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

Report No.: 50-261/92-16

Licensee: Carolina Power and Light Company
P. O. Box 1551
Raleigh, NC 27602

Docket No.: 50-261 License No.: DPR-23

Facility Name: H. B. Robinson Unit 2

Inspection Conducted: May 9 - June 27, 1992

Inspector: R E Carroll for 7/22/92
L. W. Garner, Sr. Resident Inspector Date Signed

Other Inspectors: C. R. Ogle, Resident Inspector
G. R. Wiseman, Project Inspector

Approved by: H. O. Christensen 7/23/92
H. O. Christensen, Section Chief Date Signed
Division of Reactor Projects

SUMMARY

Scope:

This routine, announced inspection was conducted in the areas of operational safety verification, surveillance observation, maintenance observation, engineered safety feature system walkdown, plant review committee, modifications, and followup.

Results:

A violation with two examples was issued for failure to adequately establish maintenance procedures. CM-303 was inadequate in that proper instructions were not provided to ensure environmentally qualified taped splices were installed in tested configurations. CM-508 was inadequate in that the steps to reassemble the emergency diesel generator fuel oil filter were out of sequence (paragraph 5).

A violation was identified for failure to provide an adequate modification procedure in that the approved modification created the potential for an unmonitored release path (paragraph 8).

A non-cited violation (NCV) was identified for failure to comply with 10 CFR 50 Appendix B Criterion III in that safety related

piping and valves were not adequately supported to meet seismic criteria (paragraph 3).

A NCV was identified for failure to have the A emergency diesel generator shutdown sequence defeat relay installed as required by plant drawings (paragraph 4).

A NCV was identified for failure to include a safety-related snubber in the Technical Specification required surveillance test program (paragraph 10).

Direct communications between the control room and the refuel floor was poor in that when an unexpected neutron count rate decrease was detected this information was not immediately transmitted to the refuel floor (paragraph 3).

A weakness was identified in Operations' corrective action program implementation in that a personnel error was not captured in a corrective action program (paragraph 3).

The licensee's approach to resolving support issues for small bore piping and valves was conservative (paragraph 3).

Corrective actions to preclude repetition of work control and scheduling problems experienced during the previous refueling outage were successful (paragraph 3).

Engineering support for scheduled and emergent outage activities were very good (paragraph 3).

Command and control during plant restart from the refueling outage was good (paragraph 3).

Technical Support Improvement Program implementation was slow; however, personnel performance had continued to improve (paragraph 3).

The outage goal for person-rem exposure was met; however, the goals for the number of contaminations and volume of radwaste generated were exceeded (paragraph 3).

Material condition of some components were found to be poor (paragraph 5).

Though deficiencies continued to occur in modification design bases, the quality of modification installation instructions have improved (paragraph 3).

Early Nuclear Assessment Department audits were oriented toward problem identification. Later audits effectively focused in on root cause determinations (paragraph 9).

REPORT DETAILS

1. Persons Contacted

*R. Barnett, Manager, Outages and Modifications
J. Benjamin, Shift Outage Manager, Outages and Modifications
*R. Beverage, Manager, Quality Control
W. Biggs, Manager, Nuclear Engineering Department Site Unit
D. Blakeney, Shift Outage Manager, Outages and Modifications
*R. Chambers, Plant General Manager, Robinson Nuclear Project
T. Cleary, Manager - Balance of Plant Systems and Reactor Engineering, Technical Support
*D. Crook, Senior Specialist, Regulatory Compliance
C. Dietz, Vice President, Robinson Nuclear Project
*J. Dobbs, Manager, Nuclear Assessment Department Site Unit
*J. Eaddy, Manager, Environmental and Radiation Support
S. Farmer, Manager - Engineering Programs, Technical Support
R. Femal, Shift Supervisor, Operations
*W. Flanagan, Manager, Operations
*W. Gainey, Manager, Plant Support
B. Harward, Manager - Mechanical Systems, Technical Support
D. Knight, Shift Supervisor, Operations
*A. McCauley, Manager - Electrical Systems, Technical Support
R. Moore, Shift Supervisor, Operations
D. Nelson, Shift Outage Manager, Outages and Modifications
A. Padgett, Manager, Environmental and Radiation Control
*M. Page, Manager, Technical Support
D. Seagle, Shift Supervisor, Operations
M. Scott, Manager - Support Systems, Technical Support
B. Clark, Manager, Maintenance
D. Stadler, Onsite Licensing Engineer, Nuclear Licensing
W. Stover, Shift Supervisor, Operations
D. Winters, Shift Supervisor, Operations

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

*Attended exit interview on July 9, 1992.

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

2. Plant Status

RO-14 ended at 11:03 a.m., on June 24, 1992, when the turbine generator was placed in service. The RO, scheduled for 70 days, was completed in 88 days. A summary of outage activities is presented in paragraph 3. At the end of the report period, reactor power was being increased to 100 percent.

3. Operational Safety Verification (71707)

The inspectors evaluated licensee activities to confirm that the facility was being operated safely and in conformance with regulatory requirements. These activities were confirmed by direct observation, facility tours, interviews and discussions with licensee personnel and management, verification of safety system status, and review of facility records.

To verify equipment operability and compliance with TS, the inspectors reviewed shift logs, Operations' records, data sheets, instrument traces, and records of equipment malfunctions. Through work observations and discussions with Operations staff members, the inspectors verified the staff was knowledgeable of plant conditions, responded properly to alarms, adhered to procedures and applicable administrative controls, cognizant of in-progress surveillance and maintenance activities, and aware of inoperable equipment status. The inspectors performed channel verifications and reviewed component status and safety-related parameters to verify conformance with TS. Shift changes were observed, verifying that system status continuity was maintained and that proper control room staffing existed. Access to the control room was controlled and Operations personnel carried out their assigned duties in an effective manner. Control room demeanor and communications were appropriate.

Plant tours and perimeter walkdowns were conducted to verify equipment operability, assess the general condition of plant equipment, and to verify that radiological controls, fire protection controls, physical protection controls, and equipment tagging procedures were properly implemented.

Leaning Fuel Assembly

On May 20, 1992, during loading of fuel assembly N45 into the core, the assembly caught on the edge of fuel assembly W08. Fuel movement was immediately stopped; however, movement was not stopped in time to prevent both assemblies from being pushed outward from the vessel wall approximately 10 degrees. To ensure that N45 would remain attached to the refuel hoist, the refueling operator directed that the refuel hoist be raised slightly to place additional weight back on the refuel hoist. The refueling bridge was then repositioned approximately 1 inch north to increase the contact area between the two assemblies. Control room personnel were then informed of the situation. After consultation with the shift supervisor and plant management, it was decided to secure fuel assembly W08 with a tag line and write a special recovery procedure prior to any additional movement of the assemblies. The inspectors observed the successful performance of SP-1143, Removal Of Leaning Fuel Assembly. The SP provided instructions, including precautions and limitations, to remove N45 from on top of W08 and place it a specified core location. The evolution was well planned and executed. Preliminary inspection by a underwater camera revealed no damage to either assembly. This was later confirmed by additional detailed inspections. The assemblies were found to be acceptable for use and were subsequently placed into core.

