirements of this action statement on March 11, 1988 with the "A" RHR train inoperable constituted a violation. At the time the issue was raised by the unit supervisor on March 13, 1988, cavity floodup was in progress.

The licensee initiated a significant CR and concluded that the root cause of the event was related to the recent change to improved TS and the failure of the outage support unit to fully understand the requirements of the improved TS. The outage support unit develops the Shutdown Safety Function Status (SSFS) which identified components protected from clearances and work performance to ensure shutdown

safety. The SSFS is updated every day and is extensively utilized by the plant, including operators.

Licensee corrective actions were to issue a Night Order to clarify the RHR train requirements, and revise OMP-003 to specify RHR train requirements for Mode 6. Additionally, the licensee plans to review the event for applicability to the schedule for RFO-19. This event was also determined to be reportable to the NRC pursuant to the requirements of 10 CFR 50.73(a)(2)(I)(B). The failure to comply with the requirements of TS 3.9.5 constitutes a violation. However, based on licensee identification and prompt corrective action, this violation will be classified as Non-Cited Violation (NCV) 98-02-01: Failure to Have Two RHR Trains Operable in Mode 6.

c. Conclusion

The inspectors concluded that drain down activities as well as activities with RCS at approximately -9 inches were conducted in a deliberate and controlled manner, with close monitoring of key parameters such as RCS level and temperature. A non-cited violation for not having both the RHR trains operable in Mode 6 was identified.

01.4 Containment Activities Following Core Off-Load

a. Inspection Scope (71707)

The inspectors reviewed an incident involving inadvertent draining of approximately 3000 gallons of water from the Refueling Water Storage Tank (RWST); approximately 200 gallons of this water spilled onto the containment floor. This incident occurred when there was no fuel in the core.

b. Inspection Findings

During the RCS draindown following completion of core off-load activities, approximately 3000 gallons of water was unexpectedly gravity drained from the RWST through the Chemical and Volume Control System (CVCS) charging line into the RCS loops. Approximately 200 gallons spilled out the SG primary side manways and onto the containment floor. The problem was caused when a normally closed valve, CVC-358, bypass suction to Charging Pumps, was caution tagged open for the outage to maintain a boration flowpath from the RWST to the RCS. Concurrently, during the performance of Operations Surveillance Test (OST-933, Containment Isolation Valve Leakage Test, charging line discharge valve CVC-121 was closed for the purpose of the OST. Following completion of OST-933, CVC-121 was opened under the assumption that CVC-358 was closed. With CVC-121 and CVC-358 open, a flowpath was created from the RWST to the containment sump.

The licensee requested Nuclear Assessment Section (NAS) to independently review the incident. NAS concluded that several factor contributed to the incident. Most notably, the planning associated with the

performance did not factor the open position of CVC-358. During previous outages CVC-358 was maintained closed. Thus OST-933 assumed status-quo. However, during this outage, CVC-358 was maintained open as a boration flow path. The affect of the change was not recognized by operations. Secondly, during the time when RWST was being drained to the RCS, the RCS loop levels were noted to be increasing on the available indications. However, operations assumed that the indications were in error and therefore did not adequately respond to the increasing loop levels.

As a result of this incident, the licensee had a "all-hands" briefing stressing the importance of good operating practices. The spill did not result in any contamination incidents.

c. Conclusions

The inspector concluded that the incident exhibited a weakness on the part of operations in the planning of OST 933. Further, the normal level of attention to detail was not exhibited by the operators, perhaps due to the core being unloaded.

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 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, including during RFO-18.

II. Maintenance

M1 Conduct of Maintenance

M1.1 General Comments

The inspector reviewed/observed all or portions of the following maintenance related work requests/job orders (WRs/JOs) and/or surveillances and reviewed the associated documentation:

- OST-410 EDG A 24 hour Test
- OST-411 EDG B 24 hour Test
- LP-702 Source Range Loop Calibration
- MST-921 Battery Test
- CM-103, WR/JO 97AHCJI Pressure Seal Valve Maintenance

M1.2 Emergency Diesel Generator Issues

a. Inspection Scope (61726)

During the inspection, the inspector observed performance of portions of operations surveillance test procedures, (OST)-410 and OST-411, Emergency Diesel Generator and A and B (Twenty-Four Load Test), as well as Operating Procedure, OP-604, Diesel Generators A and B.

b. Observations and Findings

OST-411 was performed on the B EDG without any problems. However, during the performance of OST 410, the A EDG experienced a trip approximately 18 hours into the 24 hour test run. An event review team (ERT) was formed to review the event. The ERT determined that the EDG trip was caused by multiple DC grounds that were created in the engine control circuitry due to a leaking pressure switch(PS-4517A) mounted within the engine control panel. PS-4517A monitors raw cooling water (service water) to the EDG. These grounds caused the engine stopping relay to actuate, causing the engine to trip. At the time the engine tripped, the EDG was paralled to the grid. This caused the generator to motorize momentarily and cause the EDG output breaker (52/12B) to trip on reverse power.

