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

Report No.: 50-261/87-30

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

Inspection Conducted: September 11 - October 10, 1987

Inspectors: 10/27/87
For H. E. P. Krug, Senior Resident Inspector Date Signed

10/27/87
For R. M. Latta, Resident Inspector Date Signed

Approved by: 10/27/87
P. E. Fredrickson, Section Chief Date Signed
Division of Reactor Proejects

SUMMARY

Scope: This routine, announced inspection was conducted in the areas of Technical Specification (TS) compliance; including observance of any Limiting Conditions for Operation (LCO), plant tour, operations performance, reportable occurrences, housekeeping, site security, surveillance activities, maintenance activities, quality assurance practices, radiation control activities, outstanding items review, IE Bulletin and IE Notice followup, organization and administration, independent inspection, Emergency Preparedness, Plant Status Report, Systematic Assessment of Licensee Performance (SALP) and enforcement action followup.

Results: No violations or deviations were identified within the areas inspected.

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REPORT DETAILS

1. Licensee Employees Contacted

R. Barnett, Maintenance Supervisor, Electrical
G. Beatty, Vice President, Robinson Nuclear Project Department
R. Chambers, Engineering Supervisor, Performance
D. Crocker, Supervisor, Radiation Control
J. Curley, Director, Regulatory Compliance
J. Eaddy, Supervisor, Environmental and Chemistry
R. Femal, Shift Foreman, Operations
W. Flanagan, Manager, Design Engineering
W. Gainey, Support Supervisor, Operations
P. Harding, Project Specialist, Radiation Control
E. Harris, Director, Onsite Nuclear Safety
D. Knight, Shift Foreman, Operations
E. Lee, Shift Foreman, Operations
F. Lowery, Manager, Operations
D. McCaskill, Shift Foreman, Operations
A. McCauley, Principal Specialist, Onsite Nuclear Safety
R. Miller, Maintenance Supervisor, Mechanical
R. Moore, Shift Foreman, Operations
R. Morgan, Plant General Manager
M. Morrow, Specialist, Emergency Preparedness
D. Myers, Shift Foreman, Operations
D. Nelson, Operating Supervisor
B. Murphy, Senior Instrumentation and Control Engineer
M. Page, Engineering Supervisor, Plant Systems
R. Powell, Principal Specialist, Maintenance
D. Quick, Manager, Maintenance
B. Rieck, Manager, Control and Administration
D. Sayre, Senior Specialist, Regulatory Compliance
D. Seagle, Shift Foreman, Operations
R. Smith, Manager, Environmental and Radiation Control
R. Steele, Shift Foreman, Operations
R. Wallace, Manager, Technical Support
L. Williams, Supervisor, Security
H. Young, Director, Quality Assurance/Quality Control (QA/QC)

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

2. Exit Interview (30702, 30703)

The inspection scope and findings were summarized on October 9, 1987, with the Plant General Manager, and the Acting Director of Regulatory Compliance, and the Acting Director QA/QC. The licensee acknowledged the findings without exception. The licensee did not identify as proprietary

any of the materials provided to or reviewed by the inspectors during this inspection. No written material was given to the licensee by the Resident Inspectors during this report period.

3. Plant Tour (71707, 62703, 71710)

The inspectors 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 maintenance activities, and plant housekeeping efforts were adequate. The inspectors 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 inspectors 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 fire fighting equipment, and instrument calibration dates. Some tours were conducted on backshifts. Plant housekeeping and contamination control were observed to be excellent.

The inspectors performed system status checks on the following systems:

a. Safety Injection (SI) System
b. Component Cooling Water (CCW) System
c. Auxiliary Feedwater (AF) System
d. Vital Station Batteries (VSB)
e. Electrical Switchgear
f. Chemical and Volume Control System (CVCS)
g. Residual Heat Removal (RHR) System
h. Containment Spray System
i. Emergency Diesel Generators (EDG)

No violations or deviations were identified within the areas inspected.

4. Technical Specification Compliance (71707, 62703, 61726)

During this reporting interval, the inspectors verified compliance with selected limiting conditions for operation and reviewed results of certain surveillance and maintenance activities. These verifications were accomplished by direct observation of monitoring instrumentation, valve positions, switch positions, and review of completed logs and records.

In addition, on a daily basis in the control room, the inspectors independently examined Emergency Response Facility Information System (ERFIS) and Safety Parameter Display Systems (SPDS) display of safety related parameters, including component status information, for indications related to conformance with the TS.

No violations or deviations were identified within the areas inspected.

5. Plant Operations Review (71707, 62703, 61726, 61707)

Periodically during the inspection interval, the inspectors reviewed shift logs and operations records, including data sheets, instrument traces, and records of equipment malfunctions. This review included control room logs, maintenance work requests, auxiliary logs, operating orders, standing orders, jumper logs, and equipment tagout records. The inspectors routinely observed operator alertness and demeanor during shift changes and plant tours. The inspectors conducted random off-hours inspections during the reporting interval to assure that operations and security were maintained in accordance with plant procedures.

