UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
101 MARIETTA ST., N.W., SUITE 3100
ATLANTA, GEORGIA 30303

Report No. 50-261/82-27

Licensee: Carolina Power and Light Company
411 Fayetteville Street
Raleigh, NC 27602

Facility Name: H. B. Robinson Steam Electric Plant

Docket No. 50-261

License No. DPR-23

Inspection at H. B. Robinson Unit 2

Inspector: A K Hardin for 9/1/82
S. Weise — Date Signed

Approved by: C. W. Burger 9/2/82
C. Burger, Section Chief, Division of Project and Resident Programs — Date Signed

SUMMARY

Inspection on July 11 - August 10, 1982

Areas Inspected

This routine, announced inspection involved 148 resident inspector-hours on site in the areas of technical specification compliance, plant tour, operations performance, reportable occurrences, housekeeping, site security, surveillance activities, maintenance activities, quality assurance practices, radiation control activities, outstanding items review, IE Notice Followup, TMI Action Item review, refueling startup testing, modifications, refueling surveillances, and enforcement followup.

Results

Of the 17 areas inspected, no violations or deviations were identified in 16 areas; two violations were found in one area.

DETAILS

1. Persons Contacted

   Licensee Employees

   +*R. B. Starkey, Plant General Manager
   +J. Curley, Manager Technical Support
   +*F. Gilman, Project Specialist, Regulatory Compliance
   *F. Lowery, Unit 2 Operations Supervisor
   W. Crawford, Manager, Operations and Maintenance
   *R. Chambers, Unit 2 Maintenance Supervisor
   *C. Wright, Specialist, Regulatory Compliance
   S. Crocker, Manager, Environmental & Radiation Control
   +*J. Young, Director Corporate QA/QC

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

   Other Organizations

   R. Muth, Westinghouse

   *Attended exit interview on July 27, 1982
   +Attended exit interview on August 12, 1982

2. Exit Interview

   The inspection scope and findings were summarized on July 27 and August 12, 1982 with those persons indicated in paragraph 1 above. The licensee acknowledged the violations. The commitments discussed in paragraphs 8 and 13 were requested by the inspector and agreed to by the licensee.

3. Licensee Action on Previous Inspection Findings

   (Closed) Unresolved item 81-33-01. This item dealt with the PORV block valve apparent design deficiency. As discussed in paragraph 13, the valve problems were not due to inadequate design but due to inadequate maintenance. The original indication of a design problem was due to confusion over the limitorque valve operator characteristics and recent EPRI testing results. Further discussions with the valve and operator manufacturers' representatives determined that valve design was more than adequate.

   (Open) Severity Level IV Violation 81-36-01. This item concerned the licensee's failure to implement modification procedures and to perform an adequate safety review of PORV deficiencies. The inspector reviewed the CP&L response letter dated March 10, 1982. The licensee has revised Maintenance Instruction 10, Procedure 1 to include spring tension adjustment requirements. Similar air operated valves are being reviewed to provide the same type guidance to prevent further problem recurrence. Additionally, a Plant Nuclear Safety Committee Action Item was created to emphasize the necessity of adequate review. This item will remain open until the action item is closed and necessary procedures revised.

(Closed) Severity Level V Violation 81-36-01. This item concerned a lack of procedural guidance on PORV and block valve use. The inspector reviewed the CP&L response letter dated March 10, 1982. Corrective action appears adequate and complete.

(Closed) Unresolved item 81-19-01. Amendment 70 to Facility Operating License DPR-23 has corrected Technical Specification 6.2 figures to represent the accurate offsite and onsite organizations.

4. Unresolved Items

Unresolved items were not identified during this inspection.

5. Plant Tour

The inspector conducted plant tours periodically during the inspection interval to verify that monitoring equipment was recording as required, equipment was properly tagged, operations personnel were aware of plant conditions, and plant housekeeping efforts were adequate. The inspector determined that appropriate radiation controls were properly established, excess equipment or material was stored properly, and combustible material was disposed of expeditiously. During tours the inspector looked for the existence of unusual fluid leaks, piping vibrations, pipe hanger and seismic restraint abnormal settings, various valve and breaker positions, equipment clearance tags and component status, adequacy of firefighting equipment, and instrument calibration dates. Some tours were conducted on backshifts. The inspector performed major flowpath valve lineup verifications and system status checks on the following systems:

a. Selected containment isolation valves

b. Safety Injection System

c. Motor Driven Auxiliary Feedwater System

d. Boration paths

6. Technical Specification Compliance

During this reporting interval, the inspector verified compliance with selected limiting conditions for operation (LCO's) and reviewed results of selected surveillance tests. These verifications were accomplished by direct observation of monitoring instrumentation, valve positions, switch positions, and review of completed logs and records. The licensee's compliance with selected LCO action statements were reviewed as they happened.