The potential consequences of damage to one or both assemblies was small. At the time of the event, two fuel assemblies were loaded in the core. Assembly W08, a new fuel assembly containing once irradiated source rodlets, was loaded around source range monitor N32. Another assembly was loaded around neutron monitor N51, but was too far from W08 and N45 to be effected. N45 was the second assembly to be positioned around source range detector N32. Assembly N45 was a PLSA which had been first loaded in cycle 10, October 1984. Calculations of the quantity of radioactive material present indicated that no significant release would have occurred if both assemblies had fallen and released their radioactive material.

During subsequent inspection of this event, it was determined that direct communication between the control room and the refuel bridge was poor during the event. The person in the control room who was in direct communications with the refuel manipulator crane, as required TS 3.8.1.g, heard a decrease in the audio count rate and observed an unanticipated step decrease in neutron counts when the assembly was being loaded. However, this information was not immediately provided to the refueling floor personnel nor were any inquiries initiated to determine if any thing unusual had occurred. Apparently, turnover activities had distracted the individual from pursuing the observations. After this event, shift personnel were directed to discontinue fuel movement during shift turnover. The count rate decrease was later attributed to W08 leaning away from the detector.

Mispositioned Valve During Performance Of OST-163

On June 3, 1992, during performance of OST-163, Safety Injection Test And Emergency Diesel Generator Auto Start On Loss Of Power And Safety Injection And Emergency Diesel Trips Defeat, the SS identified that RHR-744B was in the open position; however, an initialed step in the procedure indicated that the position should have been closed. RHR-744A and B valves were the cold leg low pressure safety injection valves which also served as a return path to the RCS during shutdown cooling. After securing shutdown cooling in accordance with OST-163, a test delay necessitated returning RHR to the shutdown cooling mode. The operator reopened both the RHR-744A and B valves as was the normal configuration for shutdown cooling; however, OST-163 had required shutdown cooling to be configured with only the RHR-744A valve open. The valve position discrepancy was not captured in any corrective action program. Though this event had little safety significance, it could represent either an isolated error or a potential training issue involving the proper way to back out of a procedure. Failure to capture these types of errors in a program which includes trending for similar problems precluded the opportunity to determine if the item was more than an isolated event. Failure to include the valve misposition occurrence in a corrective action program was a weakness in implementation of Operations' corrective action program.

Small Bore Piping/Valve Supports

On June 5, 1992, engineering determined that under seismic conditions the stresses in piping adjacent to five SI system valves would exceed the pipes' yield strength. A 10 CFR 50.72 report was made this same day as required. Supports to the valves, listed below, were added or modified as necessary.

SI-850E	C Accumulator Test Valve
SI-850F	Loop #3 Cold Leg Injection Test Valve
SI-851C	C Accumulator Make-up Isolation Valve
SI-856A,B	High Head SI Test Line To The RWST Isolation Valves

Subsequent to the 50.72 report, engineering walkdowns and/or evaluations determined that the following valves were also potentially inadequately supported and were modified prior to restart:

CVC-303A,B,C	RCP (A B, C) Seal Leakoff Isolation Valve
SI-845C	Containment Spray Addition Throttling Valve
V12-14	PACV Train A CV Hydrogen Exhaust Valve
V12-24A	PACV Instrument Air To CV Supply Valve
V12-24B	PACV Station Air To CV Supply Valve

It was anticipated that subsequent calculations would reveal that some of the above listed valves may have met short-term qualifications. The modifications to the valve supports were to long-term criteria. Another nine valves were identified by analysis to be short-term qualified. These nine valves were to be upgraded to long-term criteria by no later than RO-15.

V12-14, listed above, was on 3-inch analyzed piping. During the walkdowns, this valve was observed to move freely. The engineering analysis was determined to be correct; however, the clearance between the box type support and the piping was greater than allowed. A shim was installed on the support. A similarly supported valve (V12-18) on the same system was examined and was determined to be properly supported. Thus, the problem with V12-14 was considered to be an isolated occurrence.

The review of small bore piping valve supports was initiated in RO-13 in order to address the inaccurate weights and centers of gravity for Copes Vulcan valves. This information is important in accurately determining the stresses on the associated piping, because the Copes Vulcan valve actuators are rather large (extended actuators). The inaccurate information was the subject of NRC information notices 89-28 and 90-17. During RO-13 a scoping review was performed to identify and obtain design information on Copes Vulcan valves. Of the 74 Copes Vulcan valves identified in safety-related applications, 29 were in previously analyzed piping and 45 were in unanalyzed piping. During RO-14 a detailed inspection of Copes Vulcan valves in unanalyzed piping was performed to determine if they were adequately supported. During this inspection a snubber was found between the polar crane wall and the actuator for SI-850D, Loop #2 Cold Leg Injection Test Valve. This snubber had not been included in the TS snubber surveillance test program as required. This condition was reported as required by 10 CFR 50.73 (see LER 92-09 closeout in paragraph 10). An analysis was performed to determine the inoperable snubber's affect on the piping system. This analysis demonstrated that without the snubber the piping would remain functional; however, long-term seismic margins required by the code were not met. Based upon this unexpected result the inspection criteria was redefined and expedited. On June 4, 1992, a NED principal engineer reviewed the Copes Vulcan inspection status with the inspectors. During this discussion it became apparent that the problems identified up to that time (later the subject of the June 5, 10 CFR 50.72 report) were not associated with the Copes Vulcan valve design inaccurate information issue, but were related to other aspects of the original design. Subsequent management review determined that the inspection scope should be expanded to include non-Copes Vulcan valves which have extended actuators. The inspection program was expanded to include ten safety-related non-Copes Vulcan valves on unanalyzed piping inside the CV. Of the ten valves, the eight 3/8-inch valves were determined to be acceptable as found; however, the two 2-inch valves were not well supported and were modified. Based upon this result, the inspection program was expanded outside the CV to include 2-inch safety-related non-Copes

Vulcan valves with extended actuators. This selection was reasonable in that valves less that 2-inches were required by design to have a support attached to the valve/actuator and valves greater than 2-inches were required to have been previously analyzed. Using the 2-inch criteria, seven valves were identified for the expanded inspection program outside the CV. One additional valve was identified which was potentially not supported adequately. The three non-Copes Vulcan valves are included in the above "subsequent to the 50.72 report" listing.