The inspectors periodically verified the reactor shutdown margin. The inspectors also periodically observed the reactor axial flux difference and compared the observed values with those required by the TS.

While the inspectors were in the control room at 6:10 a.m., on September 20, 1987, they witnessed the response of the fire brigade to an unannounced fire drill. The simulated fire was located in the "A" Emergency Diesel Generator room, in the auxiliary building. The inspectors witnessed the response of the fire brigade team leader (senior reactor operator on duty) as well as the actions of other plant personnel who were utilizing Fire Protection Procedure FP-001 (Revision 10) titled "Fire Emergency" and OMM-002 (Revision 9) titled "Fire Protection Manual".

The inspectors observed the control room verification of the operation of the motor driven fire pump as well as the donning of full turn-out gear and self-contained breathing apparatus by all fire brigade members, and the access control of the radiologically controlled area by security personnel. The inspectors noted that the fire brigade personnel responded quickly to the alarm and exhibited good fire fighting techniques throughout the drill.

No violations or deviations were identified within the areas inspected.

6. Physical Protection (71707)

In the course of the monthly activities, the inspectors included a review of the licensee's physical security program. The inspectors verified by general observation, perimeter walkdowns and interviews that measures taken to assure the physical protection of the facility met current requirements. The inspectors routinely observed the alertness and demeanor of security force personnel during plant tours. The inspectors also visited the central and secondary alarm stations.

The performance of various shifts of the security force was observed in the conduct of daily activities to include: protected and vital areas access controls; searching of personnel, packages and vehicles; badge issuance and retrieval; escorting of visitors; and patrols and compensatory posts. In addition, the inspectors observed protected area lighting,

protected and vital areas barrier integrity and verified an interface between the security organization and operations or maintenance.

No violations or deviations were identified within the areas inspected.

7. Monthly Surveillance Observation (61726, 61700, 71710)

The inspectors observed certain surveillance related activities of safety-related systems and components to ascertain that these activities were conducted in accordance with license requirements. The inspectors observed portions of selected surveillance tests including all aspects of one major surveillance test involving safety-related systems. The inspectors determined that the surveillance test procedure conformed to TS requirements, that all precautions and LCO's were met and that the surveillance test was completed at the required frequency. The inspectors also verified that the required administrative approvals and tagouts were obtained prior to initiating the test, that the testing was accomplished by qualified personnel in accordance with an approved test procedure and that the required test instrumentation was properly calibrated. Upon completion of the testing, the inspectors observed that the recorded test data was accurate, complete and met TS requirements; verified that test discrepancies were properly rectified, and; independently verified selected test results and proper return to service.

The inspectors verified that the calculation of core thermal power was properly performed in accordance with Revision 5 to Operations Surveillance Procedure OST-010 titled "Power Range Calorimetric During Power Operation Daily;" and that the power level instruments indicated that the reactor power was within prescribed limits. The inspectors also reviewed the current revision of OST-010 for technical adequacy and reviewed the results for a specific evaluation at 100% power. The licensee performed the evaluations daily, which is within the frequency prescribed by the TS.

The inspectors also witnessed/reviewed portions of the following test activities:

OST-051 Reactor Coolant System Leakage Evaluation, Revision 7

OST-905 Radiation Monitoring System (Daily), Revision 5

FMP-007 Quandrant Power Tilt, Revision 4

No violations or deviations were identified within the areas inspected.

8. Monthly Maintenance Observation and Maintenance Program Evaluation (62703, 62704, 62705)

The inspectors observed several maintenance related activities of safety-related systems and components to ascertain that these activities were conducted in accordance with approved procedures, TS and appropriate

industry codes and standards. The inspectors determined that these activities were not violating LCO's and that redundant components were operable. The inspectors also determined (1) that the procedures used were adequate to control the activity, (2) that QC hold points were established where required, (3) that required administrative approvals and tagouts were obtained prior to work initiation, (4) that proper radiological, and appropriate ignition and fire prevention controls were implemented, and (5) that replacement parts and materials used were properly certified. The inspectors verified that these activities were accomplished by qualified personnel using approved procedures. The inspectors verified that selected equipment was properly tested before being returned to service.

Additionally, the inspectors reviewed several outstanding job orders to determine that the licensee was giving priority to safety-related maintenance and that a backlog which might affect its performance was not developing on a given system. The inspectors observed/reviewed the following maintenance activities:

MST-003	Tavg and Delta-T Protection Channel Testing (Biweekly), Revision 10
MST-016	Containment Pressure Protection Channel (Set I, II, and III) Testing, Revision 4
MST-902	Emergency Battery Test, Revision 8
WR/JO 87-AECL1	Repair of V2-14A Valve Actuator, Auxiliary Feedwater System

No violations or deviations were identified within the areas inspected.

9. Operational Safety Verification (71707, 82301)

The inspectors observed licensee activities to ascertain that the facility was being operated safely and in conformance with regulatory requirements, and that the licensee management control system was effectively discharging its responsibilities for continued safe operation by direct observation of activities, tours of the facility, interviews and discussions with licensee management and personnel, independent verification of safety system status and limiting conditions for operation, and reviewing facility records.