a. On July 15, 1982, with the plant at 275°F during conduct of steam generator crevice flushing evolutions, the Auxiliary Operator reported a potential service water leak on a containment air recirculation unit (HVH-2) motor cooler. This leak was a breach of containment integrity. The licensee inspected the cooler and determined that a cooler leak did exist. An Unusual Event was declared and reported to the NRC. The event was terminated when service water to the cooler was isolated and when reporting was completed. This placed the plant in a 48 hour LCO, and the required Containment Spray System testing was completed satisfactorily. The licensee repaired the cooler within the LCO requirements and reported the event in Licensee Event Report 82-06.

b. On July 17, 1982, a primary to secondary leak of about 3-5 gpm developed in a S/G. The plant was heating up following a refueling outage and was at about 435°F and 1500 psig when the leakage was confirmed. A plant cooldown was immediately commenced. The leak was investigated and found to be in hot leg tube R8C52. This tube was worked during the refueling outage due to a known minute leak. Based on subsequent leakage, the licensee and Westinghouse determined that an attempted weld repair would not be feasible, probably due to some form of chemical contamination in the weld area. The decision was made to rebore the tube to prepare it for use of a mechanical plug. This reboring process quite possibly exposed a .10 - .18 inch ring of carbon steel tubesheet where the tube wall was shaved away. This exposed area is just under the tubesheet inconel cladding in the area where the tube is rolled into the tubesheet. The inspector reviewed the Westinghouse safety evaluation concerning expected corrosion rates of the exposed carbon steel. Based on the expected corrosion rates and the 2-2¼ inch length that the tube is rolled into the tubesheet, the inspector was satisfied that an unreviewed safety question did not exist. The licensee plans to conduct mock-up tests of this condition to ensure no accelerated corrosion mechanism will affect this area, and steam generators will probably be replaced in 1984-1985. Repairs and testing were completed on the tube on July 26, and plant heatup recommenced. The event was reported to the NRC as required and is discussed in LER 82-07.

c. On July 31, 1982, with the plant in hot shutdown at 530°F and 2300 psig, the control operator noted the rod position indication (RPI) for rod L-11 reading significantly lower than the other shutdown bank rods. Shutdown banks had been pulled to 228 steps earlier and rod L-11 had indicated correctly. Technicians investigating the RPI problem discovered a steam leak from the top of the L-11 drive mechanism pressure housing where the eyebolt threads into the housing. Reactor coolant system leakage rate was slightly less than 0.25 gpm. The licensee decided to return to cold shutdown conditions and repair the leak. The problem was identified as poor seating of the needle valve after operation earlier in the outage, and the valve was replaced. This event will be reported in a future LER.

7. Plant Operations Review

a. The inspector periodically during the inspection interval reviewed shift logs and operation records, including data sheets, instrument traces, and records of equipment malfunctions. This review included control room logs, auxiliary logs, operating orders, standing orders, jumper log and equipment tagout records. The inspector routinely observed operator alertness and demeanor during plant tours. During abnormal events, operator performance and response actions were observed and evaluated. The inspector conducted random off-hours inspections during the reporting interval to assure that operations and security remained at an acceptable level. Shift turnovers were observed to verify that they were conducted in accordance with approved licensee procedures.

b. Prior to plant startup after the refueling outage, the inspector performed a walkthrough of portions of the Reactor Coolant system, Nuclear Instrumentation system, Control Rod Drive system, and the Safety Injection and Containment Spray systems. With the following exception, no discrepancies were noted.

On July 13, 1982 the inspector conducted a system walkdown on portions of the Safety Injection and Containment Spray Systems. This walkdown was conducted after the licensee lineup had been completed but prior to reaching 200°F reactor coolant temperature. During the walkdown the inspector determined that valve SI-892D, Spray Additive Tank Fill Valve, was open when it should have been closed. The effect of this error would be to dilute the sodium hydroxide solution used during containment spray, but would allow its addition. Because the plant had not reached 200°F at the time of discovery, the system was not required to be operable. The licensee corrected the valve position immediately and conducted an investigation. The licensee determined that the valve was opened by two Instrumentation and Control technicians performing Maintenance Procedure MP1-10. This procedure provides instructions for lining up safety-related instrument valves for plant startup. Section 7.5 required the technicians to lineup flow transmitter FT-949 (Spray Additive Flow) despite the fact that this FT is an inline rotameter. The technicians, misunderstanding the flow path, thought SI-892D was an instrument isolation valve. Based on the above and a review of the procedure revision 0, the inspector determined the procedure was inadequate in that it 1) included valve lineup requirements for FT 949 and six Safety Injection Accumulator level instruments that should not require lining up by I&C personnel and 2) did not specify for the instruments listed which type of isolation valve setup shown in the procedure was applicable. Inclusion of item 2) should insure the proper valves are checked and that the independent verifiers expect to see the same valve arrangement. The licensee's immediate corrective action was to review the procedure for additional problems and reverify those system valves that had the potential to be affected. This was completed prior to exceeding 200°F. Based on the observed procedure inadequacies, the failure to adequately review the procedure prior to approval and use constitutes a violation. (50-261/82-27-01).