The inspection acceptance criteria for valve supports were based upon subjective similarity judgements. A valve's stiffness or resistance to motion when pushed was subjectively compared with that of a valve which had been successfully analyzed. In making an acceptable determination, the frequency of the induced vibration was also considered. The inspectors accompanied the NED engineers during portions of their inspection activities inside and outside the CV. Though the subjective criteria was rather imprecise, the actual field implementation required little precision (i.e., either the valves were obviously well supported or they were not). In the few instances in which it was difficult to determine if the valve was adequately supported, the valves were either modified or analyzed for acceptability. The process, a combination of field inspections and analyses, was conservative.

The failure to have safety related piping and valves adequately supported was a violation of 10 CFR 50 Appendix B Criterion III. This violation will not be subject to enforcement action because the licensee's efforts in identifying and correcting the violation met the criteria specified in Section VII.B of the Enforcement Policy. This item is identified as a NCV: Failure To Have Small Bore Safety-Related Piping/Valves Adequately Supported, 92-16-01.

Bubble Formation In CCW

On June 18, 1992, at 9:49 a.m., CCW surge tank level oscillations and cavitating type sounds from the CCW system were detected. Investigation revealed that a bubble had formed in the B RHR Hx. The B RHR loop was declared inoperable as required by TS. Bubble formation had occurred because the CCW flow through the Hx had been secured by closing CC-749B, the B RHR HX CCW outlet valve. CCW flow to the Hx was slowly returned to normal and the system was returned to normal service the same day. The system engineer preformed a walkdown of the CCW system and determined that there was no damage to any system components. CC-749B had been closed to promote heatup of the RCS. This has been a common practice; however, in the past the CC-749A and B valves had apparently leaked enough to provide sufficient flow through the Hxs to prevent heatup of the CCW in the Hxs. During this outage, the CC-749 valves were repaired. Incorporation of a caution note in the operating procedures to not secure CCW to a HX that is in service was being considered. At this time, the inspectors have no further questions concerning this event.

Reactor Trip

On June 19, 1992, at 8:49 p.m., a reactor trip occurred during performance of surveillance test procedure EST-052, Operational Alignment Of Process Temperature Instrumentation. At the time of the trip, the unit was in hot shutdown conditions (547 degrees F and 2235 psig) with the shutdown control rod banks withdrawn. Prior to the test, loop 1 OT and Tavg RPS bistables had been placed in the tripped condition because post maintenance testing had not yet been performed on the loop's RTDs. During EST-052 when Loop 3 RTD TC-432C was placed in test, loop 3 instrumentation sensed a high temperature and initiated an OT delta T RPS actuation signal. This completed the 2 out of 3 logic for a reactor trip signal. Subsequent investigation revealed that the circuit had performed as expected. At 547 degrees F the instrumentation was operating near its lower band (540 degrees F). Therefore, when the test switch removed the RTD input, the circuit could respond in a number of different ways, including tripping. It was noted that the other RTD circuits when tested at this temperature had not tripped. The inspectors have no further questions at this time.

RO-14 Summary

The RO which began on March 27, 1992, ended on June 24, 1992, after 88 days. The major contributors to the 18 day extension were: M-1087, RHR Pumps' Minimum Flow Recirculation, (6.6 days); M-1128, Removal Of SI-857B Relief Valve, (2.8 days); refueling equipment repairs (2.5 days); OST-253, RHR Pump Flow Test, (1.4 days); and polar crane repairs (1.3 days).

Corrective actions to address items which caused significant scheduling and work control problems during the previous RO successfully precluded similar problems this RO. In particular, parts procurement difficulties were essentially avoided by the earlier identification of parts required for modifications and maintenance activities. Another area of significant improvement was maintenance preplanning. A dedicated group of planners was established to plan PM and CM work activities. Prior to the RO, all the PM work requests and 90 percent of scheduled CM work requests were planned. To preclude recurrence of an unlatched control rod, a specially fabricated tool was used to measure button position. The data was subsequently evaluated by engineering for acceptability. This process has been incorporated into FHP-007, Control Rod Drive Shaft Installation And Latching.

Modifications to meet regulatory commitments, upgrade safety-related equipment and improve plant performance were installed during this RO. Examples included modifications to: add additional RHR minimum flow capacity, provide EDG KW indication on the RTGB, replace electrical penetrations, provide improved control room annunciation, upgrade refueling equipment, and install a larger HDP FCV. Though there has been continuing deficiencies associated with modification design bases (VIO 92-11-05 and 92-16-04), the quality of the installation instructions has improved as indicated by a reduced number of field revisions. Most modification packages were approved well in advance of the RO, reflecting management's focus on eliminating the need for field revisions. The increase in modification package quality was attributed to an increased involvement by engineering and plant personnel in system teams during the review process. Use of feedback from the field revision justification form reviews was anticipated to continue improvements in this area.

Engineering support, both NED and Technical Support, for scheduled and emergent RO activities was very good. NED continued to provide onsite support as necessary. Corporate NED involvement was enhanced by periodic and special conference calls to discuss RO activities and potential emergent work items. Frequent visits by the HBR engineering manager and his principal engineering supervisors substantially improved NED management involvement in outage issues. Technical Support provided continuous around the clock engineering coverage, as well as dedicated individuals for special projects such as the SIT and ILRT. In addition, substantial engineering effort, including field verifications, was expended in completing and implementing the fuse control program.

The inspectors witnessed selected startup activities such as low power testing, placement of the unit online, and power ascension activities. The inspectors verified that evolutions were accomplished in accordance with approved procedures and that TS and other regulatory requirements were being met. Command and control during major activities was good. Access to the control room was limited to personnel supporting startup activities. Continuing a practice initiated last RO, startup managers provided around the clock management support to Operations such that the SS was free to focus on startup activities. Operations management, as well as NAD personnel, observed startup evolutions.

Control Room Instrumentation/Annunciation And Equipment Out-Of-Service

At the end of the report period there were 14 control room indicators (meters, strip chart recorders and annunciators) out-of-service. There were also 12 pieces of equipment out-of-service with an additional 4 components operating in a degraded condition. Only a few of these items were carry-overs from last cycle. The number of items appeared to be large for just having completed an RO. The inspectors were unable to determine the significance of the number of items. After restart from the RO, the number of lit annunciators typically varied from 3 to 6. Prior to the outage, the number of lit annunciators were similar (i.e., 4 or 5, with rare instances of no annunciators lit).

Equipment Identification

Management emphasis on valve tagging during the RO showed improved performance in this area when compared to the previous outage. During RO-13, valve tagging was not aggressively pursued as demonstrated by having 477 missing tags identified while conducting end-of-outage valve lineups and another 232 tags which were identified before the outage, but were not placed in the field. At the end of RO-14, 394 valve tags were identified during valve lineups; however, only a dozen or so had been identified prior to the outage.