On September 23, 1987, the inspectors witnessed the execution of a dress rehearsal for the emergency preparedness drill scheduled for October 6, 1987. The purpose of this exercise was to demonstrate the licensee's capability to respond adequately to a simulated accident involving a radiological emergency. The drill scenario was based on an earthquake, a failure of the reactor to trip on demand, fuel damage, a loss of coolant accident, and a subsequent release of radioactivity to the environment from containment exhaust purge valves which were stuck open.

The emergency exercise dress rehearsal included participation by corporate, state, and local emergency response organizations as well as site personnel in the control room, technical support center (TSC), and operations support center (OSC). The licensee assigned an evaluation team to overview and critique the exercise and to assure that the intended objectives were achieved. The stated objectives included successful demonstration of the following activities:

- The ability of the control room staff to recognize operational symptoms that are indicative of degrading plant conditions and to respond to deteriorating plant parameters.

- The ability of onsite security forces to search the TSC and place it into the protected area within the required time frame.

- To exhibit the proper and timely response of emergency personnel when activating emergency response facilities as well as to properly carry out the assigned roles and responsibilities.

- The initial activation and functional adequacy of the emergency response facilities as well as the precise and clear transfer of responsibilities and information from the control room to the TSC staff and subsequently to the EOF.

- The adequacy and effectiveness of communications methods used to inform onsite and offsite personnel of emergency situations and plant conditions as well as initial and follow-up emergency information to state and local authorities.

- The ability to respond to, and treat a contaminated medical emergency victim using the proper survey techniques and first aid equipment.

- The ability to formulate offsite radiological dose projections as well as to account for all onsite personnel.

The inspectors were notified of the initiating event by the control room emergency communicator at 8:22 a.m. on the morning of the dress rehearsal. The inspectors proceeded to the TSC where they witnessed portions of the search of that facility by members of the onsite security force. The search was thorough and expeditiously conducted. The inspectors also witnessed the manning of the TSC by emergency response personnel including the establishment of emergency communications and the assessment of simulated plant off-normal conditions. The inspectors continued to monitor the progression of the scenario through the declaration of a general emergency, the activation of the EOF and the simulated recovery efforts initiated by the licensee. These activities were observed to be well executed and controlled.

On October 6, 1987, the inspectors observed portions of an emergency preparedness exercise conducted by the licensee. The exercise commenced at 5:30 a.m., and was concluded at approximately 2:00 p.m. The accident

scenario was based on a small loss of coolant accident with elevated reactor coolant activity caused by loose parts in the core. A reactor trip combined with a loss of steam dump and loose parts in the reactor core provide a transient which produced increased reactor coolant activity. During the cool down cycle the Steam Generator Power Operated Relief Valves (PORVs) cycled open and shut continuously because the steam dump system was unavailable due to repairs. Eventually the PORV on "C" Steam Generator failed full open, thereby providing a potential release path to the environment. Shortly thereafter, a Steam Generator tube leak of approximately 600 gpm occurred in "C" Steam Generator, caused by a loose part on the secondary side.

This emergency exercise included participation by licensee personnel in the control room, technical support center, operational support center, emergency operations facility, corporate emergency operations center, and plant media center.

The licensee assigned an evaluation team to overview and critique the drill and to assure that the intended objectives were achieved. The stated objectives included successful demonstration of the following activities:

- Accident Assessment and Emergency Classification
- Notification, Mobilization, and Communication
- Dose Calculation and Radiological Assessment
- Protective Response
- Emergency Response Facility Operation
- Public Information Services
- Recovery Operations

The inspectors witnessed the initiating event response by operations personnel in the control room and the subsequent progression of events which led to assembly of the emergency response staff. The inspectors also observed the establishment of emergency communications; the response to a simulated medical emergency involving a contaminated victim; the assessment of simulated plant off-normal conditions; the activation of the technical support center and the emergency operations center; and the simulated recovery efforts initiated by the licensee.

No violations or deviations were identified within the areas inspected.

10. ESF System Walkdown and Monthly Surveillance Observation (71710, 61726, 56700)

The inspectors verified the operability of certain engineered safety features systems by performing a walkdown of the accessible portions of the safety injection, residual heat removal and containment spray systems, including the associated valves inside the containment. This walkdown was conducted in accordance with the licensee's Operations Surveillance Test

Procedure OST-158 (Revision 4) titled "Safety Injection and Containment Spray Systems Flowpath Verification Monthly Interval (At Power)". The inspectors ascertained that the current revision of the subject procedure was used by qualified operations personnel and that the monthly performance of this surveillance test satisfied the requirements of Section 4.5.2.2 of the TS.

The inspectors noted that the latest revision of the subject procedure was modified to include the positional verification of valve RHR-764. As previously documented in Inspection Reports 50-261/87-15 and 87-23, this manual valve was incorrectly left in the closed position by operations personnel during valve line up activities. This action effectively blocked the low pressure injection path from the discharge of the Residual Heat Removal (RHR) pumps to the cold leg injection points. Inclusion of RHR-764 into OST-158 means that the position of this valve will be verified every month - at power.

