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

Report No: 50-261/89-08

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: March 11 - April 17, 1989

Inspectors: R E Carroll Jr 5/18/89
L. W. Garner, Senior Resident Inspector Date Signed

R E Carroll Jr 5/18/89
K. R. Jury, Resident Inspector Date Signed

Approved by: H C Dance 5/18/89
H. C. Dance, Section Chief Date Signed
Reactor Projects Section 1A
Division of Reactor Projects

SUMMARY

Scope:

This routine, announced inspection was conducted in the areas of operational safety verification, surveillance observation, maintenance observation, ESF system walkdown, onsite followup of events at operating power reactors, mid-loop operations, dry storage of spent nuclear fuel, onsite review committee, drawing system verification, and followup on previous inspection items.

Results:

One violation was identified involving a failure to follow a procedure. A personnel error by a licensed operator resulted in a reactor trip, paragraph 6.a.

Using TI 2515/101 as a guide, the inspectors verified that the licensee's actions prior to and during mid-loop operations, successfully addressed the recommended expeditious actions of Generic Letter 88-17, paragraph 7.

A failure of an Electro-hydraulic power supply resulted in a reactor trip. The failure appeared to be a result of aging balance of plant equipment, paragraph 6.b.

The practice of lifting leads to unnecessarily defeat safeguard logic to preclude inadvertent ESF logic actuations during shutdowns was discussed with plant management, paragraph 4.

A drawing system verification was conducted in accordance with Resident Action Item 88-01. No areas of significant concern were identified, paragraph 10.

During this inspection period, the first dry shielded canister containing spent fuel was successfully loaded into the horizontal storage module, paragraph 8.

Housekeeping practices in the cointainment vessel need improvement as demonstrated by the various miscellaneous pieces of debris and trash observed during a management inspection, paragraph 2.

REPORT DETAILS

1. Licensee Employees Contacted

R. Barnett, Maintenance Supervisor, Electrical
C. Bethea, Manager, Training
R. Chambers, Engineering Supervisor, Performance
D. Crocker, Supervisor, Radiation Control
*D. Crook, Senior Specialist, Regulatory Compliance
*J. Curley, Director, Regulatory Compliance
*C. Dietz, Manager, Robinson Nuclear Project Department
R. Femal, Shift Foreman, Operations
W. Flanagan, Manager, Design Engineering
W. Gainey, Support Supervisor, Operations
D. Knight, Shift Foreman, Operations
D. McCaskill, Shift Foreman, Operations
R. Moore, Shift Foreman, Operations
*R. Morgan, Plant General Manager
D. Myers, Shift Foreman, Operations
D. Nelson, Maintenance Supervisor, Mechanical
*M. Page, Acting Manager, Technical Support
D. Quick, Manager, Maintenance
D. Seagle, Shift Foreman, Operations
*J. Sheppard, Manager, Operations
R. Steele, Acting Supervisor, Operations
H. Young, Director, Quality Assurance/Quality Control

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

*Attended exit interview on April 24, 1989.

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

2. Operational Safety Verification (71707)

The inspectors observed licensee activities to confirm 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. These activities were confirmed by direct observations, tours of the facility, interviews and discussions with licensee management and personnel, independent verifications of safety system status and LCOs, and reviews of facility records.

Periodically, the inspectors reviewed shift logs, operation's records, data sheets, instrument traces, and records of equipment malfunctions to verify operability of safety-related equipment and compliance with TS. Specific items reviewed include control room logs, auxiliary operator logs, operating orders, and equipment tag-out records. Through periodic observations of work in progress and discussions with operations'staff members, the inspectors verified that the staff was knowledgeable of plant conditions; responded properly to alarm conditions; adhered to procedures and applicable administrative controls; and was aware of equipment out-of-service, on-going surveillance testing, and maintenance activities in progress. The inspectors observed shift changes to verify that continuity of system status was maintained and that proper control room staffing existed. The inspectors also observed that access to the control room was controlled and operations personnel were carrying out their assigned duties in an attentive and professional manner. The control room was observed to be free of unnecessary distractions. The inspectors performed channel checks, reviewed component status and safety-related parameters, including SPDS information, to verify conformance with TS.