Technical Support Improvement Program

The Technical Support Improvement Program implementation has been slow. Based upon the May 1992 improvement program schedule, 3 items have been completed, 1 item has been accelerated, 13 items have been extended, 6 items were unchanged, and 1 new item was added. The completed items were repetitive failure program restructuring, and thermograph PM and post maintenance testing procedure developments. Extended items included component engineer development (6 months), managed valve maintenance program (20 months), procedure writer's guide development (12 months), and performance monitoring program (6 months). EDBS development was added to the improvement program. Though not scheduled to be completed until the end of 1995, a significant amount of resources (11 engineers) were dedicated to the EDBS effort.

At the end of June 1992, the number of outstanding Technical Support work items was 1494. The number included 973 projects with their associated 380 subtasks and 141 routine, but unscheduled, activities. The number of work items has remained fairly constant during the last year. Progress has been made in the management of the items. The projects have been grouped into three categories: active, backlog (need to work), and enhancements (wish list). Also, estimated manpower requirements for each item have been developed. With contractor assistance, there has been a small but steady decline in the number of projects during the last few months.

Progress has continued in several areas which were to improve personnel performance. For example, the number of system teams has increased in the last 15 months. For the seven months starting in September 1991, there were 15 active system teams. This was nearly double that from the preceding year. The system engineering system certification program, developed in early 1991, was adversely impacted by preparations for the RO. Only four engineers were certified on systems at the end of this report period.

DBD Status

The development and validation of DBDs was complete except for the validation of the RVLIS DBD, which was scheduled for completion in July 1992. Progress was continuing in resolution of DBD discrepancies. By the end of June 1992, 214 of the identified 240 DBD discrepancies had been addressed. Resolution of the remaining items was scheduled to be completed by early fall; however, implementation of recommended actions was anticipated to take several years.

Maintenance Improvement Plan

In October 1990, a Maintenance Improvement Plan was established to address self-identified and maintenance inspection team report items. The improvement plan was closed in January 1992, after all short-term items were deemed to have been addressed. Outstanding long-term items included the maintenance procedure upgrade project, safety related equipment PM upgrade and new maintenance facilities.

A pilot program was implemented in April 1991, to provide around the clock maintenance coverage. In December 1991, management review determined that the program's disadvantages such as lack of direct supervision (time to coach) and craft unavailability to support Technical Support projects, necessitated that a modified coverage scheme be implemented. The modified plan provided for two instrumentation and control technicians and two mechanics to work 11:00 p.m. to 7:30 a.m., Monday through Friday, excluding holidays. At the end of the IR period, management was considering an increase in maintenance staffing to allow larger crew sizes and provide for a permanent maintenance procedure writing group.

Two major performance indicators, trended by the maintenance unit, showed a continued high performance level during 1991. Overall unit capacity factor and availability were 80 percent, with a 98 percent safety system availability. The primary system leak rate also continued to be maintained at a small value (i.e., approximately 0.1 gpm). Furthermore, no plant transients were initiated by maintenance personnel; however, the December 1991 maintenance monitoring report identified concerns in the areas of work control, planning, and work practices. Maintenance supervision spent approximately 25 percent of their time on plant tours and observations of work in progress. Non-outage outstanding WRs continued to remain at about the same level experienced in the previous year (i.e., 906 and 835 at the end of 1990 and 1991, respectively).

E&RC Outage Goals

The E&RC outage goal for person-rem exposure was met, whereas those associated with the number of contaminations and radwaste volume were not met. Total dose exposure as measured by TLDs was 298.4 person-rem. The outage goal was 350 person-rem with a stretch goal of 318 person-rem. The number of skin and clothing contaminations were 53 and 68, respectively. The total number of contaminations (117) exceeded the outage goal of 90. However, this was a reduction from the 292 contamination events which occurred during RO-13. Most of the reduction can be attributed to differences in outage durations. RO-14 was completed in 88 days whereas RO-13 was completed in 183 days. However, the number of contamination events per day showed a small, but noticeable decline. The radwaste volume generated exceeded the 1650 cubic feet volume goal by 559 cubic feet.

One NCV was identified. Except as noted above, the program area was adequately implemented.

4. Monthly Surveillance Observation (61726)

The inspectors observed certain safety-related surveillance activities on systems and components to ascertain that these activities were conducted in accordance with license requirements. For the surveillance test procedures listed below, the inspectors determined that precautions and LCOs were adhered to, the required administrative approvals and tagouts were obtained prior to test initiation, testing was accomplished by qualified personnel in accordance with an approved test procedure, test instrumentation was properly calibrated, the tests were completed at the required frequency, and that the tests conformed to TS requirements. Upon test completion, the inspectors verified the recorded test data was complete, accurate, and met TS requirements; test discrepancies were properly documented and rectified; and that the systems were properly returned to service. Specifically, the inspectors witnessed/reviewed portions of the following test activities:

EST-050	Refueling Startup Procedure
EST-067	Intermediate Range Detector Setpoint Determination
EST-124	Response Time Testing Of Reactor Coolant System RTDs
OST-163	Safety Injection Test And Emergency Diesel Generator Auto Start On Loss Of Power And Safety Injection And Emergency Diesel Trips Defeat
SP-1144	Returning SDAFW Pump To Service After Governor Repair
SP-1149	Testing SI-863A And SI-863B At Hot Shutdown

OST-163

On June 3, 1992, during the performance of OST-163, the A SI logic train failed to reset properly. At approximately the same time, the A EDG also tripped on mechanical overspeed. Subsequent investigations into these occurrences determined that they were unrelated.

The SI reset was later accomplished after repeated attempts. Initial troubleshooting of the A logic train indicated that the reset switch contacts for the A train had higher resistance than that for the B train. The switch, manufactured by GEMCO, was replaced. During the subsequent performance of OST-163, the same train again failed to reset. The reset feature associated with the A SI train master relay, SIA, was determined to be malfunctioning. The relay was replaced and OST-163 was successfully completed on June 6, 1992.

By the process of elimination, it was determined that the probable cause for the A EDG mechanical overspeed trip was a failure to properly reset the trip mechanism after the previous run. Routine practice was to manually actuate the trip mechanism after surveillance testing to allow the EDG to be rolled over with air to clear the cylinders of lube oil. Apparently, Operations personnel were unaware of the importance of moving the reset lever through its full travel. Resetting was difficult for some individuals due to the reset lever's location (i.e., some individuals could barely reach the lever which was positioned over the exhaust manifold). Operations personnel were subsequently trained on the proper method to actuate the reset lever. Also, a step was provided for the shorter operators to allow them easier access to the reset lever. Additional information concerning this event is contained in IR 92-19.