While in the RHR Heat Exchange Room and with the reactor at power, the inspectors determined that valve RHR-764 was locked open as now required by OST-158. The inspectors also noted throughout the conduct of this surveillance test that the auxiliary operator involved utilized a working copy of the procedure instead of hand written notes, that he demonstrated a strong working knowledge of the systems involved, and that his execution of the procedure was well disciplined.

The inspectors determined that the prerequisites for the performance of this surveillance test were performed and that all specified precautions and limitations were observed, including the wearing of full anti-contamination clothing in posted high radiation areas. The inspectors looked for equipment conditions, maintenance status and items that might degrade performance (that hangers and supports were operable, acceptable housekeeping, etc.). The inspectors verified that valves were in proper position, power was available, and valves were locked as appropriate. The inspectors compared both local and remote position indications. The inspectors notified the licensee that housekeeping in the Residual Heat Removal Pump Pit was below the conditions maintained throughout the rest of the plant, and that the "B" RHR pump exhibited excessive gland seal leakage. The inspectors were informed that this condition had been recently documented on a work request form and that it was a planned maintenance activity. The inspectors subsequently verified that this item was on the licensee's automated maintenance management system and that corrective maintenance was scheduled.

No violations or deviations were identified within the areas inspected.

11. Cold Weather Preparations (71714)

The inspectors conducted a review of licensee cold weather preparations to ascertain that the licensee maintained effective implementation of the program of protective measures for extreme cold weather. During the

inspection, the inspectors verified that the licensee had inspected systems susceptible to freezing to verify the presence of heat tracing, strip heaters and insulation; the proper setting of thermostats; and that the heat tracing and strip heating circuits were energized. The inspectors also determined that, for systems which had been subjected to maintenance and/or modification during the past year, that any required protective measures were reestablished, and during periods of prolonged shut down, that areas that are no longer kept warm by normal plant operations are adequately protected.

The inspectors witnessed the operability verification performed by instrumentation and control technicians of selected freeze protection panels throughout the plant. The inspectors observed channel current testing in the subject panels and, for those freeze protection channels which indicated a faulted condition, the inspectors determined that corrective maintenance work requests were initiated. Subsequent to the completion of the cold weather preventative maintenance testing, the inspectors independently witnessed the return to service of the affected equipment.

No violations or deviations were identified within the areas inspected.

12. Onsite Followup of Events and Subsequent Written Reports of Nonroutine Events at Power Reactor Facilities (92700, 90714, 93702)

a. Hagan Rack Supports

On September 4, 1987, Unit No. 2 was operating at one hundred percent power. During a walkdown of electrical equipment by licensee personnel, for unrelated reasons, the Reactor Protection and Control Analog Instrumentation Racks (Hagan Racks) anchorage to maintain Seismic Class I requirements was questioned. The Hagan Racks consist of Analog Protection and Logic Channels which monitor parameters for nuclear and on-nuclear instrumentation and actuate signals for Reactor Protection and Safeguards initiation. The Hagan Racks are located in a room behind the Unit 2 Control Room.

On September 11, 1987, Unit 2 was placed in hot shutdown to replace the main generator hydrogen coolers. Preliminary calculations indicated that the Hagan Rack anchorage was insufficient and that additional seismic supports and braces may be needed for the racks to meet their Seismic Class I requirements to assure continued operation of instrumentation within the cabinets during a Design Basis Earthquake.

On September 13, 1987, as a precautionary measure while more detailed analyses were performed, a Plant modification was implemented to upgrade the Hagan Rack anchorage. This modification was completed on September 15, 1987.

On September 18, 1987, the results of an interim analyses, supported by information provided by Westinghouse concerning the original testing and seismic qualification of the Hagan Racks, also indicated that the Hagan Rack anchorage may be inadequate. The existing anchorage was not the same as the anchorage used in the generic testing which provided the original seismic qualification of the racks. The licensee stated that the information available from the construction time period (1968 - 1970) was inadequate to verify the required installation procedure for the Hagan Racks.

As a result, also on September 18, 1987, the licensee reported the status of its evaluations to the NRC in accordance with 10 CFR 50.72-(b)(2)(i). The licensee also notified the Senior Resident Inspector.

Concurrent with analysis of Hagan Rack anchorage seismic qualification, a design solution was pursued as a precautionary measure to assure the Class I requirements of the Hagan Racks. An investigation of the seismic qualification of equipment associated with the operation of the Hagan Racks has been performed. This investigation included the Reactor Protection and the Safeguards Relay Racks, since these racks receive signals from the Hagan Racks and actuate Reactor Protection and Engineered Safety Feature initiation circuitry. The anchorage of this equipment was found satisfactory by the licensee to assure operability during a Design Basis Earthquake. On September 12, 1987, this design solution was incorporated into a Plant Modification. The seismic support work for this modification, which added supports and braces to the Hagan Rack Cabinets was completed on September 15, 1987. This modification increased the functional and structural integrity of the Hagan Racks over the minimum safety margin required by a factor of two.