During this reporting interval, the inspectors verified compliance with selected LCOs. This verification was accomplished by direct observation of monitoring instrumentation, valve positions, switch positions, and review of completed logs and records.

Plant tours were routinely conducted to verify the operability of stand-by equipment; assess the general condition of plant equipment; and verify that radiological controls, fire protection controls, and equipment tagging procedures were properly implemented. These tours verified the following: the absence of unusual fluid leaks; the lack of visual degradation of pipe, conduit and seismic supports; the proper positions and indications of valves and circuit breakers; the lack of conditions which could invalidate EQ, the operability of safety-related instrumentation; the calibration of safety-related and control instrumentation, including area radiation monitors and friskers; the operability of fire suppression and fire fighting equipment; and the operability of emergency lighting equipment. The inspectors also verified that housekeeping was adequate (except as noted below) and areas were free of unnecessary fire hazards and combustible materials.

On April 10, 1989, the inspectors accompanied plant management during their closeout inspection of the CV. This inspection was conducted in preparation of Unit restart after work had been performed in the CV. Several items of trash and miscellaneous debris were observed and removed. These included: small pieces of insulating materials, partial rolls and loose pieces of duct tape, small plastic bags, and cable tie wraps. The need to improve housekeeping practices in the CV was discussed with plant management.

In the course of the monthly activities, the inspectors included a review of the licensee's physical security and radiological control programs. The inspectors verified by general observation and perimeter walkdowns that measures taken to assure the physical protection of the facility met current requirements. The inspectors verified station personnel adhered to radiological controls.

No violations or deviations were identified within the areas inspected.

3. Monthly Surveillance Observation (61726)

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

EST-048 (revision 6) Control Rod Drop Test

OST-406 (revision 6) TSC/EOF/PAP Diesel Generator

OST-617 (revision 8) Low Voltage Fire Detection and Actuation System Zones 24, 25A, 25B, 25C, and 26

OST-910 (revision 11) Dedicated Shutdown Diesel Generator

No violations or deviations were identified within the areas inspected.

4. Monthly Maintenance Observation (62703)

The inspectors observed several maintenance activities on safety-related systems and components to ascertain that these activities were conducted in accordance with approved procedures and TS. The inspectors determined that these activities did not violate LCOs, and that redundant components were operable. The inspectors also determined that activities were accomplished by qualified personnel using approved procedures, QC hold points were established where required, required administrative approvals and tagouts were obtained prior to work initiation, proper radiological controls were adhered to, and the effected equipment was properly tested before being returned to service. In particular, the inspectors observed/reviewed the following maintenance activities:

WR/JO 89-AEDN1 - C S/G Tube Inspection

WR/JO 89-AEDS1 - HVH 4 Tube Inspection for Presence of Biological Fouling

WR/JO 89-ADUB1 - Troubleshooting and Repair of E-H Power Supply
WR/JO 89-AEBK1

For several years, the licensee has had an established practice of defeating safeguard actuation logic when safeguards is not required to be in-service per TS. Specifically, logics for phase A and B containment isolation, containment spray, and SI initiation are defeated by lifting leads from their respective actuation relay coils. The lifting of leads and system restoration is controlled by procedure SPP-011, Removal and Restoration of SI Actuation. This procedure contains verification sign-off steps that the lifted leads are reconnected. However, the verification is not specified to be an independent process (i.e., the second person does not necessarily verify the restored connections are tight). Furthermore, neither post-maintenance nor functional testing of the restored system is performed. The manner in which safeguards is defeated does not result in abnormal indications in the control room; restoration of the system is strictly governed by administrative controls. The failure of these administrative controls is unlikely; however, the potential consequences of having safeguards unavailable if needed, are large (i.e., severe accident). In the past, utilization of SPP-011 has sometimes been for reasons other than personnel safety and equipment protection (i.e., eliminating the chance of an inadvertent ESF actuation, thereby reducing the number of reportable events). Defeating safeguard logic for these other reasons constitutes an unnecessary defeating of safeguards. The currently authorized method of and reasons for defeating safeguards is a poor practice. The inspectors discussed with plant management recommended controls for lifted wire leads contained in IEN 84-37, Use of Lifted Leads and Jumpers during Maintenance or Surveillance Testing. The inspectors requested that management review under what conditions SPP-011 should be utilized, and whether development of an alternate methodology is practical, preferably a method which would provide indication that safeguards is defeated. Management agreed to review their current utilization of SPP-011.