During investigation into the probable cause for the A EDG mechanical overspeed trip, it was discovered that relay 4AX had its coil wire lifted. This relay had been installed in 1984 as part of M-817, Diesel Generator Stopping Sequence. The relay allows a normal or emergency start signal to override the shutdown sequence (see IEN 83-17). The wire was reterminated and the feature was satisfactorily tested. A review of plant records could not determine when the wire was lifted. The B EDG control circuit was examined and its 4AX relay was verified to be operational. Testing of the shutdown sequence override feature was not required; however, engineering recommended that test procedures be revised to periodically verify this feature. Futhermore, other EDG control circuit features not required by TS to be tested were also to be evaluated for incorporation into the test program. Failure to have the 4AX relay connected to the A EDG control circuit as required by plant drawing no. 5379-1153 was a violation of 10 CFR 50 Appendix B Criterion V. This violation will not be subject to enforcement action because the licensee's efforts in identifying and correcting the violation met the criteria specified in Section VII.B of the Enforcement Policy. This item is identified as an NCV: Failure To Have The Shutdown Sequence Defeat Relay Installed In The A EDG Control Circuit As Required By Plant Drawings, 92-16-02.

One NCV was identified. Except as noted above, the program area was adequately implemented.

5. Monthly Maintenance Observation (62703)

The inspectors observed safety-related maintenance activities on systems and components to ascertain that these activities were conducted in accordance with TS, approved procedures, and appropriate industry codes and standards. The inspectors determined that these activities did not violate LCOs and that required redundant components were operable. The inspectors verified that required administrative, material, testing, radiological, and fire prevention controls were adhered to. In particular, the inspectors observed/reviewed the following maintenance activities:

CM-303	Installation Of Environmentally Qualified Taped Splices
SP-1145	Spent Fuel Pit Upender Cable Repair
WR/JO 92-AHPK1	Repair The B EDG Fuel Oil System Malfunction

B EDG Failure Due To An Improperly Assembled Fuel Oil Filter

On May 26, 1992, at 8:39 a. m., while conducting the biweekly surveillance test OST-401, Emergency Diesels (Slow Speed Start), on B EDG, the operator noted an increase in fuel oil filter inlet pressure and a rapid decrease in fuel oil filter outlet pressure. The operator manually tripped the EDG when the fuel oil filter outlet pressure decreased to zero and the EDG engine speed began to decrease. Prior to this time, the EDG had operated for approximately 30 minutes at rated load, 2500 Kw. Subsequent troubleshooting determined that the fuel oil filter cartridge assembly had been improperly assembled during the PMs performed earlier in the RO. The assembly was correctly reassembled and the B EDG successfully tested. Based upon the potential for a similar problem to exist on the A EDG, its fuel oil filter cartridge assembly was examined later that same day. The A EDG fuel oil filter cartridge assembly had been properly installed.

The improper assembly of the B EDG's fuel oil filter assembly is discussed below. The cartridge's outlet tube and spool piece which collects and routes the filtered fuel oil to the outlet port of the fuel filter housing had been inserted upside down. The outlet tube and spool piece is blanked off at one end to prevent unfiltered fuel oil from entering the outlet tube. The closed end is recessed approximately two inches inside the spool piece and was not readily visible. The bottom edge of the spool piece contains four small notches. When the tube and spool pieces were installed upside down, these notches had allowed some unfiltered fuel oil, a sufficient quantity to operate the engine, to directly enter the fuel filter housing outlet port. As a random occurrence, the rubber grommet gasket installed on the spool piece moved up the spool piece covering the notches. This blocked the fuel oil from the outlet port and caused the effects described above. The event is described as random since the time at which the rubber grommet gasket moved up the spool piece could not be predicted. Due to the randomness of the failure mode, post maintenance testing may or may not, as in this case, detect an improperly assembled fuel cartridge. In this instance, the B EDG had been successfully started several times after the outage PMs had been performed. The PM post maintenance testing had involved operation for approximately 8 hours at various loads including rated load for at least 1 hour.

A review of the circumstances surrounding the A and B EDG fuel filter cartridge assembly was conducted. The B EDG fuel oil filter reassembly had been performed by a member of the traveling maintenance crew assigned to the site for the RO. The mechanic which performed the work was not the same mechanic involved in the disassembly. The work order required the work to be performed in accordance with CM-508, Emergency Diesel Generator A And B Fuel Oil Filters. The steps in CM-508 to reassemble the fuel oil filter could not be performed as written. The steps specified that both the top and bottom plates, attached to the outlet tube and spring plate assembly respectively, be connected to the spool piece prior to placing a new filter element on the spool piece. With the top and bottom plates installed, the filter element can not be placed on the tube and spool pieces. If the spring plate assembly with the bottom plate is placed on top of the tube and spool pieces instead of the bottom, then this would allow the filter element to be installed after the other pieces are connected. Since this was the as found condition of the assembly, it appears that the mechanic had attempted to follow the procedural steps when assembling the fuel oil filter cartridge assembly. The procedural steps did refer to the separate pieces by item numbers which correspond to those shown on an attached figure of the assembled fuel oil filter cartridge assembly. The figure, attachment 8.1, showed the assembly correctly assembled. CM-508 had been performed on the A EDG earlier in the outage by a mechanic assigned to the site maintenance unit. Apparently he had assembled the fuel oil cartridge assembly using skill of the craft and then verified the completed assembly was in accordance with the procedure by comparing the completed assembly with than shown in the attached figure. This work practice is in accordance with the maintenance's unit procedural usage guidelines. This was the same procedure usage practice observed by the inspectors when the B EDG fuel oil filter cartridge assembly was properly reinstalled (i.e., the work was completed prior to the discovery by the inspectors that the procedure steps were out of sequence).

CM-508, Revision 0, had been issued on July 26, 1991, in accordance with the contractor assisted revised maintenance procedure upgrade program (see UNR 89-12-01 followup in IR 91-05). Revision 1, issued on February 5, 1992, to correct typographical errors was in effect on May 26, 1992. Record review indicated that the only prior use of either CM-508 Revision 0 or 1, was that associated with the A EDG during this RO. The failure to detect that the procedure was incorrect during this previous usage was a concern. The maintenance manager issued an instruction to direct that the first use of new or revised procedures be completed step by step.

This event was significant in that there existed a potential for a common mode failure. It was just happenstance that the individuals involved in the A and B EDG fuel oil filter work activity were different and that the failure occurred during testing and not during an actual demand. The incorrect steps in CM-508 were considered to be a major contributor to the improper assembly. Failure to establish adequate maintenance procedures for activities affecting safety related equipment is identified as a VIO: CM-508 Was Not Adequately Established In That Steps Provided For EDG Fuel Filter Assembly Were Out Of Sequence, 92-16-03.