As corrective action, PS-4517A was replaced and the EDG control circuitry was tested. Following satisfactory testing, on March 25, 1988 during the performance of post maintenance testing (PMT) using operating procedure, OP-604, the EDG was started. After running the EDG for approximately one hour, the operator attempted to shutdown the A EDG after opening the EDG to E1 bus tie breaker 52/17B, from the Generator Control Panel. The operator noted that the engine would not shutdown. Then the operator attempted to shutdown the EDG from the Engine Control Panel. Again, the operator was not able to shutdown the engine. Consequently, the operator contacted the control room and an attempt was made from the control room to shutdown the engine, and shutdown attempt from the control room was also unsuccessful. The engine was finally

secured by manually tripping the fuel racks, i.e., emergency shutdown. Troubleshooting identified that two contacts associated with the engine stopping relay (5/DG-A) had fused shut causing the stopping relay to fail. These contacts had fused most likely due to a short that was created following the leak in PS-4517A. However, this condition had not been identified during the initial troubleshooting that was conducted. The licensee replaced the stopping relay and preformed additional more indepth testing. Subsequently, OST 410 was rerun successfully in its entirety.

As a longer term corrective action, the licensee is considering relocating the raw water switch from the engine control panel to an external location, thus eliminating the possibility of a water leak and its impact on electrical circuitry.

c. Conclusion

The conduct of operations during testing was professional and well coordinated. Licensee response, including formation of an ERT to investigate the cause of failures was appropriate. However, a more detailed and thorough investigation after the first problem (engine trip) could have potentially identified the failure of the stopping relay.

M1.3 Inservice Inspection of Safety-Related Welds and Components

a. Inspection Scope (73753/73755)

The inspector verified by work observation and document review that nondestructive examinations of safety-related welds were being performed in accordance with the licensee's implementing procedures and applicable code requirements. The controlling code for H. B. Robinson ISI activities was ASME Code, Section XI, 1986 Edition with no Addenda, (Code).

Procedures used for the examination of ISI welds selected by the inspector for review were as follows:

NDE-101, Rev. 15	Radiographic Examination
SP-1217, Rev. 2	Liquid Penetrant Examination Procedure
SP-1218, Rev. 1	Magnetic Particle Procedure
SP-1230, Rev. 1	Visual Examination (VT-3) of Nuclear Power Plant Components
PDI-UT-1, Rev. 0	Ultrasonic Examination (UT) Ferretic Piping Welds

Request for Relief No. 12, October 19, 1992	Examination Category B-D, Item B3.140, SG Nozzle Inner Radius Section

b. Observation and Findings

Background

At the time of this inspection, ISI examinations on safety-related welds was nearing completion. As such, the inspector observed the examination of certain welds Class II to verify compliance with the applicable code and procedural requirements. These welds were as follows:

Item	System	NDE Method	Comments
Integral Attachment Welds; A-WS, 97	Loop 3, Feedwater Elbow Outside Radius	Liquid Penetrant (PT)	No recordable Indications (NRI) Thickness verified With UT
K-WS, 94	Loop 3, Main Steam, Box Restraint	Magnetic Particle (MT)	NRI, limited access, less than Code minimum
Circumferential Weld 11	Loop B, 16 Feedwater	MT	NRI Backing ring Verified with UT
Integral Attachment Support WS-2	RHR Heat Exchanger "A"	PT	Ground and Weld repaired or reduce fabrication indications to facilitate inspectability
"B" Cold Liquid Hot Leg, Items 7 and 8	SG "B" Noz. Inner Radius	VT-3/ Remote	NRI
Seismic Support CH-4/26 Item 113.3	3-CH-15A Dwg. AB-CAR-CH-4-26, Rev. 1	VT-3	*Rejected Support not Functional. ISI Report: 1230-98-014, 03/13/98

Field Inspections

By observation the inspector verified that examinations were performed in a conservative manner. Examiners were well qualified to carry out their assigned tasks and performed them in accordance with requirements of the code and applicable procedures. However, during this observation, the inspector noted that welds were being examined in the as-welded condition even though certain fabrication indications made it difficult for the NDE examiner to perform the examination and thereby determine the acceptability of the weld surfaces examined. The conditions observed included coarse ripples, surface slag and undercut. Following the examination, the inspector held discussions with technical personnel and the Corporate Level III Examiner's on this matter. The inspector stated that as-fabricated weld surfaces should be inspected

first by welding inspectors to assure that weld surface conditions were conducive to PT or MT examinations, before the NDE examination took place. This would relieve the NDE examiner from having to determine the acceptability of the weld surface, and instead concentrate on the examination he/she was assigned to perform. In response to this discussion, the licensee issued CR 98-00726, dated March 26, 1998. This CR will document and track the development of a programmatic instruction to address fabrication related indications identified during ISI examinations. The licensee's lack of a working program to identify and disposition weld fabrication surface indications prior to ISI related, NDE inspections was regarded as a weakness.