c. On July 12, 1982,the inspector conducted a review of safety-related system valve lineups in order to verify that systems disturbed during the refueling outage were returned to an operating status. Of those valves lineups completed and filed, the inspector determined that the following Operating Procedure (OP) valve lineups were not conducted using current approved valve checkoff lists:

1. OP-25A, Reactor Coolant System was completed on July 9, 1982 using revision 8. Revision 9 was approved on June 10, 1982 and added valve RC-540A.

2. OP-40A Component Cooling Water System was completed on July 11, 1982 using revision 6. Revision 7 was approved May 3, 1982 and added valves CC-788G and CC-788GG installed by a plant modification.

3. OP-45A Isolation Valve Seal Water was completed on July 12, 1982 using revision 4. Revision 5 was approved October 30, 1981 but constituted only a retyping.

4. OP-9A Instrument and Station Air was completed July 10, 1982 using revision 10. Revision 11 was approved June 28, 1982 and added instrument air isolation valve IA-390 to V12-24A (air to containment) used in conjunction with the Post-Accident Containment Venting System.

The licensee investigated these occurrences and preliminarily determined that the master list used to verify the current procedure revision was in some cases not up-to-date and in other cases was apparently not used. The licensee immediately initiated a review of all safety-related valve checkoffs and checked those valves missed due to the superceded valve lineups. Failure to prevent the use of superceded documents is a violation. (50-261/82-27-02).

d. On July 13, 1982 the Plant Nuclear Safety Committee (PNSC) approved a temporary repair to the containment air recirculation fan motor coolers due to known and potential leakage. Cooler tube degradation is a breach of containment integrity in the event of an accident. The inspector reviewed Temporary Repair 82-05 and the accompanying safety evaluation. This evaluation only addressed the pressure retaining property of the polymer material used in the leak repair. The inspector raised the concern that the safety evaluation did not address the post-accident environment that the motor cooler could be exposed to. An engineering evaluation was made for peak containment temperature and determined the material was acceptable. The inspector then questioned the material's susceptability to post accident chemical attack. The licensee then evaluated this area and found the material adequate and also determined that the sealed unit design of the motor cooler should prevent entrance of the post-accident chemical environment. This satisfied the equipment concerns, however, the inspector is concerned that the licensee's safety evaluation procedures and/or training for personnel conducting safety reviews for design changes (permanent or temporary) are inadequate. This item will remain open until the licensee investigates this inadequate safety evaluation and PNSC review and determined appropriate corrective actions. (50-261/82-27-03).

e. During the inspection period, the inspector observed plant startup, heatup, and approach to criticality. The inspector verified that the startup activities were performed in accordance with approved, adequate procedures and that Technical Specification requirements were met. Startup procedures were reviewed to ensure that applicable prerequisites were met, including surveillance testing. The reactor achieved criticality on August 10, 1982 and the inspector verified that the critical boron concentration was acceptable when compared with the predicted value. Low power physics testing was beginning at the end of the inspection period and will be documented in a future report. The inspector also observed several reactor coolant boron analyses for procedure adherence and laboratory technique.

8. Plant Trips

a. On July 16, 1982 with the plant being heated up and feeding steam generators, the reactor trip breakers opened on steam flow and feedwater flow mismatch with coincident low level associated with B steam generator (S/G). S/G level instrument LI-485 was out of service for maintenance and its associated bistables tripped. A brief steam flow spike occurred causing the reactor trip. This steam flow spike was believed to have resulted from a voltage spike caused by maintenance on the LI-485 level instrument electronics. S/G instrumentation sensing lines were also vented to remove any foreign material.