Non-EQ Splices In Safety-Related MOVs

On June 16, 1992, engineering determined that CM-303, Installation Of Environmentally Qualified Taped Splices, did not contain sufficient instructions to ensure that taped splices were installed in a qualified configuration. Specifically, CM-303 did not require that all non-qualified or braided jacket material be removed from the splice area. CM-303 was used primarily for splicing motor leads. Based upon application and available information concerning the type of wiring used, components were selected to be inspected. Three MOVs (SI-869, SI-863A and SI-863B) were found to have motor leads with tape over the braided area. These valves were the SI hot leg injection flowpath isolation valves. The unqualified splices were replaced with qualified splices.

Review of CM-303 was initiated by engineering after it was identified that this procedure did not contain the same cable preparation instruction for removal of braided material as that contained in CM-309, Environmentally Sealing Low Voltage Electrical Splices. CM-309 provided instructions for use of shrinkable tubing for sealing splices. This discrepancy was apparently detected approximately two weeks earlier by an engineer during work observations on CC-716A, CCW supply header to RCS cooling water CV isolation valve. This particular valve was not required to have EQ taped splices; however, CM-303 was being used. The engineer directed the technicians to remove the braided jacket from the splice area. At that time, the engineer failed to pursue the generic implications of his observation. Only when this event was discussed in a summary report of field observations was the potential implications of the observations recognized. The failure to adequately establish procedures for the installation of EQ taped splices as required by TS 6.5.1.1.1. is a second example of VIO 92-16-03: Failure To Adequately Establish Procedure CM-303 To Install EQ Taped Splices.

MATERIAL CONDITION

As observed during the previous RO, the internal material condition of infrequently inspected components was poor. The check valve inspection of 66 valves determined that 9 valves required replacement and 16 others required replacement of one or more major parts such as discs, hinge pins or bushings. For example, all four service water pump discharge check valves (SW-374, 375, 376, and 377) required replacement of the discs, disc bushings, and pivot pins. The north service supply header check valve SW-541 had erosion/corrosion of the disc ears such that weld repairs were necessary. Both of the A EDG air start check valves (DA-20A and DA-24A) were replaced with a new type of valve because the discs were found lying in the valve bodies. Other inspections identified conditions which required replacement or repair. The C S/G MFRV manual isolation valve FW-7C required a patch to be welded over an area of localized cracking which had apparently originated at a manufacturing defect in the valve body.

The exterior material condition of some components were also determined to be poor. In particular, the SW pumps' discharge piping adjacent to the cross-connect header had severe external pitting which resulted in this piping being replaced. Furthermore, the cross-connect header also had exterior pitting which was weld repaired. While re-installing the A SW pump discharge check valve, movement of the piping resulted in the development of a through wall leak in the pump's discharge piping elbow. Inspection of Copes Vulcan valve supports, see paragraph 3, identified 18 of 45 valve supports which had miscellaneous deficiencies such as missing or loose fasteners.

One violation with two examples was identified. Except as noted above, the program area was adequately implemented.

6. ESF System Walkdown (71710)

The inspectors performed an inspection of the RHR system. The inspectors verified that the system was properly aligned in its safety standby mode and necessary instrumentation was valved into service and was functioning correctly. Inspection of system components identified only minor conditions which were identified to the licensee for correction as appropriate. Outstanding work request were reviewed to verify that no know condition existed which could prevent the system from performing its intended safety function.

The inspectors also verified that drawings were revised to reflect the new recirculation piping configuration and that these drawings were available in the control room prior to startup.

No violations or deviations were identified. Based on the information obtained during the inspection, the program area was adequately implemented.

7. Onsite Review Committee (40500)

The inspectors evaluated certain activities of the PNSC to determine whether the onsite review functions were conducted in accordance with TS and other regulatory requirements. In particular, the inspectors attended PNSC meetings on June 5 and June 12, 1992. The June 5 PNSC approved a proposed TS amendment to change the calibration method for R32A and R32B, CV High Range Area Radiation Monitors. The June 12 PNSC reviewed open issues to verify that the necessary actions were being taken to address any potential safety issue prior to restart. In particular, the small bore piping/valve support startup presentation, though very technical, provided reasonable assurance that small safety-related valves and associated piping were or would be adequately supported to withstand a DBE prior to restart. The small bore piping/valve issue is discussed in paragraph 3. It was ascertained that provisions of the TS dealing with membership and review process were satisfied.

No violations or deviations were identified. Based on the information obtained during the inspection, the program area was adequately implemented.

8. Modifications (37828)

The inspectors witnessed various modification activities associated with the following minor modifications: M-1091 (FCV-6416 Upgrade), M-1100 (Single Failure Elimination For ESF Pump Room Coolers) and M-1128 (Removal Of SI-857B Relief Valve). The inspectors performed field walkdowns of the modified systems and observed selected portions of the acceptance testing. The inspectors verified that the acceptance testing performed was adequate to demonstrate operability of the modified equipment. By record reviews the inspectors confirmed that modification comments were successfully resolved and that turnover packages were completed as necessary prior to modification acceptance by Operations.

After field installation had been completed on M-1128, the licensee identified that the modification was technically deficient. The modification had been developed to address a CV isolation issue involving the SI cold leg injection flowpath (see IR 92-07). The modification removed the single boundary relief valve, SI-857B, and reconfigured the normal system valve lineup such that another relief valve (SI-857A) upstream of two MOVs would provide overpressure protection for the lower than RCS pressure rated SI piping. Unlike SI-857B which had its discharge routed to an auxiliary building floor drain, SI-857A had its discharge routed directly to the RWST. The RWST's vent to the atmosphere was not monitored. Thus during normal operation any leakage from SI-857A would result in an unmonitored release from the site. After an accident, the SI cold leg injection pathway, containing SI-857A, would be isolated after an accident's injection phase and thus would not be a release path. Subsequently, the discharge from SI-857A was routed to an auxiliary building floor drain. The failure to incorporate in M-1128 instructions appropriate to the circumstances as required by 10 CFR 50 Appendix B Criterion V was a VIO: Instructions In M-1128 Were Not Appropriate To The Circumstances In That The Modification Created An Unmonitored Release Pathway, 92-16-04.

One violation was identified. Except as noted above, the program area was adequately implemented.

9. Self Assessment (40500)

In January 1991, NAD assumed the oversight responsibilities which were previously assigned to the onsite and corporate nuclear safety organizations. Permanent NAD onsite staffing was not completed until the site manager and engineering/ technical support supervisor positions were filled in October 1991 and March 1992, respectively.

The inspectors reviewed 1992 assessment reports in the areas of E&RC, measuring and test equipment, operations, corrective actions, and emergency preparedness. The quality of the reports have improved from those performed in the early part of the year. The earlier reports were more oriented toward problem identification, whereas the later reports focused on both problem identification and root cause determinations. The operations and corrective action audits were considered to be very beneficial. Though there was improvement in the root cause determination area, there was little progress in barrier identification (e.g., inadequate communication of management standards was determined to be the root cause of a problem, but the reason behind the standards not having been communicated was not addressed).