No violations or deviations were identified within the areas inspected.

5. ESF System Walkdown (71710)

The inspectors performed a field walkdown of portions of the RHR system located in the RHR pit and inside the CV. Additionally, the SI accumulator discharge piping from the accumulators to the RCS cold legs was inspected. The inspectors verified that the sub-systems are configured as shown on drawings 5379-1082 sheet 4 (revision 20) and sheet 5 (revision 25) and 5379-1484 sheet 1 (revision 16). Items examined included pumps, valves, instrumentation, piping, and pipe supports. The inspectors verified that all major valves were in their correct position,

manual valves were locked as required, instrumentation was valved into service, and that power was available to MOVs, as indicated by the RTGB indicators and MCC breaker positions. Minor seal weepage (approximately 1 drop or less per minute) was observed on both the A and B RHR pumps. The licensee was already aware of and evaluating this condition. Additionally, two conduit clamps on the hard conduit associated with SI-862A, the RHR suction valve from the RWST, were observed to be missing and loose. This condition was reported to the licensee for correction. No conditions which could render these sub-systems incapable of performing their safety function were observed.

No violations or deviations were identified within the areas inspected.

6. Onsite Followup of Events at Operating Power Reactors (93702)

a. Personnel Error Causes Reactor Scram

On March 22, 1989, at 2:22 a.m., the reactor experienced a low-low steam generator level trip from 100% power. All ESF systems performed as expected. The trip occurred when an operator inadvertently closed MS-V1-3A, the A main steamline isolation valve, while performing OST-202, Steam Driven Auxiliary Feedwater System Component Test. Step 7.2.30.3 of OST-202 requires closing MS-V1-8C, the SDAFW C steam supply isolation valve. During performance of this step, the nearby MS-V1-3A switch was operated instead of the MS-V1-8C switch. Failure to follow procedure OST-202 is a violation: Failure to Follow OST-202 Results in a Reactor Scram, (89-08-01).

The licensee performed a post trip review and returned the Unit to service at 5:27 p.m., on March 22. The inspectors reviewed the post trip review and do not have any unresolved concerns.

b. E-H Power Supply Failure Initiates a Reactor Trip

On March 30, 1989, at 3:20 a.m., the reactor experienced a turbine trip/reactor trip from 100% power. The plant responded as expected to the transient and all ESF systems functioned as required. Subsequent investigation by the licensee revealed that both the A and B E-H +15 Vdc power supplies had blown fuses. Troubleshooting of the A +15 Vdc power supply revealed that several transistors were leaking. As a result, the licensee replaced five transistors in the power supply. Testing of other components in both the A and B E-H power supplies did not reveal further cause for the failures. The licensee determined that the most probable failure scenario involved the following sequence:

- Failure of the transistors in the A +15 Vdc power supply resulted in an excessive output voltage.

- A sensing circuit detected the overvoltage condition and blew the A +15 Vdc power supply fuse as designed. This circuit also appeared to prepare the B power supply to pick up the load.

- This preparation or feedback, plus the loading of the B power supply, coupled with an output voltage set close to its maximum limit, resulted in the B +15 Vdc power supply output fuse being blown by its output voltage sensing circuit.