b. On July 17, 1982, with the plant being cooled down for S/G maintenance, a reactor trip occurred due to a voltage spike on instrument bus no. 4. The voltage spike apparently spiked the first stage turbine pressure instrument, clearing permissive P-7. Because plant parameters were consistent with cooldown conditions vice power operation, a trip of the shutdown banks (only rods out) occurred. No associated maintenance was in progress at the time. The licensee investigated the event and determined that the affected instrument buses had numerous loose terminations. The inspector determined that no formal preventive maintenance program existed to conduct cleaning and termination tightening on safety-related instrument buses. The licensee committed to instituting such a preventive maintenance program prior to the next refueling outage (IFI 82-27-10).

c. On August 5, 1982 with the plant in hot shutdown, a reactor trip of the shutdown banks occurred on nuclear instrumentation high flux trips due to a temporary loss of voltage to instrument bus no. 1. The loss of voltage to instrument bus no. 1 was caused by a failure to defeat the degraded grid voltage feature when starting 'C' reactor coolant pump. The attendant drop in voltage caused a loss of emergency bus E-1, which feeds instrument bus no. 1, until the diesel generator picked up the bus. An unusual event was declared which was terminated when the normal electrical lineup was restored. The Plant General Manager has taken corrective action with the shift involved.

9. Physical Protection

The inspector verified by observation and interview during the reporting interval that measures taken to assure the physical protection of the facility met current requirements. Areas inspected included the organization of the security force, the establishment and maintenance of gates, doors and isolation zones in the proper condition, that access control and badging was proper, that search practices were appropriate, and that escorting and communications procedures were followed.

10. Refueling Startup Testing (72700)

The inspector observed the control rod drop-time testing, Periodic Test R 4.10.2, to verify it was performed in accordance with technically adequate, approved procedures and met Technical Specification (TS) requirements. The inspector observed the testing from the control room and the rod drive cabinet area. Drop-time strip charts were reviewed and all rods satisfied the requirements of TS 3.10.4.1. Due to the problems encountered with rod L-11 during Cycle 8 (IE Report 50-261/80-32) and the installation of new drive shafts at L-11, L-5, and K-4, the licensee instituted a program to determine drive shaft alignment and to ultrasonically evaluate the rods during drop-time testing. This program is intended to identify those drive shafts to be replaced during future refueling outage in an attempt to avoid control rod inoperability problems. No violations or deviations were noted.

11. Modifications (37700)

The inspector reviewed several modifications discussed below that did not require NRR approval. Specifically, the inspector verified that: 1) modifications were reviewed and approved in accordance with 10 CFR 50.59, Technical Specifications, and quality assurance controls, 2) modifications were performed and tested in accordance with established procedures, 3) that test results were acceptable, and 4) that procedure and drawing changes were made or were in progress.

a. Modification-573 provided for unganging the steam generator blowdown and sample valves to allow phase A containment isolation reset on a line by line basis.

   The inspector reviewed the modification package, discussed the modification with licensee and contractor personnel, and observed portions of the circuit wiring and post-modification testing. Installation and testing appeared to be adequate and met the licensee's

TMI Action Plan commitments. The inspector noted that the change was not properly identified as a change to the FSAR, however, necessary safety reviews were adequately completed. The licensee indicated and the inspector concurred that the FSAR update program identified the modification as a change. Procedural changes to Emergency Instruction-1 were reviewed and appeared acceptable. No violations or deviations were noted.

b. Modification-629 was issued to relocate the steam-driven auxiliary feedwater (AFW) pump steam supply valves and add manual isolation valves for maintenance.

The inspector reviewed portions of the modification package, discussed the modification with licensee and contractor personnel, and observed portions of the piping and valve installation and circuit wiring. Portions of the post-modification testing were observed, including testing under hot plant conditions. Valve wiring deficiencies were encountered during testing, and corrective action taken to adequately correct these items. Procedural changes to Operating Procedure-17A and Standing Order-14 were reviewed and found acceptable. Revision 3 to this modification resulted in a change to Technical Specification 3.13 in that additional snubbers were added to the steam driven AFW pump inlet line. This change is allowed without a prior licensee amendment provided the revisions are included in a subsequent amendment request. The inspector also questioned licensee management on the need to review the requirements of IE Bulletin 81-01 and develop an inspection program. An action item has been initiated by the licensee to address this issue. Completion of a Technical Specification amendment and development of a mechanical snubber inspection program are an inspector followup item. (82-27-04).

c. Modification-445T to add emergency communications and lighting capabilities in the event of a fire with loss of existing capabilities.