Review and Evaluation of NDE Records

Degraded Charging Line Support CH-4/26

*A review of licensee Inspection Report 1230-98-014, March 13, 1998, disclosed that support CH-4/26 was found in a degraded condition. Problems identified included a ½ inch gap between the nut on the top end and the base plate. Also the licensee found that the base plate of the support had pulled away from the wall a distance of approximately 1/4 inch. The support was analyzed by engineering who determined that it was not functional in its present state. Condition Report 98-00694 was issued to document the problem. The recommended corrective action was to de-activate the support and to redesign associated support AC-8-35/24 to carry the additional load of the failed support. As required by IWF-2430 of the code, the licensee increased the scope of the inspection to include an additional five supports for inspection. Results of this inspection revealed that one of the five supports selected, Item 113.8, No. P, on ISO CPL-123A was found to be non-functional due to several loose washers and a gap between the wall and the base plate from the top down to the middle of the base plate. Through discussions with the licensee's ISI site supervisor, the inspector ascertained that following this failure, the number of supports to be examined was expanded to include all supports of the same classification in the system. The sum total of additional supports earmarked for inspection came to approximately 26 supports.

RHR Exchanger "A", Support WS-2, Surface Indications Exceed Code Allowable

By review of liquid penetrant reports on integral attachment welds of the RHR Heat Exchanger "A", Support WS-2, the inspector determined that the licensee had properly examined and dispositioned all identified indications in a conservative manner and in accordance with code requirements.

c. Conclusion

ISI examinations observed were performed following approved procedures and applicable code requirements. Technicians were well trained and qualified to perform their assigned ISI examinations. The licensee's

evaluations of unacceptable field conditions were evaluated and dispositioned with conservatism. Guidelines were planned to be established to inspect and determine the surface condition of welds prior to performing ISI related surface examinations.

M1.4 Eddy Current Examination of Steam Generator Tubes

a. Inspection Scope (50002)

The inspector reviewed and evaluated the adequacy of the EC examination on SG tubing; the analysis and resolution of potentially rejectable indications and the qualifications of personnel performing analysis and resolution as applicable. The examination was conducted in accordance with requirements of ASME Code Section XI, 1986 Edition with no Addenda; Regulatory Guide 1.83, Revision 1 and Code Case N-401-1, Digital Equipment and Robinson's applicable Technical Specifications.

b. Observations and Findings

At the time of this inspection, the acquisition phase of EC examination of SG tubing was finished and analysts were in the process of concluding evaluation of the data. ASEA Brown Boveri (ABB), performed data acquisitions and the primary analysis. Framatome was responsible for the secondary independent analysis. Discrepancies between the primary and secondary evaluation results were reviewed by an independent Lead Level III analyst. In addition, the licensee contracted an independent Level III, Qualified Data Analyst (QDA) to observe the acquisition and analysis aspects of the inspection. Applicable procedures for acquisition and data analysis were as follows:

HBR-100-011, Rev.0	Multifrequency Eddy Current Examination of Nonferromagnetic SG Tubing using MIZ-30 Equipment
HBR-100-012, Rev. 0	Eddy Current Data Analysis Procedure for Evaluation of Westinghouse Steam Generator Tubing

The eddy current examination plan during the current outage (RFO-18), provided for bobbin probe examinations on 63 percent of the tubes in SG "B" and 50 percent of the tubes in SG "C". Examinations utilizing the rotating pancake/plus point (RPC) coil were as follows:

Rotating Pancake (Plus Point):	50% Top of Tubsheet, SGs "B" and "C" 50% Row 1 and 2 U-Bend, SGS "B" and "C" 50% Hot Leg manufacturing furnishing marks (MBMs) 50% Previous dents at supports, hot leg side, SGs "B" and "C"

Preliminary results of this examination indicated that no tubes were identified with indications that exceeded acceptance standards. Possible loose part indications were identified in two tubes of SG "B". When several attempts to retrieve the short wire-like objects were unsuccessful, the licensee evaluated the problem and determined that it was acceptable to leave the foreign material in place and to continue to monitor the neighboring tubes for wear during future fueling outages. Anti-vibration bar indications were identified in three tubes of SG "G". These were R35C61, R37C45 and R38C62. These indications were observed and recorded on previous outages and showed little to no change during this outage. A list of the subject tubes and corresponding through-wall depth measurements were as follows:

Row/Column	Location	Previous Outage %	1998 %
R35C61	2A	4	5
R37C45	3A	3	3
R38C62	3A	5	9

In addition, the eddy current examination identified dent type indications at support plate locations and in the free span regions. There were 167 of these indications in SG "B" and 25 in SG "C". These have been documented into the database for tracking during subsequent outages. Personnel qualifications and calibration records were reviewed and found to be in order. Applicable procedures were consistent with applicable code requirements and industry standards.

c. Conclusion

The eddy current examination of SG tubes during this outage was well managed and executed following approved plans, procedures and industry guidelines. Technical personnel were well qualified to perform their assigned tasks. The eddy current inspection plan and the results attained met code and industry standards.

M1.5 Modification of the SI System to Establish Additional NPSH Margin for SI Pumps "B" and "C"

a. Inspection Scope (62700/55050)

The inspector determined by work observation and document review the adequacy of work activities in regards to replacement and rerouting of piping to improve NPSH margin on SI pumps "B" and "C".

b. Observations and Findings

Background: During an NRC/AE onsite inspection conducted in April and May 1997, the adequacy of available NPSH margin to SI pumps "B" and "C" came into question. After further review and evaluation, the licensee determined that the present piping configuration did not provide the required NPSH margin during transfer to the recirculation mode. This problem and the planned corrective actions to improve this condition

were documented CR 97-01217. Corrective actions included reconfiguration of the suction piping for SI pumps "B" and "C"; reconfiguration of the alternate suction piping from the RHR system and straightened the normal suction piping to "C" pump to help achieve the above mentioned objective which was to provide at least two (2) feet of design margin between the available and required NPSH through the normal suction from the refueling water storage tank. The controlling document of this modification was ESR 9700366, Rev. 5, SI Pump NPSH Improvement.

Work Observation: By review of this document the inspector ascertained that the quality class of the modification was Category A, safety-related. Design requirements were governed by the code of record, B31.1-67. All welding and inspections were required to be performed in accordance with CP&L Corporate Welding Manuals, NGG-PM-0003 and CPL-HBR2-M-048 which were written utilizing ANSI/ASME B31.1-86. Welds were fabricated with the Tungsten Inert Gas (TIG) process by the Beacon Company who had been contracted to provide welding services. At the time of this inspection, fabrication and inspection of six-inch welds had been completed. The inspector inspected the completed welds and noted that their surface quality/condition was consistent with subject code requirements. Weld fabrication control records showed that required QC inspections had been performed; piping and filler metal used had been documented. The names of welders and QC inspectors who participated in the fabrication and testing of the new welds appeared at the appropriate spaces. A review of welder and QC inspector qualification records showed they had been qualified to perform their assigned tasks. During this review, the inspector noted that out of the 10 weld records reviewed, seven required radiography and the remaining three received liquid penetrant examination. Four of the seven welds that were radiographed for acceptance required multiple weld repairs to remove code rejectable indications associated with weld fabrication. These indications included lack of fusion, cracking, porosity and lack of penetration. The inspector expressed concern over the number of repairs required per weld. In discussing this matter with the licensee's welding representative, the inspector noted that although welding light-wall stainless steel pipe could present some problems, these could have been overcome by onsite training. The licensee stated that the welders had been qualified to use the TIG process but had not received additional onsite training to verify their proficiency for welding light-wall stainless steel material. The inspector considered this matter as a weakness in the licensee's welding program.

Radiography of Completed Welds

Welds requiring radiography were examined using procedure, NDE-P107, Rev. 2, Attachment "A". The inspector reviewed the films on the seven welds that required radiography for acceptance and noted the following. Film artifacts were evident on the films reviewed. This condition is usually associated with poor conditions in film developing materials and equipment in the dark room. Also the inspector noted that the sensitivity on certain radiographs showed a need for improvement in that the outline of the essential hole on the penetrameter was extremely

difficult to detect without prior knowledge of its location. This problem is usually associated with inattention to details with respect to the technique used to shoot the radiographs. The inspector discussed this matter with the licensee's Level III Radiography Examiners. The inspector considered this matter as a weakness in the area of radiography.

c. Conclusion

The pipe modification to increase the NPSH margin on SI pumps "B" and "C" was implemented following approved design specifications and applicable code requirements. Fabrication of the new welds were accomplished with some difficulty as evidenced by numerous weld rejections. Also, radiographs of the new welds exhibited film artifacts and the sensitivity on a number of films showed a need for improving the shooting technique and film developing practices. These observations demonstrated a need for raising quality standards and licensee expectations in the subject areas. The inspector considered these observations as weaknesses in the areas of welding and radiography.