The licensee was unable to verify this failure sequence as the vendor no longer employed anyone technically familiar with this obsolete type of power supply. The original drawings of the power supplies had been transmitted to the licensee in late 1988; however, they could not be located. After repairs were completed, testing was performed to demonstrate that the +15 Vdc power supplies would function properly. Included in this testing was turning the power supplies on and off one at a time, with the turbine on-line and below the turbine trip/reactor trip setpoint. The inspectors witnessed portions of this troubleshooting and testing. In addition, the inspectors attended a special PNSC which evaluated the actions taken prior to Unit restart. The inspectors determined that the licensee performed as thorough a root cause determination as possible with the information available.

During the November 1988 refueling outage, the licensee had originally planned to replace the E-H power supplies with a state of the art model. However, due to miscommunication with the vendor, all the required replacement components were not ordered. The licensee then attempted to rebuild the power supplies; however, they were limited to only a partial overhaul due to lack of available parts. The most likely failure cause was end of life or aging. Much of the BOP control systems are original hardware which is outdated. This situation has been discussed with plant management. A study, initiated earlier this year, is in progress to evaluate what E-H system improvements need to be made. Results of the study should be available by the end of 1989.

c. Loose Part in C S/G

On April 2, 1989, at 7:09 p.m., 7:57 p.m., and 8:12 p.m., both the secondary and primary side LPMS channels associated with the C S/G alarmed. The last alarm sealed-in and could not be reset. Tours of the secondary side, both inside and outside of the CV, as well as the primary side of the C S/G, revealed no external cause for the LPMS alarm. Evaluation of the audio signals indicated that there was a loose part in the C S/G hot leg channel head. As a result, at 3:52 a.m. the reactor was placed in hot shutdown. Based upon a recommendation from Westinghouse, the C RCP was left in service until preparations were completed to enter the C S/G. Retrieval of

the loose part required the Unit to be placed in mid-loop operation. Inspection of mid-loop operations per TI 2515/101 is discussed in paragraph 7.

On April 7, during removal of the manway diaphragm, a loose part fell out of the S/G. Having prepared for such a potential event, the part was rapidly retrieved with tongs and placed in a lead pig. Visual inspection by a remote monitor identified the piece as a control rod guide tube support pin nut (split pin nut). The piece was found to be approximately 1 inch in length, 0.75 inches in diameter, and 0.25 lbs. in weight. The piece was shipped to Westinghouse, which subsequently confirmed the piece as being a split pin nut with the threaded section of the split pin stud contained within. Examination by a scanning electron microscope of the pin shank fracture face revealed that the failure of the pin shank below the threaded area resulted from stress corrosion cracking.

No previous failures of split pins have occurred at HBR; however, similar failures have occurred at other plants. The only difference between this occurrence and other industry occurrences is the long service time before failure. Typically, split pin failures have occurred within the first four or five years of commercial operation. HBR has been in operation for eighteen years. Westinghouse has performed a safety evaluation, SECL-89-630, which justifies operation of the Unit until the next refueling outage. The inspectors reviewed this evaluation and attended the special PNSC meeting on April 10, which reviewed Westinghouse's input and authorized restart of the Unit. In summary, restart was authorized based upon the following:

(1) Operation with a broken split pin is not a concern since the remaining section is trapped between the guide tube flange and upper core plate, thereby still providing lateral support for the guide tube as designed.

(2) Examination of the C S/G internals, including tubesheet, tube ends, tube to tubesheet welds, divider plates, and tube sheet to divider plate welds, did not reveal conditions that required immediate repair (initial phase of the inspection was observed by the inspectors).

(3) Failure of other split pins would not cause damage to the reactor or its internals which would result in a safety concern. Transport of a loose part into a S/G would result in a forced shutdown to remove it.