The inspector reviewed the modification package, discussed lighting and communications system locations with licensee personnel, field verified selected lighting and communications stations, periodically tested selected emergency lights, and observed selected lights during a station blackout surveillance. The inspector noted several poorly lit safety-related equipment areas and determined that the licensee had identified these and other deficiencies. While the modification appears to meet 10 CFR 50 Appendix R requirements, the licensee intends to upgrade the lighting in selected plant areas. (IFI 50-261/82-27-05). A plant surveillance procedure has been approved and performed and appeared adequate. No violations or deviations were noted.

d. Modification-518 was issued to provide keyswitches vice jumpering for defeat of steam generator low-low level protective features in cold shutdown.

The inspector reviewed portions of the modification package, discussed the modification with appropriate licensee personnel, observed portions of the circuit wiring and post-modification testing. Installation and testing appeared to be adequate to ensure that safety system response was not affected. Procedure changes were made to General Procedures-2, 3A, 5, 6 and Operating Procedure-17-1A. The inspector reviewed the procedures, and they appeared acceptable. No violations or deviations were noted.

e. Modification-604 was issued to monitor for radioactive contamination in the service water discharge lines from the containment air recirculation cooling unit motor coolers.

The inspector reviewed portions of the modification package, discussed the modification with licensee and contractor personnel, and observed portions of the valve and piping installation and nondestructive testing. Intended testing appeared adequate to meet ASME Code and USA B31.1 (1967) requirements. The licensee has not made the system operable due to the need to verify that flow through the system is in the correct direction at appropriate flow rates. The licensee does not expect the necessary equipment to make this verification will be available for several weeks. (IFI 50-261/81-31-03).

12. Refueling Surveillance

The inspector conducted a comprehensive review of those tests and calibrations required by Technical Specifications to be conducted on a refueling interval. The inspector reviewed this area to see that formal procedures existed for conduct of the calibrations and/or tests, that a scheduling program existed to insure that all surveillances were scheduled, and that refueling/startup procedures verified that the surveillances had been conducted. The inspector determined that all Technical Specification surveillances appeared to have been satisfactorily conducted, but found the following deficiencies:

a. Calibration of the 480 volt bus item undervoltage relays as required by Technical Specification Table 4.1-1, item 32a, is not in a scheduling program. The calibration was conducted by work request, and scheduling is accomplished by review of the past refueling outage work request index. The inspector reviewed the work request and discussed the calibration procedure with licensee personnel. Instrument and control technicians have no written implementing procedure for conducting this calibration. Failure to have a written implementing procedure for this Technical Specification required calibration is an open item (50-261/82-27-06).

b. Testing of the main feed pump breaker auto-initiation of auxiliary feedwater (AFW) system circuitry is not on any schedule, and was not scheduled. A surveillance procedure does not exist for conducting this testing as required by Technical Specification Table 4.8-1, item e. Due to a modification (518) to the AFW circuitry conducted during this outage, the inspector is satisfied that the post-modification testing was adequate to cover this item. Adequate scheduling and an approved surveillance procedure are needed to ensure Technical Specification compliance in the future. The failure to have a written implementing procedure for this Technical Specification required test is incorporated in the open item noted in subparagraph a. above. (50-261/82-27-06)

c. The licensee has several methods of scheduling Technical Specification required surveillances:

(1) Use of surveillance procedures such as Periodic Tests (PT), Health Physics (HP) procedures, and Maintenance Instructions (MI).

(2) Use of preventive maintenance scheduling sheets by planners. These sheets are developed by equipment area and calibration frequency. Their function is to prompt the planners to initiate calibration work requests on the equipment (both Technical Specification and non-Technical Specification).

(3) Review of past outage work request index sheet for items previously conducted but not covered under 1) and 2). This method gives no priority to Technical Specification items in that they are not identified as such.

From the above, it appear concerned that no formal comprehensive program exists to ensure that all Technical Specification required surveillances and calibrations are scheduled. (Open Item 82-27-07).

d. The licensee has several methods of verifying that Technical Specification required refueling surveillances are performed prior to plant startup:

(1) Periodic Test 1.0, Overall Refueling Interval Test Procedure, which has the stated purpose of outlining those procedures which are to be completed at refueling to comply with all Technical Specification requirements. This procedure is not up-to-date in that at least the following refueling Technical Specification surveillance procedures are not addressed:

Procedure	Technical Specification
PT 5.8 - Low Temperature Over Pressure Protection	Table 4.1-3, item 16
PT 6.2 - 480 Volt Degraded Voltage	Table 4.1-1, item 32.b
PT 22.3 - AFW Flow Indication	Table 4.1-1, Item 33
PT 25.5 - Safety Relief Valve Position Indication	Table 4.1-1, item 37

PT 31.1 and 31.2 - Inspection and Testing of Hydraulic Shock Suppressors	3.13/4.13
PT 42 -PORV and Block Valve Position Indication	Table 4.1-1, items 35, 36
PT 51 - Post Accident Sampling Valves Local Leak Rate Test	4.4.2.a
HP-42 and MI-4 - Radiation Monitoring System	Table 4.1-1, item 19

(2) General Procedures GP-2 and 3A for plant heatup and startup require preliminary review of some refueling and operational PTs to ensure startup prerequisites are met. The items listed in 1) above do not appear in either of these procedures.