The licensee is a member of the Seismic Qualification Utility Owner Group (SQUOG) which is working with the NRC to disposition Unresolved Safety Issue (USI) A-46, "Seismic Qualification of Equipment in Operating Plants." The SQUOG Group is in the process of developing methodology for the evaluation of the seismic qualification of equipment at older plants using a new data base which includes the measured responses of installed equipment during actual seismic exitation. A preliminary assessment by the licensee indicates that the original mountings of the Hagan Racks may be qualified using the evolving SQUOG methodology. The licensee has included the issue of the Hagan Rack supports into its program for the resolution of SQUOG issues. The inspectors will monitor this program during future inspections.

b. Emergency Diesel Generators (EDG) (Closed) LER 87-23

Paragraph 12 of Inspection Report 50-261/87-28 described a problem with the "A" EDG in that the "A" EDG was found to trip on overspeed. As previously noted, this problem was attributed to a sticking

plunger assembly in the number four injection pump, and was reported by the licensee on August 26, 1987, in accordance with 10 CFR 50.72. The number four injector pump was removed and replaced with a pump from plant inventory. None of the other 23 fuel injection pumps on "A" EDG nor any of the 24 injection pumps on the "B" EDG were found, by the licensee, to have sticking plungers. Thus the problem with overspeed trips on the "A" EDG was considered to be solely due to the sticking plunger on the number four injector pump. Unit 2 remained at 100% power throughout the event. The "A" EDG was tested in accordance with OST-401 on September 1 and 7, 1987 and performed satisfactorily. However, subsequent events disclosed that the problem was more complex.

On September 8, 1987, with Unit 2 operating at 100 percent power and "B" EDG out of service for scheduled maintenance since 5:45 a.m. under TS 3.7.2.d, the "A" EDG automatically tripped on overspeed upon starting at 10:36 p.m. during the required operability testing under OST-401. A second local start also resulted in an automatic overspeed trip, and the "A" EDG was declared inoperable pending resolution of the overspeed trip condition. The "B" EDG was returned to service at 4:06 a.m. the following morning, September 9, 1987, approximately four hours and thirty minutes following the initial overspeed trip of the "A" EDG on September 8. The engineered safety features associated with the "A" EDG were operable throughout the incident.

Licensee personnel found two fuel injection pumps on the "A "EDG which appeared to be sticking open; one on the number 5 cylinder and one of the number 9 cylinder. Both were replaced with pumps from the plant inventory. Unit 2 remained at 100 percent power throughout the event.

On September 9, 1987, the Senior Resident Inspector was notified of the significant event described above which was reported by the licensee in accordance with 10 CFR 50.72.

The TS allowed the "A" EDG to remain inoperable for up to seven days if the "B" EDG was tested daily to ensure operability, and the engineered safety features associated with the "B" EDG were operable. On Thursday, September 10, 1987, however, Unit 2 was taken into a forced outage to repair a leak on the main generator hydrogen coolers, and the plant was taken off-line at 8:15 p.m.

During the forced outage, troubleshooting into the cause of the overspeed trips of the "A" EDG was performed by licensee maintenance personnel as well as by the diesel manufacturer's technical representative, a diesel engineering consultant, and the diesel governor manufacturer's technical representative, under contract to the licensee. The evaluations considered the sticking fuel injection

pumps and determined this condition aggravated the situation but was only a contributing factor in causing the engine to reach the overspeed trip point.

The licensee's detailed evaluation disclosed that the overspeed trips on the "A" EDG were primarily the result of two contributing factors. The first was the resetting of the load limiting set points on both EDG's. Prior to the series of overspeed trips on the "A" EDG, the licensee executed its procedure for the setting of the load limiting set point on the "B" EDG to ensure that the engine could develop the required power to allow the generator to maintain 2750 kw - 110 percent of the continuous rating. The load limiting set point for the "B" EDG was thus found to be 9.0.

Licensee personnel recognized that following the same calibration process on the "A" EDG would result in a lower load limiting set point for the "A" EDG, but elected to set the "A" the same as the "B" EDG for human factors considerations - since the same load limiting setting on the "A "EDG, while reasonable, was conservative with respect to developing the required 2750 kw. Subsequent initial testing of the "A" EDG with the load limiting set point of 9.0 disclosed no overspeed problems. However, the higher the load limiting set point, the higher the expected maximum engine rpm on start, before the governor begins to narrowly control the engine rpm. Also, the maximum starting engine rpm will vary in a distribution about the mean.

Similarly, the licensee determined that the spring loading on the centrifugal weights on the mechanical overspeed trip device related with age, as evidenced by a relatively wide distribution in trip rpm coupled with a mean trip valve which drifted downward with time. Also, the licensee found that the shim washers used to adjust the trip point were burred along the edges as a result of original fabrication, and thus could have caused the large variation in trip point rpm variation. Thus, whether or not an overspeed trip would occur on the "A" EDG depended upon two statistical variations - that associated with the rpm overshoot above 900 rpm on startup, and that associated with the mechanical overspeed set point.