With regard to item (1) above, prolonged operation could result in wear of the pin body, thereby allowing a limited amount of lateral displacement of the bottom of the guide tube. This could result in additional RCCA frictional forces with the side of the guide tube,

resulting in an increase in scram time for the affected RCCA. Based upon analysis for similar plants, the maximum scram time is not anticipated to exceed the HBR TS value. However, as no analysis was performed specifically for HBR, Westinghouse recommended the following actions be taken: (1) verify rod drop times prior to power operation; (2) review previous rod drop data to assure no increasing trend of rod drop times; and (3) evaluate rod stepping tests for abnormalities. On April 11, the licensee performed cold rod drop tests and subsequently determined that there was not an increasing trend for any rod. The inspectors independently reviewed the rod drop test times since November 1984, and determined that there was not a discernible pattern which would indicate degradation of any RCCA scram time. Cold rod drop times ranged between 1.14 and 1.30 seconds. Comparison of previous hot and cold test data indicates there is no significant difference between the times measured at cold conditions and those measured at hot conditions.

The licensee anticipates an action plan will be developed by July 1989, to address this issue during the upcoming refueling outages. This issue is an IFI: Review Long Term Resolution of Split Pin Cracking Issue, (89-08-02).

d. RHR System Leakage Design Basis Not Met

On April 10, 1989, the licensee determined that potential leakage into the RHR pit could not be successfully isolated under certain accident conditions as required by the system design basis. Actions taken by the licensee to address this item, as well as the inspectors' verification of these actions, will be discussed in Inspection Report 89-09.

One violation was identified in the areas inspected.

7. Loss of Decay Heat Removal, GL 88-17 (TI 2515/101)

On October 17, 1988, the NRC issued GL 88-17, Loss of Decay Heat Removal. On January 3, 1989, the licensee submitted to the NRC what actions they had taken in implementing the eight GL recommended expeditious actions. The inspectors reviewed the licensee's response and verified that the actions committed to had been implemented prior to entering mid-loop operations on April 6, 1989.

The inspectors verified, via review of the lesson plan, Mid-loop Operations and Loss of Decay of Heat Removal, and training records from TI-906 and TI-301, that licensed operators were provided training on previous applicable industry events that occurred during mid-loop operations. This included the April 10, 1987 Diablo Canyon event, as well as events at San Onofre 2 and Waterford 3. For each case, the sequence of events, safety concerns, and contributing factors were addressed with

emphasis placed on applicability to HBR. Special topics included pressurization, vortexing, RHR flow changes, and instrumentation problems. Through discussions with licensed operators, the inspectors verified they were knowledgeable of the concerns associated with mid-loop operations. The procedural changes to address the GL 88-17 action items were developed subsequent to this training. The inspectors verified, via interviews with operations personnel, that shift training was conducted on the revised procedures prior to the shifts standing watch during mid-loop operations. The inspectors consider item 1 of GL 88-17 as being satisfactorily addressed.

The inspectors reviewed OMM-030, Control of CV Penetrations During Mid-Loop Operations, revision 0, and GP-008, Draining the Reactor Coolant System, revision 18. The inspectors verified that these procedures contain sufficient detailed instructions to establish and maintain CV integrity as described in GL 88-17. This meets GL 88-17 item 2. GP-008 also provides the requirements to record four core exit thermocouple temperatures (one of the designated two trains) every 15 minutes and log water level values every 15 minutes. The water level values are taken remotely from control room instrument LT-403 and locally at the B loop standpipe (associated with LT-403) and at a standpipe installed on loop C. The inspectors evaluated the routing of the C loop standpipe to ensure that no conditions existed to render the indicated level inaccurate. The inspectors also visually verified that the B loop and C loop standpipes provided consistent level data and that this data, as well as temperature data was recorded every 15 minutes. In addition, GP-008 provides that one SI pump and one charging pump must be available with injection pathways, prior to reducing water level to -36 inches below the RV flange. These procedure requirements are sufficient to address items 3, 4, and 6 of GL 88-17.

GP-008 also contains a caution note to: "Avoid any evolution that could result in perturbation of the RCS while in a reduced inventory condition. This includes anything that could impact RCS level, the operating train of RHR, or any required support equipment." The inspectors consider this caution note coupled with the recovery steps provided in AOP-020, Loss of Residual Heat Removal (Shutdown Cooling), Revision 6, to be adequate to address item 5 of GL 88-17.