M1.6 Flow Accelerated Corrosion (FAC) on SG "A" Nozzle to Pipe Reducer

a. Inspection Scope (62700)

The inspector determined that results of the FAC inspection on component FW09-55, feedwater nozzle to pipe reducer of SG "A" revealed that certain areas on the reducer to nozzle interface exhibited thinning. Engineering analysis results showed that the component was acceptable for one fuel cycle. The licensee will replace the reducer and evaluate those in SGs "B" and "C" during the next refueling outage, No. RO-19.

b. Observations and Findings

Through discussions with the licensee's cognizant engineer, the inspector ascertained that a planned inspection of the feedwater nozzle to pipe reducer (18" to 16") revealed that certain locations exhibited wall thinning to just below code minimum on the large end (nozzle side) of the reducer. The thickness of the area in question was 0.628 inches versus 0.633 inches code hoop stress minimum. The licensee's engineering analysis determined that the reducer was acceptable for one additional cycle and issued Condition Report 98-00731 to document the problem and to lay the groundwork for its replacement during the next scheduled refueling outage RFO-19. The reducers in "B" and "C" SGs were planned to be inspected and evaluated at that time.

c. Conclusion

Records of Flow Accelerated Corrosion Testing that were reviewed were found to be complete and accurate. Isolated areas with below code minimum thickness were identified in SG "A" Feedwater nozzle reducer. These areas were properly evaluated by engineering and accepted for an additional fuel cycle.

III. Engineering

E1 Conduct of Engineering

E1.1 Reactor Coolant Pump (RCP) Diffuser Cap Screws

a. Inspection Scope (37551)

The inspector reviewed licensee activities related to three Reactor Coolant Pump (RCP) diffuser cap screws that were found missing and an additional one that was found to be broken.

b. Observations and Findings

During performance of procedure CM-001, RCP Seal Assembly Maintenance during RFO-18, the "C" RCP would not lower itself on the backseat as expected following un-coupling of the motor from the pump. Backseating is the lowering of the pump shaft to seat between the lower end of the pump shaft journal bearing and the mating surface at the thermal barrier heat exchanger. Typical downward travel associated with the backseat is approximately 1.0 inch. The "C" pump was noted to lower approximately 0.25 inches. A WR/JO was initiated to investigate the cause.

The investigation required disassembly and removal of the pump impeller. During the investigation, the licensee noted that three of the sixteen RCP diffuser adapter cap screws were missing and an additional one was broken. The RCP not backseating, was attributed to the broken cap screw that had backed off slightly.

The diffuser cap screws are 5/8 inch in diameter, 4 inches in length, and are made from 302, 304, 305, or 316 stainless steel. The diffuser cap screws secure the diffuser to the diffuser adapter joint, and do not provide an RCS pressure boundary seal. Upon identification of the condition, the licensee reviewed industry data as well as contacted the vendor (Westinghouse) and confirmed similar occurrences at other sites. Further, the licensee reviewed past history at Robinson for similar occurrences. ESR 98-00160 documented licensee evaluations related to this matter.

The ESR, which included input from Westinghouse, had the following conclusions:

- Up to 8 Diffuser adapter cap screws may be missing in the diffuser/adapter joint if a symmetrical bolt pattern exists without any adverse impact on RCP operation. This evaluation enveloped the condition identified on the "C" RCP.

- The presence of loose parts originating from the missing screws in the RCS flow stream would not decrease the performance of equipment to safety. This was substantiated by no changes in documented RCP trend over the last two fuel cycles as well as no negative data related to fuel failures or other impacts.

- An unreviewed safety question did not exist for the as found condition involving less than 16 diffuser cap screws.

Based on the this, the licensee chose to leave the "C" RCP in the as found condition.

c. Conclusions

The inspector concluded that the licensee appropriately evaluated the condition involving missing RCP diffuser cap screws that was identified during RFO 18. This evaluation included the potential impact reviewed impact of the loose parts that were created by virtue of the missing cap screws on the RCS system pressure boundary as well as fuel integrity as well as operating the RCP with less than 16 diffuser cap screws.

E1.2 Motor Operated Valve Changes and Modifications

a. Inspection Scope (37551)

The inspector reviewed the licensee's plans and activities related to ESR 9700250, and ESR-9700538. These two ESRs involved motor operated valve (MOV) modifications and testing. Selected portions of these activities were witnessed and the procedures were reviewed.

b. Observations and Findings

The GL 89-10 program at Robinson was inspected and closed in NRC Inspection Report 50-261/98-01. The closure was based partially on a letter, dated February 20, 1998, from CP&L which contained several commitments. CP&L management stated that it would be taking actions to enhance performance of a number of MOVs. These activities were performed during RFO-18. The objective of these ESRs was to perform configuration changes to either obtain more actuator capability to perform design basis functions or obtain a better configuration with respect to actuator setup, or both.