With respect to the mechanical overspeed set point, the licensee installed a new spring to load the centrifugal weights and removed the burrs on the spring adjusting shims. Additional shims were also added to bring the mean overspeed set point up to approximately 1020 rpm - the vendor optimum setting.

The licensee also reset the load limiting set point (LLSP) on the "A" EDG from 9.0 - the optimum setting for the "B" EDG, to 8.1 - the optimum setting for the "A" EDG. Thus, the "A" EDG overshoot rpm was reduced.

The combination of the two readjustments provided the appropriate separation between the startup rpm overshoot rpm and the lowest trip point on the mechanical overspeed trip, as was confirmed by subsequent testing of the "A" EDG.

c. Reactor Trip

On September 28, 1987, at 2:45 p.m., Unit No. 2 experienced a reactor trip from one hundred percent power when an Instrumentation and Control (I&C) technician performing a surveillance test on the "B" train reactor trip breaker (RTB) inadvertently opened the "A" train RTB with the "B" train RTB still in test. This satisfied the one-out-of-two logic and the reactor tripped with the first out annunciator being Over-Power Delta Temperature and Turbine Trip. All safety systems operated as designed, including auxiliary feed water, which was secured by the control room operators when steam generator levels reached normal levels.

At the time of the reactor trip, plant operations and I&C personnel were performing a monthly Maintenance Surveillance Test MST-010, (Revision 16) titled "Reactor Protection Logic Train "A" and "B" at Power, Safeguard Relay Rack Train "A" and "B", in accordance with TS Table 4.1-1, Item No 27. During the execution of the final portions of MST-010, step 7.6.8, which returns the "B" train RTB to service from the by-pass position, the I&C technician responsible for inserting the key into the "B" train RTB test automatic shunt trip test switch inadvertently inserted the key into the "A" train RTB test automatic shunt trip test switch and turned it to the test position. This action resulted in the "A" train RTB opening. With the "A" bypass breaker already open, the opening of the "A" RTB satisfied the reactor protection system logic for a train "A" reactor trip signal with a resultant reactor trip/turbine trip.

The inspectors determined that the subject test automatic shunt trip test switches for both the "A" and "B" train are physically located next to each other in the same electrical cabinet and that they are both operated with the same key. As stated by the licensee, this colocation feature of the test switches along with the protracted duration of the MST, were contributing factors to the personnel error which resulted in the reactor trip. The licensee is currently reviewing potential corrective measures which include human factors considerations such as separate (unique) keys for each train, enhanced labeling, color coating to distinguish the two switches, and dividing the current MST, which require 6 to 7 hours to complete, into two separate surveillance test procedures.

The inspectors determined that the last reactor trip associated with a personnel error while conducting surveillance testing took place on September 10, 1986. The attendant circumstances surrounding this event indicate that this is an isolated case of cognitive personnel

error aggravated by equipment and procedural inadequacies. The inspectors will continue to monitor the licensee's corrective measures during subsequent inspections.

A minor personnel injury was sustained during a incident related to the above described reactor trip when hot water from relief valves on the condensate system overflowed a drain funnel and splashed a nearby maintenance worker. Investigation by the inspectors disclosed that the relief valves on the condensate system recovery heat exchanger emptied into a common funnel drain near the area where the injured worker was installing isolation lagging. The individual received minor first and second degree burns around the neck and shoulders and returned to work the same day following a hospital examination.

For onsite followup of nonroutine events, the inspectors determined that the licensee had taken corrective actions as stated in written reports of the events and that these responses to the events were appropriate and met regulatory requirements, license conditions, and commitments. During this reporting period, the inspectors reviewed the events described above 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. When the LER has been issued during the report period, the LER number is identified in the appropriate paragraph.

No violations or deviations were identified within the areas inspected.

13. Onsite Review Committee (40700, 36700)

The inspectors reviewed the on-site licensee organization to ascertain whether changes made to the licensee's onsite organization are in conformance with the requirements of the TS by verifying that (1) the established organization is as described in the TS and is functioning effectively, (2) personnel qualification levels are in conformance with applicable codes and standards, and (3) the lines of authority and responsibility are in conformance with TS and applicable codes and standards. The inspectors also reviewed appropriate licensee records to ascertain whether the licensee's use of overtime is in conformance with regulatory requirements and that any deviations from maximum overtime limits were authorized in accordance with TS and/or plant administrative procedures.

Comprehensive discussions of current safety-related activities were conducted with plant management and technical personnel during this reporting period including, and in particular, Operations, Environmental and Radiation Controls, Quality Assurance, Regulatory Compliance and Onsite Nuclear Safety organizations. Topics discussed included licensee

activities associated with emergency preparedness, plant operations activities; plant modifications, including the security system upgrade; the fire protection system; ongoing construction activities; and communications interfaces.