The Robinson NAD issue tracking system and the Robinson observation data bases were reviewed by the inspectors. As of June 25, 1992, 22 issues had been entered into the issue tracking system. The first item entered was dated September 12, 1991. As of June 25, none of the items entered into the system had been closed out. The issue tracking system contained a running history of proposed plant actions and their present status. Thus, this system was a good management tool to provide an overall current perspective on each identified issue. The observation data base contained all field observations associated with the conducted audits and thus contained supporting data for the audits, as well as providing a data base with which to analyze future observations.

Other management review processes were utilized to provide assessment of plant performance. Especially noteworthy was the Adverse Trend Meeting. This meeting of key plant managers periodically reviewed items such as LERs, NRC findings, INPO findings, NAD issues and other plant documentation such as ACRs to determine current trends. For example, in the April 7, 1992 Adverse Trend Meeting it was identified that the most prevalent causal factors were those pertaining to management standards and expectations not being effectively communicated. The specific causal factors discussed were: work practices; supervisory methods; resource management; and managerial methods.

In summary, the self-assessment processes discussed above appeared to be effective in identifying problems and root cause determinations; however, additional efforts were required to help focus in on the barriers or those specific items which need to be addressed to correct weaknesses.

10. Followup (92700, 92701, 92702)

(Closed) VIO 90-20-01, Failure To Adequately Perform Bearing Temperature Tests As Required By TS And ASME Section XI. This violation concerned numerous safety-related pump bearing temperature tests conducted from 1986 to 1990, in which stabilized bearing temperatures were not obtained as required by TS 4.0.1.a and ASME Section XI, paragraph IWP-3500. The reason stabilized bearing temperatures (i.e., variance of no more than 3 percent between three successive readings taken at 10 minute intervals) were not obtained, was the result of limiting pump operation to 30 minutes due to vendor recommended limitations on recirculation flow. Also of concern (addressed under IFI 90-20-02), was the timeliness and adequacy of the review/analysis of non-stabilized bearing temperatures by the ISI/IST coordinator. Subsequently, by letter dated July 9, 1991, NRC approved the licensee's relief request (included in the H.B. Robinson ten-year interval IST program submittal dated April 15, 1988) from performing the annual safety-related pump bearing temperature measurements. The approval of this relief request was based on the quarterly performance of vibration measurements per IWP-4500 and the unlikely detection of possible bearing failure by annual temperature measurement. Accordingly, this item is considered closed.

(Closed) IFI 90-20-02, Evaluate Adequacy And Timeliness Of ISI/IST Evaluations For Non-Stabilized Bearing Temperatures. (See Violation 90-20-01 above.)

(Closed) URI 90-22-01, Review If TMI Item Commitments Have Been Improperly Superseded By RG-1.97 Commitments. This licensee review was prompted when NRC recognized that planned modification M-1005, upgrade plant vent radiation monitoring and stack flow meter, specified a replacement high range noble gas effluent monitor with a maximum detection range one decade less than that required by NUREG-0737, item II.F.1-1. The licensee had improperly determined that item II.F.1-1 was no longer applicable based upon commitments to RG-1.97, Instrumentation For Light-Water Cooled Nuclear Power Plants To Assess Plant And Environs Conditions During And Following An Accident. Completed in July 1991, the licensee's review (which focused on NUREG-0737 Sections II and III.D) encompassed modifications which were performed to provide general compliance to H. B. Robinson's RG-1.97 program submittal, as well as those modifications performed and/or designed since the plant went into compliance with the RG-1.97 program in September 1987. As the review indicated that no similar impact to NUREG-0737 commitments were identified, the inspector had no further concerns. This item is considered closed.

(Closed) LER 91-05, TS 3.0 Entry To Adjust SI Accumulator Boron Concentration. This event (which was also the subject of NCV 91-14-02) involved the April 12, 1991, intentional draining of SI accumulator B below the minimum volume required by TS 3.3.1.1.b. The draining was part of a 40 minute evolution to restore the accumulator's boron concentration, which had been diluted below the administrative limit by RCS backleakage through the accumulator's discharge check valves (SI-875B and SI-875E). Although the inoperable SI accumulator action statement of TS 3.3.1.2.a was applied during this evolution, a subsequent literal reading of the TSs (prompted by NRC) established that the proper action statement to address this condition was provided by TS 3.0. Specifically, since TSs only address SI accumulator inoperability due to isolation (a condition that SI accumulator B was not in), entry into TS 3.0 was required. Corrective actions included: determining the RCS backleakage past SI-875B and SI 875E to be only 7 gallons/day; effecting repairs to check valves SI-875B and SI-875E during RO-14; and submitting a TS change request (dated November 27, 1991) to recognize accumulator inoperability due to volume, pressure, boron concentration, isolation, etc. As NRR is currently reviewing the TS change request; boron concentration in the B accumulator had not gone below the minimal limit of TS 3.3.1.1.b; RCS leakage at no time approached the 1 gpm limit of TS 3.1.5; and post maintenance leak testing of check valves SI-875B and SI-875E (per OST-160, Pressure Isolation Check Valve Back Leakage Test) was satisfactory, the inspector had no further concerns. Accordingly, this item is considered closed.

(Closed) LER 91-10, Source Range Reactor Trip While In Hot Shutdown. As discussed in IR 91-19, the unit had been shut down on August 16, 1991, in order to resolve concerns with time delays associated with OT Delta T RPS circuitry. After being shut down for over an hour with all but the A shutdown bank control rods inserted, a reactor trip occurred when source range instrumentation was manually energized. The manual energization of source range instrumentation (i.e., pushing the "permissive P-6 defeat" buttons) was initiated under the guidance of OP-002, Nuclear Instrumentation System, when the P-6 permissive associated with intermediate range channel N-36 would not clear. Subsequently, when energized, source range channel N-31 indicated above the high flux level trip and a reactor trip resulted. The cause of the trip event stems from the replacement of both source range and both intermediate range detectors during RO-13 due to water intrusion. The discriminator (source range) and compensating (intermediate range) voltages had been set using a depleted fuel assembly as the radiation source since the newly installed secondary sources had not yet been activated. Consequently, when the detectors were exposed to a full, recently shutdown core with activated secondary sources, they over responded due to their conservative calibrations. Hardware related corrective actions included: (1) retracting/moving the source and intermediate range detectors in order to reduce the incident neutron flux, and (2) readjusting the discriminator and compensating voltage settings. The inspector reviewed associated ACR 91-285 and confirmed that training on this event (encompassing nuclear instrument calibrations following detector replacement or changes to plant conditions that affect performance with respect to this event) had been incorporated in the reactor engineer qualification program. Accordingly, this item is considered closed