Generally the modification involved changing the motor pinion gear and worm shaft gear or changing the spring pack. Installation of new gears affects the stroke time of the valve. For example, auxiliary feedwaer (AFW)-V2-16A is a Limitorque SMB-00 and the existing motor pinion has 32 teeth and the worm gear shaft has 33 teeth. With these gears, the stroke time is 30 seconds maximum. The replacement motor pinion gear has 25 teeth and the worm shaft gear has 40 teeth. These gears will cause the stroke time to increase to 50 seconds.

The licensee completed the necessary reviews and calculations to support the changes. The inspector verified selected calculations, reviewed changes to operating procedures, and revisions to the UFSAR, where necessary.

The inspector verified that all of the planned enhancements have been scheduled during RFO-18. At this time 16 valves have been scheduled, and during the first week of the outage 4 of the valves have been modified. A static post test has been successfully completed for these valves. A dynamic test has been scheduled, later in the outage, for these valves to complete the post modification test requirement. The diagnostic traces for the static tests have been reviewed by the inspector and the licensee. The results looked favorable.

The licensee utilized procedure, TMM-035, Analysis and Trending of MOV Performance. This procedure defines the methods of analysis and trending which is used for MOV diagnostic, preventive, and corrective maintenance to comply with GL 89-10. A special Attachment was prepared for the static testing scheduled to be performed during RFO-18. This Attachment was reviewed by the inspector. It is a good way to standardize the analysis of the MOV modification work performed during this outage.

c. Conclusions

The inspector concluded that the modifications scheduled to enhance MOV performance to be a well-planned program. The administrative and technical components of the effort have been well planned and should result in achieving the objective. Responsibilities have assigned to successfully complete the program. An MOV engineer has been assigned to oversee the activities to completion. The ESRs prepared for these activities contain design information to describe the issue, installation instructions for the craftsmen, and acceptance criteria for analysis.

E1.3 Instrument Sensing Line Slope (62707, 62704)

a. Scope

The NRC Architectural Inspection (A/E) (Inspection Report, 50-261/97-201) identified that the sensing lines for a number of flow measurement instruments did not have proper slope. The licensee prepared Engineering Service Request, ESR 97-00257 to correct the slope for the following flow measurement devices:

- AFW System: FI-1424, FT-1425A, FT-1425C
- SI System: FT-940, FI-941 and FT-943.

The design changes were implemented during RFO-18. The inspector reviewed the procedures related to this activity and witnessed some of the field work.

b. Observations and Findings

The inspector verified that these activities were scheduled to be performed during RFO-18. Each of these activities was assigned a Responsible Engineer (RE). The RE is the single point of contact and is

accountable for all activities through closeout. The inspector reviewed the modification package which determined that this work was considered a configuration change. The ESR restores the sensing lines to their design condition. The original design drawings stated that all sensing lines shall have a slope of 1/4 inch to the foot from the flow element to the flow metering device. The inconsistencies between the actual installation is probably due to improper installation during original work.

The inspector witnessed portions of the work on the SI system and reviewed the installation procedure. This activity required new supports to obtain the proper slope. These were installed as prescribed by the procedure and the sensing lines now have the proper slope. Post modification test consisted of leak check of the connections and re-calibration of the instrument. QC was responsible to visually inspect the sensing lines at their normal operating pressure, and was also responsible to validate the post modification testing of the tubing installation.

c. Conclusions

The inspector concluded that the activities for correcting the instrumentation sensing line slope were performed in a technically and administratively acceptable manner. All aspects of the work have been well thought out, and the work that was witnessed by the inspector was properly performed. The work packages had been reviewed and approved by all necessary disciplines. A brief discussion was held with some of the craftsman while they were performing work. The inspector concluded that they were knowledgeable and were doing satisfactory work.

E1.4 Generator Retainer Ring Modification and Related Heavy Load Lift

a. Inspection Scope

The inspector reviewed ESR 97-00067 related to main generator retainer ring replacement as well as observed portions heavy load lift activities associated with the modification.

b. Observations and Findings

The purpose of ESR 97-00067 was to replace the existing 18Mn-05Cr material generator retaining rings with 18Mn-18Cr material rings to eliminate the possibility of retainer ring tooth top cracking that has occurred on some Westinghouse designed rotors. In addition, the modification involved machining of stress relief grooves on both the generator blower hubs and rotor forging. To implement this modification, the licensee built a temporary structure in the Turbine Building lay down area, within the protected area. The generator rotor was lifted and housed in this temporary building to accomplish activities related to the modification. The temporary building was constructed prior to the unit shutdown.