The licensee's response concerning use of aluminum nozzle covers on the S/G to preclude materials from falling into the RCS is considered adequate to address item 7 of the GL 88-17. However, the inspector requested the licensee to establish measures to ensure this item is addressed if nozzle dams are used in the future at HBR. The inspectors determined that item 8, concerning loop stop valves, is not applicable to HBR, as HBR does not have loop stop valves.

Based upon the above described inspections, the inspectors considered that the licensee has taken satisfactory interim measures to address the eight expeditious actions of GL 88-17 (which preclude or successfully mitigate

the loss of decay heat removal during reduced inventory conditions.) The inspectors plan to review implementation of the six programmed enhancement recommendations of GL 88-17 as they are completed.

No violations or deviations were identified in the ares inspected.

8. Loading of First DSC Into HSM (TI 2690/004)

On March 16, 1989, the inspectors witnessed the alignment and loading of the first DSC containing spent fuel assemblies into the HSM. The inspectors verified that the docking of the trailer and movement of the DSC into the HSM was performed in accordance with procedure ISFS-004, Alignment and Loading of the Dry Shielded Canister into the Horizontal Storage Module, revision 4. The inspectors verified that HP practices and procedures were adhered to during this operation.

Additionally, the inspectors reviewed ISFS-006, Startup Monitoring of the Horizontal Storage Module, revision 0. ISFS-006 acceptance criteria for dose rates are less than 200 mrem/hr (neutron + gamma) at center of air inlets or outlets and front access cover, and less than 50 mrem/hr (neutron + gamma) at other locations. Data taken indicated that these acceptance criteria were met. Maximum total neutron and gamma doses measured were 40.5 mrem/hr at the front air outlet (200 mrem/hr allowable), 20.5 mrem/hr on the roof (50 mrem/hr allowable), and 15.5 mrem/hr at the air inlet (200 mrem/hr allowable). The maximum neutron dose measured was 2.5 mrem/hr in front of the HSM. In addition, the air temperature rise through the HSM was shown to be 28 degrees F, well below the 100 degrees F allowable.

On April 12, 1989, a second DSC was loaded into the HSM. Measured radiation dose rates for the second DSC were similar to those described above for the first DSC. The inspectors plan to periodically monitor future activities involving the DSCs and their respective loadings into the HSMs.

No violations or deviations were identified in the areas inspected.

9. Onsite Review Committee (40700)

The inspectors evaluated certain activities of the PNSC to determine whether the onsite review functions were conducted in accordance with TS and other regulatory requirements. In particular, the inspectors attended the March 30, 1989 PNSC concerning the E-H power failure, the April 10, 1989 PNSC concerning the loose part in C S/G and RHR pit leakage, and the April 13, 1989 PNSC concerning the RHR pit leakage issue. It was ascertained that provisions of the TS dealing with membership, review process, frequency, and qualifications were satisfied. The inspectors followed up on selected previously identified PNSC activities to independently confirm that corrective actions were progressing satisfactorily.

No violations or deviations were identified within the areas inspected.

10. Drawing System Verification

The inspectors performed an inspection per RAI 88-01 of the licensee's drawing control system. The inspection consisted of: (1) a drawing quality review (legibility); (2) current revision verification; (3) verification that drawings reflect as-built configuration; (4) review of the drawing backlog requiring changes; and (5) review of controls to maintain marked up drawings prior to permanent revisions. Items (1) and (5) involved review of the controlled copies assigned to the control room. Item (2) involved controlled copies assigned to the control room and the EOF. The drawing control system associated with drawings important to facility operation was found to be adequate, with minor exceptions.