(Closed) LER 91-13, Diesel Driven Fire Pump Inoperable. This LER concerns the "Technical" inoperability of the diesel engine driven fire pump and the subsequent failure to submit a special report per TS 3.14.2.2.. As part of the TS required operability demonstration for the diesel engine driven fire pump, the licensee inspects the diesel engine (per TS 4.14.6.1.b.1) in accordance with PM-103. During the performance of PM-103 on April 29, 1991, the diesel engine's reservoir oil temperature was determined to be less than the manufacturer's minimum requirement of 120 degrees F. This condition was subsequently identified via ACR 91-324 on September 10, 1991, resulting in the engine driven fire pump being declared inoperable on September 17, 1991. Although concluding that the fire pump would have operated as required (based on vendor information), the licensee determined that a violation of TS 3.14.2.2 occurred since the fire pump had not been repaired (i.e., temperature problem corrected) within 7 days from April 29, 1991, and a special report was not submitted to the Commission within the following 30 days. The inspector reviewed ACR 91-324 and associated ACR 91-333, which resulted in the following corrective actions being taken: replacement of original 100-120 degrees F thermostat with a 120-150 degrees F thermostat per M-445P and DCN-NUS-245; replacement of original 150 watt oil heater with a 300 watt oil heater based on the manufacturer's recommendation; and revising PM-103 to obtain/document as found (i.e., prior to drain/refill activities for filter replacement) oil reservoir and water jacket temperatures, specify appropriate temperature monitor locations, and recognize TS implications if temperatures are less than 120 degrees F. As subsequent performance of PM-103 has demonstrated satisfactory resolution to this event, this item is considered closed.

(Closed) LER 92-009, Safety Related Hydraulic Snubber Not Included In Snubber Inspection Program. As discussed in paragraph 3, a snubber was identified which was attached to SI-850D, but had not been included in the TS required inspection program. The snubber was found to be inoperable (i.e., the snubber contained no hydraulic fluid). Engineering analysis determined that the associated piping met short-term criteria and therefore remained operable. The snubber was repaired and returned to service. Inspections were conducted as part of the Copes Vulcan valve support inspections and the licensee's self-initiated Piping Improvement Program inspections to locate other snubbers which were not included in the Snubber Inspection Program. No other examples were identified. Failure to include the snubber in the inspection program is a violation of surveillance requirements contained in TS 4.13. This violation will not be subject to enforcement action because the licensee's efforts in identifying and correcting the violation met the criteria specified in Section VII.B of the Enforcement Policy. This is identified as an NCV: Failure To Include A Snubber In The Snubber Inspection Program, 92-16-05. This item is considered closed.

One NCV was identified.

11. Exit Interview (71701)

The inspection scope and findings were summarized on July 9, 1992, with those persons indicated in paragraph 1. The inspectors described the areas inspected and discussed in detail the inspection findings listed below and in the summary. On July 17, 1992, plant management was informed of recharacterization and/or deletion of some report findings. Dissenting comments were not received from the licensee. The licensee did not identify as proprietary any of the materials provided to or reviewed by the inspectors during this inspection.

Item Number	Description/Reference Paragraph
92-16-01	NCV - Failure To Have Small Bore Safety Related Piping/Valves Adequately Supported (paragraph 3).
92-16-02	NCV - Failure To Have The Shutdown Sequence Relay Installed In The A EDG Control Circuit As Required By Plant Drawings (paragraph 4).
92-16-03	VIO - CM-508 Was Not Adequately Established In That Steps Provided For EDG Fuel Filter Assembly Were Out Of Sequence (paragraph 5) and Failure To Adequately Establish Procedure CM-303 To Install EQ Taped Splices (paragraph 5).
92-16-04	VIO - Instructions In M-1128 Were Not Appropriate To The Circumstances In That The Modification Created An Unmonitored Release Pathway (paragraph 8).
92-16-05	NCV - Failure To Include A Snubber In The Snubber Inspection Program (paragraph 10).

12. List of Acronyms and Initialisms

a.m.	Ante Meridiem
ACR	Adverse Condition Report
ASME	American Society of Mechanical Engineers
CC	Component Cooling
CCW	Component Cooling Water
CFM	Cubic Feet Per Minute
CFR	Code of Federal Regulations
CM	Corrective Maintenance

CV	Containment Vessel
CVC	Chemical & Volume Control
DBD	Design Basis Documentation
DBE	Design Basis Earthquake
DCN	Design Change Notice
EDBS	Equipment Data Base System
EDG	Emergency Diesel Generator
EQ	Environmental Qualification
ESF	Engineered Safety Feature
EST	Engineering Surveillance Test
F	Fahrenheit
FCV	Flow Control Valve
FHP	Fuel Handling Procedure
FW	Feedwater
gpm	gallons per minute
HDP	Heater Drain Pump
Hx	Heat Exchanger
i.e.	That is
IEN	Inspection Enforcement Notice
IFI	Inspector Followup Item
ILRT	Integrated Leak Rate Test
INPO	Institute Of Nuclear Power Operations
IR	Inspection Report
ISI	Inservice Inspection
IST	Inservice Test
KW	Kilowatt
LCO	Limiting Condition for Operation
LER	Licensee Event Report
M	Modification
MFRV	Main Feedwater Regulating Valve
MOV	Motor Operated Valve
MST	Maintenance Surveillance Test
NAD	Nuclear Assessment Department
NCV	Non-cited Violation
NED	Nuclear Engineering Department
NRC	Nuclear Regulatory Commission
NRR	Nuclear Reactor Regulation
OP	Operations Procedure
OST	Operations Surveillance Test
OT	Overtemperature
OT Delta T	Overtemperature Delta Temperature
P	Permissive
p.m.	Post Meridiem
PACV	Post Accident Containment Vent
PLSA	Part Length Shield Assembly
PM	Preventive Maintenance
PNSC	Plant Nuclear Safety Committee
psig	Pounds per square inch - gage
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RG	Regulatory Guide
RHR	Residual Heat Removal

RO	Refueling Outage
RPS	Reactor Protection System
RTD	Resistance Temperature Detector
RTGB	Reactor Turbine Generator Board
RWST	Refueling Water Storage Tank
RVLIS	Reactor Vessel Level Indication System
SDAFW	Steam Driven Auxiliary Feedwater
S/G	Steam Generator
SI	Safety Injection
SIT	Structural Integrity Test
SS	Shift Supervisor
SW	Service Water
Tavg	Temperature - average
TMI	Three Mile Island
TS	Technical Specification
UNR	Unresolved Item
URI	Unresolved Item
V	Ventilation
VIO	Violation
WR/JO	Work Request/Job Order