The inspector reviewed the ESR, and determined that the implications of lifting the rotor from its installed location to the temporary structure were appropriately analyzed. The rotor, which constituted a heavy load, was lifted outside the plant designated "safe load path". The licensee appropriately analyzed the new load path and also had adequate contingency plans in the event of a problem. The inspector monitored the rotor lifts and determined that the activity was accomplished without any adverse or unsafe incidents.

c. Conclusions

The inspector also concluded that the generator rotor retaining ring modification, including the lifting of the rotor, which constituted a "heavy load" was appropriately accomplished.

E2 Engineering Support of Facilities and Equipment

E.2.1 Inspection and Repair of the Containment Vessel Liner

a. Inspection Scope

The inspector reviewed the licensee's program for inspection of the containment vessel liner and examined the containment vessel liner in areas where the insulation had been removed for performance of the inspections.

b. Observations and Findings

The pressure retaining containment vessel for Robinson is a reinforced concrete structure with vertical post-tensioning bars. A containment liner which consists of steel plate with a nominal thickness of 3/8 or ½ inches serves as a leak proof membrane. The liner plate is covered with insulation panels which are in turn covered with sheet metal panels (sheathing panels) to protect the liner and insulation from moisture. The sheathing panels are held in place by metal studs which were welded to the liner plate. The original construction requirements specified that joints in the sheathing panels were to be caulked. The steel liner plate was coated with protective coatings (paint) to prevent corrosion. The coatings were classified as Service level II (nonsafety-related) since the coatings were covered by the insulation and sheathing panels.

During an inspection documented in NRC Inspection Report No. 50-261/96-14, the inspector identified areas where the caulking was deteriorated or was missing, and areas where the sheathing panels were loose or displaced. The liner plate appeared to be corroded in these areas. Further review of this concern disclosed that the liner had been inspected in selected area during previous refueling outages. Some corrosion had been identified which was documented as a nonconforming condition on an adverse CR. The corrosion was repaired by cleaning of the liner plate and recoating with paint. Additional liner inspections

were planned based on the results of the initial inspections and to close the CR. Work requests were issued to schedule performance of the inspections and close out the CR. However, the work requests were canceled prior to performing the planned inspections. A violation was identified for failure to perform the additional liner inspections and for improperly canceling the work requests. In response to the violation, the licensee committed to perform additional inspections of the liner during the current refueling outage (RFO 18).

The inspector reviewed CP&L Special Procedure SP-1419, Inspection and Repair of CV Liner and Insulation, Revision 0, dated March 6, 1998. This procedure established the liner inspection requirements, instructions for cleaning the liner plate prior to performing the inspections, inspection methodology, acceptance criteria, and requirements for engineering assessments and repairs, if necessary. The acceptance criteria were based on calculation number RNP-C/STRU-1128, Minimum Allowable Containment Liner Thickness, Revision 0, dated March 4, 1998. The inspector also reviewed procedure MMM-039, Control of Protective Coating, Revision 8. This procedure specifies the requirements for coating materials, surface preparation, and application of new coatings after the inspections were completed. The acceptance of the coating materials were evaluated in ESR 9800092, Containment Vessel Liner Plate Coatings.

Revision 0, dated March 13, 1998. The new coating are classified as Service Level II since they will also be covered by the sheathing panels and insulation.

The inspector also examined the licensee's scope of work which identified the locations of the sheathing panels to be removed for inspection. The locations were selected to resolve the violation, to inspect areas where the caulking was missing, to replace studs that were missing, to replace degraded sheathing, and other areas selected by the project manager.

The inspector walked down the reactor building and examined the liner in the areas where the panels had been removed for liner inspection. The majority of the panels were located at elevation 228 where the concrete floor intersects with the liner plate. This was the area most suspectable to corrosion. The licensee had completed cleaning and inspection of the liner plate, and had performed ultrasonic testing to measure the liner thickness. The UT measurements showed that the liner plate exceeded the minimum thickness (0.9 of the nominal) required by the acceptance criteria. In most cases, the liner thickness exceeded the nominal thickness shown on design drawings. The depth of corrosion was determined by performance of physical measurements. The inspector performed some independent measurements of the corroded areas in randomly selected areas and compared the measured corrosion with that measured by the licensee. No discrepancies were identified.

During the walkdown inspection, the inspector noted that sheathing panel number 228C was loose and pulled away from the liner plate. The inspector requested that the licensee remove this sheathing panel and one additional panel to inspect the liner in these areas. The additional panel was selected based on the quantity and depth of corrosion in an adjacent panel which appeared to extend under the panel. After the panels were removed, the inspector examined the liner and noted that the corrosion was within acceptable limits for the additional panel. However inspection of the liner under panel 228C disclosed some areas of heavy pitting and corrosion. The corrosion was being evaluated in accordance with procedure SP-1419.