The inspector identified no T.S. surveillances that were not conducted, therefore the inspector will monitor licensee correction of this deficiency. (Open item 50-261/82-27-08.)

13. Power Operated Relief Valves (PORVs) and Block Valve Issues

a. The deficiencies and issues of this item have been previously discussed in IE Inspection Reports 50-261/81-32, 81-33, 81-36, and 82-11. The licensee has completed his investigation and will submit a supplemental report to Licensee Event Report 81-31. The licensee has determined that the block valve was correctly designed but did not meet its design specifications due to lack of periodic maintenance. This problem is to be corrected by implementation of a preventive maintenance program on the block valves. The inspector has reviewed the applicable supporting documentation which is summarized below.

1) Original PORV and block valve design: Through discussions with licensee and Westinghouse personnel, the inspector learned that the original design criteria (prior to 1971) was to consider the PORV as a special type of relief valve which did not require two valves between the reactor coolant system and downstream components. Therefore, the block valves were added as maintenance valves to allow isolation of PORV leakage during the period between refuelings should it develop. The block valves could also be used to allow PORV repairs. Isolation of the PORVs is allowed since no credit for the PORV's is taken in the FSAR Accident Analyses. The small break LOCA analyses (Volume III of WCAP 9601) are reported to bound the potential for a PORV and block valve to fail open simultaneously. the designer, however, did design the block valves to shut against a 2500 psi differential.

2) Condition of PORV block valves: Licensee, Westinghouse, Velan, and Limitorque personnel inspected and repaired the block valves.

This inspection found the following lack of preventive maintenance:

(a) The grease inside the main housing for the valve gears was low and hard.

(b) All the gaskets were either damaged or missing allowing moisture to corrode limit switches.

(c) Cover bolts were either missing or loose.

(d) The valve worm gear in one valve was replaced due to wear.

(e) The valve packing was old and adhered to the value stem and was of a type not recommended by Velan

3) Design versus as-left condition of the block valves: Based on the design of the Robinson block valve and the forty second closing requirement, the valve operator must provide 79 ft-lb of torque to close the valve. For the Robinson motor operator, this corresponds to a torque setting of two. After refurbishment, the valve operator was removed and a torque wrench used to determine output torque versus torque setting. For the torque setting of 2, Westinghouse personnel determined this yielded 102-ft-lb torque.

   The licensee has decided to leave the valve torque setting at 3 to account for any valve deterioration over the present operating cycle. This corresponds to a design torque of 109 ft-lb. The licensee and valve manufacturer have evaluated this torque value and determined that valve damage should not occur. The valve torque limiter device will not allow a torque setting greater than five. Based on the data above, it appears that the refurbished block valve is capable of satisfying its design specification. Westinghouse has also compared the Robinson block valve data with the EPRI block valve data and expects the valves to perform as required.

4) Analysis of PORV block valve flow: Based on the review of Westinghouse calculations, it appears that the open PORV is the limiting orifice until the block valve reaches 30% from full closed. The block valve limits flow from fully closed to 30% open.

5) PORV spring tension testing and instrument air modification: Modification-418, Overpressure Protection System, was installed using a PORV setpoint of 400 psig and a two second opening time utilizing an instrument air pressure regulator delivering about 95 psig. The spring compression for the PORV's was set for a 2300 psi opening pressure and the PORV achieved a two second opening time. However, under Modification-480, PORV Valve Internal Replacement, the spring compression was reduced to achieve the two

second opening time since packing tightness increased this time beyond two seconds. On December 12, 1981, in order to determine the real opening response of the PORV's, Special Procedure-353 was conducted with no pressure on the PORV plug. Both valves opened in two seconds or less, which was conservative to having plant pressure under the plug. However, due to the regulatory set pressure being close to instrument air inlet pressure at the regulator, the licensee determined that removal of the 95 psig regulators would provide increased instrument air operating pressure for valve actuation. Post-modification testing indicated valve closure times of 1.7 and 1.8 seconds respectively.