The inspectors reviewed certain activities of the plant nuclear safety committee (PNSC) to ascertain whether the onsite review functions were conducted in accordance with TS and other regulatory requirements. The inspectors (1) attended the special PNSC meeting held on September 24, 1987, concerning the "A" diesel generator and observed the conduct of the meeting, (2) ascertained that provisions of the TS dealing with membership, review process, frequency, qualifications, etc., were satisfied, and (3) reviewed meeting minutes to confirm that decisions and recommendations were accurately reflected in the minutes, and (4) followed up on previously identified PNSC activities to independently confirm that corrective actions were progressing satisfactorily.

No violations or deviations were identified within the areas inspected.

14. Low Temperature Over Pressure Protection (TI 2500/19)

As delineated in Unresolved Safety Issued A-26, an item was identified concerning the safety margin to failure for Pressurized Water Reactors should they be subject to severe pressure transients while at a relatively low temperature. Primarily, the subject transients can occur during startup or shutdown operations when the reactor coolant system (RCS) is in a water solid condition. During such conditions, the RCS is susceptible to a rapid increase in system pressure through thermal expansion of the RCS water or through injection of water into the system without adequate relief capacity or discharge flow path to control the pressure increase.

In order to verify that the licensee implemented their commitments as described in their safety evaluation report concerning overpressure transient mitigation, the inspectors reviewed the following documents:

(1) Westinghouse Owners Group, Pressure Mitigation Systems Transient Analysis Results, July 1977

(2) H. B. Robinson Plant Modification No. 418, Plant Overpressure Protection System

(3) Reactor Coolant System - Flow Diagram, 5379-1971, Revision 25, Sheet 2 of 2

(4) Instrument and Station Air System - Flow Diagram, G-190200, Revision 3, Sheet 9 of 10

(5) Safety Evaluation Performed by NRR Supporting Amendment No. 42 to Facility Operating Licensee No. DPR-23, CP&L, H. B. Robinson Steam Electric Plant, Unit No. 2, Docket No. 50-261, dated September 14, 1979

(6) Code of Federal Regulations, 10 CFR 50, Appendix G, Fracture Toughness Requirements

(7) H. B. Robinson Engineering Evaluation No. 86-117 with Attachments 3 and 4

(8) General Procedure GP-002, Revision 24, Cold Solid to Hot Subcritical at No Load T-Avg

(9) General Procedure GP-007, Revision 10, Plant Cooldown From Hot Shutdown to Cold Solid

(10) Annunciator Panel Procedure, APP-003, Revision 7, Reactor Coolant and Makeup Systems

(11) General Procedure GP-008, Revision 11, Draining the Reactor Coolant System

(12) Modification and Design Control Procedure, MOD-013, Revision 0, Safety Review

(13) Maintenance Surveillance Test Procedure, MST-007, Revision 7, Reactor Coolant Low-Temperature Overpressure Protection System Test

(14) Operating Procedure, OP-006, Revision 5, Low Temperature Overpressure Protection System

(15) Operations Surveillance Test Procedure, OST-703, Revision 10, ISI Primary Side Valve Test

Utilizing the reference Westinghouse Owners Group Study, the licensee developed a plant modification (Document 2) which provided the design basis for the Low Temperature Over Pressure (LTOP) system at H. B. Robinson. This system, as depicted in Documents (4) and (5), utilizes the existing power operated relief valves (PORVs) located on the pressurizer. These valves may be operated remotely from the control room or automatically activated by redundant channels of protective instrumentation which compare actual pressure and temperature with Document (6) above requirements. The PORVs are spring loaded, normally closed valves which require air to open. Air is supplied from the plant instrument air system and is backed up by two redundant and independent nitrogen gas systems.

a. Design

The inspectors determined through a review of the pertinent drawings (Documents 3 and 4) that the LTOP system was designed with redundant actuation controls which protect the pressure vessel given a single failure, in addition to the failure that initiated the pressure transient. These features are incorporated in redundant and

independent control and indication systems which include two appropriately sized PORV's which conform to the opening time requirements of 10 CFR 50 Appendix G, two DC powered solenoid valves per PORV supplied by uninterruptable power. Two independent nitrogen (N2) gas systems are provided, one to each PORV, which serve as a backup for the normal motive source, which is instrument air. Additionally, the system contains two independent pressure controllers and two accumulators, one for each of the N2 supply lines. As determined by the inspectors, these two accumulators are physically separated to preclude damage from missiles generated during accident conditions. The inspectors also ascertained that control room indication is provided on the reactor turbine generator board (RTGB) in the form of independent key lock switches for each PORV, position indicators for each valve, a dedicated annunciator, and independent temperature indicators for each PORV (located on the down stream piping).