Legibility of drawings was found to be acceptable. All major flow path components on P&IDs were readable. Cross references from one line to another were readable. A small number of vent valves, drain valves and instrument numbers were legible, although difficult to read. Examples include: instrument TE-3092C on G-190199 sheet 1, valve SW-88 on G-190199 sheet 6, instruments TX-1683B and PX-1619 on G-190199 sheet 9, valves SW-548 and SW-549 on G-190199 sheet 4, and valve CAR-10 on G-190197 sheet 2. On sheet 11 of G-190199, a valve adjacent to valve SW-616 on piping line 2-CW-280 was not readable. One example was identified in which the revision number was not legible (drawing 5379-3232). Each P&ID maintained in the control room for major systems are in plastic folders. This practice greatly reduces the amount of drawing wear and tear which would normally otherwise occur. Safeguards System and Reactor Protection System drawings contain many relay contacts and instrument numbers which are difficult to read. However, by comparing similar components in the other division, process of elimination, and/or deductive reasoning, (e.g. sequence and patterns of labeling nomenclature), it is possible to deduct the proper numbers. This area could be improved.

Eighty-eight drawing revisions were audited for latest revision number by comparing the field revision with that contained in the master index or with the current document control drawing. Discrepancies were not identified in the control room drawings set; however, the drawings in the EOF were observed to have three discrepancies. An outdated revision of drawing G-1990199 had not been removed. Sheets 3 and 4 of drawing 5379-1082 were misfiled with HBR2-8255 sheet 5. Revision 13 of drawing 5379-3238 was filed in the proper place; however, the latest revision, number 14, was filed in front of the set of drawings. The TSC drawing set was not audited, as the same individual responsible for the EOF drawings is also responsible for the TSC drawings. The licensee conducts an audit program which verifies proper revisions and current condition of each controlled drawing set. During review of this program, the inspectors discovered that the audit scope and frequency were not procedurally well defined. Thus, when drawing personnel were changed in 1987, the audit

scope changed due to a misunderstanding of what had been done previously. The inspectors also determined that the master index (hard copy) had four erroneous entries (i.e., revision numbers were not updated when the drawings were issued). This condition apparently occurred earlier in the year when reassignment of duties resulted in a different individual being responsible for maintaining the master index current. In each instance, the computer-based system, utilized by operations, contained the correct information.

Selected portions of the SI, RHR, and Feedwater Systems drawings were as-built verified by field walkdowns. No major discrepancies were identified. The P&IDs as structured by the licensee do not contain instrument vent, drain, and other instrument valving downstream of the instrument root isolation valves. The inspectors observed that check valves were not consistently tagged. Also, several valve tags were missing. Examples include FCV-1424 and small test vent and drain valves. With one exception, all configurations were correct as shown. Flow element FE-1425C and associated root valves AFW-96 and AFW-97 are shown on sheet 4 of G-190197 as being located in the AFW pump room. A November 1988 outage modification moved the flow element to the auxiliary building hallway. The inspectors were unable to determine before issuance of this report if a change had been initiated to correct G-190197.

Revisions to drawings are controlled in accordance with MOD-004, Plant Drawing Preparation, Revision, and Approval. Attachment 6.3 of MOD-004 requires updating of priority A drawings within 28 working days of submitted changes to drafting. Priority B drawings are required in 35 working days and priority C drawings are to be issued within 60 days. Flow, logic, safeguard, reactor protection, and control wiring diagrams are assigned a priority A. Valve and instrument lists, as well as piping and electrical penetration drawings are categorized as priority B. The drawings in the control room, TSC and EOF are generally priority A drawings. A review of the outstanding drawing revision index did not reveal a backlog of priority A and B drawings, (i.e., outstanding A and B priority drawing revisions were either in process, or the change had just been received). Review of revision issue dates for this year indicated that the 28 and 35 day requirements were being met. During the last refueling outage, priority A and B drawings were completed prior to declaring modifications operable. Thus, there are no "red-lined" drawings currently in use. MOD-004 does provide for issuing a drawing revision notification to all controlled drawing copy holders if a drawing revision is not anticipated to be available when needed.

In summary, the drawing control system is well maintaining drawings most likely required to be referenced during normal and emergency operating situations with only minor exceptions as noted above. The exceptions were discussed with the licensee for correction as they deem necessary.

No violations or deviations were identified within the areas inspected.