The inspector reviewed the licensee's records which document the inspection results. These records included visual inspection (VT-3) performed on 228C, 228N, 228Y, 228Z, 228AA through 228FF, 228KK, 228PP through 228UU, and 232KK. The results of the licensee's inspection of the liner disclosed areas where the depth and area of corrosion exceeded the acceptance criteria. Engineering evaluations were performed using the methods specified in procedure SP-1419 to determine if additional evaluations or repairs were required. The inspector reviewed the results of the evaluations performed for locations 228 SS, 228UU, and 228YY. The evaluations showed that the liner plate was acceptable and weld repairs were not required.

c. Conclusions

The licensee's program for inspection of the containment vessel liner was effectively implemented. The results of the inspections showed that the liner met design requirements.

E7 Quality Assurance in Engineering Activities

E7.1 Special UFSAR Review (37551)

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 inspections discussed in this report, the inspector reviewed the applicable portions of the UFSAR 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 (92903) (37551)

E8.1 Partial Control Rod Drive Mechanism (CRDM) Motor Tube Cracking

a. Infection Scope

The inspector reviewed licensee evaluation (ESR 9800100) that was conducted to access the susceptibility to cracking of the motor tube components of the partial length CRDM .

b. Inspection Findings

Partial length CRDM cracking was experienced at another plant (Prarie Island Unit 2) in February, 1998, resulting in a RCS leakage that required a unit shutdown. The leak was the result of a through wall crack located in a weld buttering pass of a dissimilar metal transition between type 403 and 404 stainless steel. Corrective actions at Prarie Island included cutting and capping of the affected four partial length CRDMs.

Robinson reviewed the details associated with the problem at Prarie Island for applicability. Robinson originally had eight partial length CRDMs of the same design, vintage, and fabricator as Prarie Island. One of the original eight had ben capped due to a different reason (canopy seal leak). The remaining seven partial length CRDMs are not in service and are withdrawn to the refueling configuration. However, the associate motor tubes, which are a RCS pressure boundary remain, creating the possibility of a similar failure.

The evaluation determined the issue was not of immediate concern to Robinson and would be evaluated through the coordination efforts led by Westinghouse Owners Group (WOG) Regulatory Response Group (RRG). Notwithstanding, individuals involved in performing the RFO-18 reactor vessel head inspection were asked to pay closer attention to evidence of boric acid leak. No obvious leakage was identified.

c. Conclusion

The inspector concluded that the licensee is appropriately tracking and addressing the partial length CRDM cracking issue. Long term licensee actions will be developed through the efforts led by Westinghouse Owners Group.

IV. Plant Support

R1 Radiological Protection and Chemistry Controls

R1.1 General Comments (71750)

The inspector 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 conditions. The inspector concluded that radiation control practices were proper.

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.

P8 Miscellaneous Security and Safeguards Issues (92904, 92702)

P8.1 (CLOSED) Violation B, 50-261/97-07-06, Unauthorized structure in protected area and inadequate microwave configuration.

The corrective actions presented in the licensee's response, dated August 5, 1997, were reviewed and verified by the inspector. New microwave intrusion detection equipment was installed and the stairwell wall has been modified. The NRC accepted the licensee's response on August 18, 1997. 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 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
J. Keenan, Vice President, Robinson Nuclear Plant
R. Duncan, 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
M. Shymlock, Branch Chief, Region II

INSPECTION PROCEDURES USED

IP 37551:	Onsite Engineering
IP 40500:	Effectiveness of Licensee Controls in Identifying, Resolving, and Preventing Problems
IP 61726:	Surveillance Observations
IP 62707:	Maintenance Observation
IP 71707:	Plant Operations
IP 71750:	Plant Support Activities
IP 92904:	Followup - Plant Support
IP 73753:	Inservice Inspection
IP 73755:	Inservice Inspection Data Review and Evaluation
IP 55050:	Nuclear Welding
IP 50002:	Steam Generators
IP 62700:	Maintenance Observation
IP 62704:	Instrument Maintenance
IP 92903:	Followup - Engineering

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

Type	Item Number	Status	Description and Reference
NCV	50-261/98-02-01	Open	Failure to Have Two RHR Trains Operable in Mode 6 (Section O1.3).

Closed

Type	Item Number	Status	Description and Reference
NCV	50-261/98-02-01	Closed	Failure to Have Two RHR Trains Operable in Mode 6.(Section O1.3).
VIO	50-261/97-07-06	Closed	Unauthorized structure in protected area and inadequate microwave configuration (Section P8.1).