b. On July 20, 1982, the inspector held discussions with licensee personnel to determine CP&L's position on use and maintenance of the PORVs and block valves. The PORVs and block valves will be used as per the original design with the PORVs set up to respond to plant pressure transients and the block valves open. The use of Standing Order 17 to address interim valve use and operability has been terminated. Since the PORVs are used cold for the Low Temperature Overpressure Protection System, they will be stroked and timed prior to that usage on cooldown and any necessary spring adjustments procedurally controlled.

c. On July 28, 1982, with the plant at 530°F and 2280 psig, the control operator attempted to shut valves RC-535 and 536 (block valves) in preparation for the reactor coolant system leakage test and inspection. Valve 535 went to an intermediate position and had to be closed manually. Valve 536 did not appear to leave its backseat. Personnel sent in containment to investigate and close the valves, operated the 536 declutch lever to allow the motor to close the valve. The valve was then driven closed into the seat and continued to operate, causing failure of three of the four actuator to valve yoke bolts. The breaker was opened to deenergize the motor. Investigation into this event yielded the following:

1) Because of the valves' location above the pressurizer in the pressurizer cubicle, they experience a larger temperature change which affects packing loading. Since the valves are packed cold, they require special maintenance attention hot to ensure packing tightness and stem lubrication are conducive to proper valve stroke. The licensee experienced considerable difficulty in setting up these valves to ensure proper operation. Additionally, maintenance personnel used the wrong stem lubricant during the valve refurbishment during the refueling outage. This was corrected after the July 28 problems.

2) The operator, contrary to instructions on the declutch lever, forced the declutch lever into the motor operate position and affected the actuators performance. The licensee has developed a report for training of personnel in proper operation of limitorque operators. This is to be disseminated to operations, maintenance, and engineering personnel.

3) Valve RC-536 continued to operate after valve closure due to the torque switch assembly being installed too loosely. Thus, the torque switch feature did not actuate to deenergize the valve motor when the valve reached the closed position.

4) The bolts on valve RC-536 broke because they were of the wrong material. The bolts used had a yield strength of 70 Ksi vice the 125 Ksi required by design.

5) The valve steam of valve RC-536 was inspected for damage and none was found. The valve manufacturer indicated that the torque experienced by the valve was insufficient to damage the gate valve internals.

Based on the above, block valve maintenance is not being adequately controlled. While the valve actuators are treated as non-Q, better control of operations and maintenance activies affecting the valves is necessary if they are to remain operable for leak isolation. Licensee management agreed to develop an operating experience report for use in upgrading maintenance activities on these valves. (IFI 82-27-09)

d. During early August, further maintenance was conducted to correct actuator and packing problems. The block valves were cycled every 100°F increase in pressurizer temperature during plant heatup to hot shutdown. The valves operated properly and within their required stroke times. The licensee instituted a surveillance program to test the valves with periodicity to be determined by valve performance. Eventually the licensee expects to return the valve testing to its inservice inspection (ISI) frequency. The inspector verified that the ISI procedures were revised to test the valves under hot plant conditions.

Based on the above reviews and discussions, block valve testing is adequate to identify block valve inoperability and that the block valves can be reasonably expected to satisy their design specifications should a PORV fail open. The licensee is using a torque setting which yields valve closing times of about 30 seconds. The inspector will monitor the licensee's establishment and implementation of a comprehensive preventive maintenance program as part of the LER followup. This review closes out the concerns of unresolved item 81-33-01 and open item 81-32-01.

14. Licensee Event Report (LER) Followup

The inspector reviewed the following LER's to verify that the report details met license requirements, identified the cause of the event, described appropriate corrective actions, adequately assessed the event, and addressed any generic implications. Corrective action and appropriate licensee review of the events listed below was verified. The inspector had no further comments.

LER	EVENT
82-02	HVH-1 Trip
82-06	HVH-2 Motor Cooler Leak

15. Review of IE Notices (IEN's)

The inspector verified that IEN's had been received onsite and reviewed by cognizant licensee personnel. Selected applicable Notices were discussed with licensee personnel to ascertain the licensees actions on these items. The inspector also verified that Notices were reviewed by the Plant Nuclear Safety Committee in accordance with facility administrative policy. Licensee action on the following IE Notices were reviewed by the inspector and are closed.

IE Notices

82-03
82-13

16. Corrective Action Review

a. Based on operational problems experienced at CP&L's Brunswick facility, the corporate staff formed an oversight review committee to review Robinson's quality of operations and pre-startup activities. This review encompassed the areas of surveillance activities, procedural adherence, procedural adequacy, personnel training, plant modifications, and regulatory and internal commitments. Additionally, independent assessments of operational activities and communications were conducted by Corporate Nuclear Safety Department, Corporate Quality Assurance Department, and the Institute For Nuclear Power Operations. The inspector attended the review committee meeting conducted on August 6, 1982. The licensee identified no safety concerns precluding startup but has established several programs to correct identified deficiencies. The inspector will continue to monitor the progress of licensee corrective actions.

b. The inspector reviewed various problem reports generated over the preceding two to three months in order to identify trends or recurring failures and to determine the adequacy of licensee corrective action with respect to safety-related equipment. Licensee efforts appeared adequate, and maintenance activities in general appear satisfactory to maintain equipment operability. No violation or deviations were noted.