11. Licensee Action on Previously Identified Inspection Items (92701)

(CLOSED) LER 87-30, Non-redundant Power Supply To Vital Equipment Due to Original System Design. On December 20, 1988, the licensee declared Modification M-966 operable. M-966 corrected the single failure design problem by installing two redundant pressure switches, PC-600A and PC-601A, such that each switch is associated with only one division of SI-862 and SI-863 valves. Switch PC-600A is wired into the control circuit for SI-862B and 863B; switch PC-600B is wired into the control circuit for SI-862A and 863A. Hence, the failure of one switch would affect only the valves of its respective division. The inspectors verified that the redesigned circuit meets the single failure criteria, the switches had been calibrated, and the acceptance test performed was adequate to demonstrate proper functioning of the valve interlocks.

(OPEN) IFI 88-28-05, Licensee to Develop Methodology to Detect Biological Growth in HVH 1-4. During the November 1988 refueling outage, the licensee modified the SW System to control the biological growth in the SW System by chlorination. The purpose was to reduce the potential for biological fouling of components served by the SW System (i.e., the CV fan coolers). On April 9, 1989, the inspectors witnessed inspection of the HVH-4 upper tube bundle. An insignificant amount of what may have been biological material was observed at the tube edges. This material did not effect the SW flow through the cooler; however, six tubes were partially blocked by foreign material, possibly weld slag. The small number of affected tubes have no impact on the operability of the cooler. Based upon the inspection, the chlorination program appears to be successfully controlling biological growth.

During early April, a monitoring system was installed to help detect fouling in HVH-4. This item remains open pending review of the effectiveness of this system and the effectiveness of the chlorination system during different seasons.

No violations or deviations were identified within the areas inspected.

12. Exit Interview (30703)

The inspection scope and findings were summarized on April 24, 1989, with those persons indicated in paragraph 1. The inspectors described the areas inspected and discussed in detail the inspection findings listed below and those addressed in the report summary. Dissenting comments were not received from the licensee. Proprietary information is not contained in this report.

Item Number	Description/Reference Paragraph
89-08-01	VIO - Failure to Follow OST-202, Resulting in a Reactor Scram, paragraph 6.a.
89-08-02	IFI - Review Long Term Resolution of Split Pin Cracking Issue, paragraph 6.c.

13. Acronyms and Initialisms

AOP	Abnormal Operating Procedure
AFW	Auxiliary Feedwater
BOP	Balance of Plant
CV	Containment Vessel
DSC	Dry Shielded Canister
E-H	Electro-hydraulic
EOF	Emergency Operation Facility
ESF	Engineered Safety Feature
EST	Engineering Surveillance Test
F	Fahrenheit
FCV	Flow Control Valve
FE	Flow Element
GL	Generic Letter
GP	General Procedure
HBR	H. B. Robinson
HP	Health Physics
HSM	Horizontal Storage Module
HVH	Heating Ventilation Handling
IFI	Inspector Followup Item
ISFS	Independent Spent Fuel Storage
LCO	Limiting Condition for Operation
LER	Licensee Event Report
LPMS	Loose Parts Monitoring System
LT	Level Transmitter
MCC	Motor Control Center
MOV	Motor Operated Valve
mrem/hr	millirem/hour
MS	Main Steam
NSSS	Nuclear Steam Supply System
OMM	Operations Management Manual
OST	Operations Surveillance Test
P&ID	Piping and Instrumentation Diagram
PAP	Personnel Access Portal
PNSC	Plant Nuclear Safety Committee
RAI	Resident Action Item
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RHR	Residual Heat Removal

RTGB	Reactor Turbine Generator Board
RV	Reactor Vessel
SDAFW	Steam Driven Auxiliary Feedwater
S/G	Steam Generator
SI	Safety Injection
SPDS	Safety Parameter Display System
SPP	Special Process Procedure
SW	Service Water
TI	Temporary Instruction
TS	Technical Specification
TSC	Technical Support Center
Vdc	Volts Direct Comment
WR/JO	Work Request/Job Order