In considering the design adequacy of the LTOP system and its ability to provide overpressure protection which would prevent exceeding the TS Section 3.1.2 (Figures 3.1.2 a or b) and Appendix G limits for the reactor pressure vessel during plant cooldown and startup, the inspectors reviewed the engineering evaluation (Document 7 above.) This engineering evaluation which was performed to justify increasing the PORV stroke times from 2.0 sec. to 2.5 sec., affirmed the LTOP setpoint at which the PORV's would open to be 400 psig, and that this value was constant for temperatures of 350 degrees Fahrenheit and below. For this analysis, the inspectors determined that the bounding case ie., large mass input transient due to inadvertent charging pump startup was conservative, given that the safety injection pumps power supply breakers are racked out below 350 degrees Fahrenheit in accordance with TS 3.3.1.3. The inspectors examined the supporting licensee documentation contained in Document (7) above and concluded that the LTOP set point was supported by plant specific analysis and that there is reasonable assurance that the system can effectively prevent RCS overpressurization for the stated conditions. Within this area, the inspectors also reviewed the licensee's 10 CFR 50.59 evaluation included in Documents (2) and (7) above.

b. Administrative Controls and Procedures

The inspectors reviewed the licensee's procedural controls necessary to mitigate the LTOP transient and determined that the licensee does utilize a procedure Document (9) above to manually align the LTOP system for operation which requires verification of valve lineups, and that equipment de-energization is being accomplished in accordance with this procedure. The inspectors also determined that Document (10) above addresses RTGB annunciator alarms which would alert operators to the automatic operation of the LTOP system. The inspectors verified that Document (8) above does procedurally control

the allowable temperature differentials between the steam generators and reactor vessel, while in a water-solid condition, and that collectively Documents (8) and (9) above restrict the use of high head safety injection pumps when the RCS is in a low-temperature overpressure condition.

Additionally, the inspectors determined that Document (8) above, provides for removal of LTOP on RCS temperature increase such that inadvertent actuation of this system at power is avoided, and that the TS had been modified to reflect the actually installed system.

c. Training and Equipment Modifications

Within the area of operator training, the inspectors reviewed the 1986 training records and determined that all operators received training concerning RCS low-temperature overpressure events, the function of the system that is used to mitigate the event, and the consequences of inadvertent actuation, including reporting requirements. The inspectors determined that the licensee does procedurally disable the high head safety injection pumps prior to decreasing RCS temperature below 350 degrees F and that similarly, the safety injection accumulator isolation valve breakers are opened at this point, as described in Document (9) above. Additionally, the licensee de-energizes the pressurizer heater in Document (9) above and opens the pressurizer heater circuit breakers prior to lowering the RCS level as described in Document (11) above.

Modifications to the system such that an alternation to a specific piece of equipment will not result in the equipment being out of its design basis are identified in Document (12) above. This procedure requires the review of proposed modification packages by qualified personnel to ensure that unreviewed safety questions, technical specification changes or conflicts, and Final Safety Analysis Report changes or additions are identified. In order to verify the adequacy of this program, the inspectors selected the recent change to the LTOP system as described in Document (7) above and determined that the required reviews had been performed, including appropriate valve operability testing.

In reviewing the original plant modification which installed the LTOP system described in Document (2) above, the inspectors determined that the nitrogen accumulators, provided for each PORV, were sized to provide 100 cycles of valve operation in the event of loss of N2 from the storage bottles. This redundancy and capacity is well within the design requirements of the system and constitutes adequate design margin.

The inspectors also determined that the instrumentation and control system does not incorporate an alarm with a set-point below the maximum allowable pressure for existing temperature conditions, to

alert the operator of a pressure transient. The lack of an anticipatory alarm in the above noted condition was not identified as a deficiency in that, as previously described, the bounding case for overpressurization occurred from mass input. With the high head safety injection pumps and accumulator isolation valves de-energized, the next transient source mass input would be the three positive displacement charging pumps. The limiting case then would be with one charging pump running for normal letdown and purification, and that the remaining two charging pumps would be inadvertently started. This transient, in a water solid condition, would not allow for operator initiated response based on an anticipatory alarm since the PORV's (one or both) would lift almost immediately.

d. Surveillance

The inspectors ascertained that the licensee was complying with their TS requirements as specified in Section 4.1, Table 4.1-1, Item 31, which requires monthly verification of the operability of the overpressure protection system channels. Implementation of the surveillance requirements is achieved through the execution of Document (13) above, which confirms proper annunciator response and checks control voltages. Additionally, the inspectors examined the PORV stroke times for the most recent test, which was performed on May 31, 1987, in accordance with Document (14) above. Section 5.1 of the subject procedure confirmed that the PORV opening strike times were within the design basis requirements.

Surveillance requirements, as specified in Document (13) above, also require that, prior to placing the unit in cold shutdown, the operability of the LTOP system be verified. The inspectors examined the March 29, 1987 records (Unit going to cold shutdown for refueling) pertaining to Document (13) above, and determined that the LTOP system had been tested, and verified operable. Similarly, the licensee verifies the operability of the LTOP system prior to plant heat-up in accordance with Document (15) above. The records for the May 5, 1987 execution of this test were examined and found to be in compliance with the stated acceptance criteria.

As determined by the inspectors, the licensee is also constrained by the requirements of Document (15) above, to operationally test the LTOP system subsequent to any system, valve, or electronics maintenance.

No violations or deviations were identified within the areas inspected.