17. TMI Action Item I.A.1.3.2a, NUREG 0737, Shift Manning

This item required the inspector to verify that either the licensee's Technical Specification or Administrative Instructions (AI) incorporated the shift manning requirements. The inspector reviewed Technical Specification section 6.2, AI Sections 2 and 4, and CP&L letters dated December 15, 1980, February 10, 1982, and June 9, 1982. The licensee's four Senior Reactor

Operator (SRO) candidates were successfully licensed, therefore July 1, 1982 is the commitment date. The inspector determined that Technical Specification 6.2.2.e and AI 4.1.1 satisfied all but the following requirements:

a. That one licensed SRO be in the control room at all times above cold shutdown conditions.

b. That when the plant is shutdown, one Shift Foreman, one reactor operator, and one auxiliary operator compromise the shift complement.

The licensee approved a revision to AI 4 on August 11, 1982 which incorporates the above requirements. This item is closed.

18. Outstanding Item Review

(Closed) IFI 82-02-05. This item concerned the need to revise General Procedure (GP)-6 to provide a signoff blank for actual step completion. The inspector reviewed revision 16 to GP-6 which provides an explanatory note and signoff at step 4.18 and a signoff at step 4.24.8. This appears acceptable.

(Closed) IFI 82-20-07. This item concerned the need to clarify Emergency Instruction (EI)-1. The inspector reviewed revision 32 to EI-1. This revision clarifies, reorganizes, and adds information to improve operator response and understanding and appears acceptable. The licensee's Corporate Nuclear Safety Department is continuing to review this procedure.

(Closed) Open Item 82-11-03. This item concerned the need for a surveillance program on the Dedicated Shutdown System (DSDS). The inspector reviewed Periodic Tests 48.2 and 48.3 and Maintenance Instruction-4, Appendix A, pages 118-119. The inspector is satisfied that an adequate surveillance program exists to ensure operability of the DSDS.

(Closed) Open Item 81-32-01. As discussed in paragraph 13, the questions concerning block valve adequacy and usage have been resolved.

(Closed) IFI 82-20-01. This item concerned the SI 870 A and B position indication discrepancy. This has been corrected.

(Closed) IFI 82-04-16. This item concerned the licensee's need to upgrade several fire barrier seals. This upgrade was accomplished during the refueling outage via work requests FP 82-85 and 82-86 and Modification-646.

(Closed) IFI 82-04-17. This item concerned the need to restore fire damper operability. The fire dampers in question were modified under Modification-650 by adding springs and weights and were satisfactorily inspected and tested after modification.

(Closed) Open Item 82-07-06. The licensee investigated this item and determined that the disconnected line should in fact be connected. The line was reconnected and the system restored to original design. The licensee determined that the line was probably disconnected about the time of original plant startup during installation of effluent accountability equipment. Apparently, modification procedures were not developed or implemented to adequately control the work. The licensee has taken corrective action to control future modifications.

(Open) IFI 82-20-08. This item concerned the licensee's commitments and actions in response to pressurized thermal shock (PTS) concerns. Emergency Instruction (EI)-1, Revision 32 clarified the various accident safety injection termination criteria. The revision was reviewed and appeared adequate. This closes part a., of this item. Additionally, a licensee walkthrough of EI-1 was conducted on July 9, 1982 using the Robinson control board. No problems were identified and the walkthrough commitment was satisfied. The licensee is still reviewing the feedback from the attendees of the PTS training to determine any necessary revisions or classifications. This item will remain open until items c. and d. are completed.

(Closed) IFI 81-25-08, Amendment 70 to CP&L license DPR-23 corrected the discrepancy between Technical Specification Section 6.2 text and Figure 6.2-2.

(Closed) Open Item 81-27-01. Revision 3 to the Corporate Quality Assurance Manual adds the required responsibilities and authorities descriptions.

(Closed) Open Item 81-26-01. The licensee has written new Periodic Test 19.2 for zero power testing of reactor protection and safeguards train operability. This procedure provides necessary guidance on dummy signals to compensate for existing plant conditions when in cold shutdown. This procedure appears adequate to answer this concern.