U. S. NUCLEAR REGULATORY COMMISSION

REGION III

Report No.: 50-331/90014(DRP)

Docket No.: 50-331 License No.: DPR-49

Licensee: Iowa Electric Light and Power Company
IE Towers, P. O. Box 351
Cedar Rapids, IA 52406

Facility Name: Duane Arnold Energy Center

Inspection At: Palo, Iowa

Inspection Conducted: June 28 through August 24, 1990

Inspectors: M. Parker
C. Miller
J. McCormick-Barger
D. Butler
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Approved: R. L. Hague, Chief
Reactor Projects Section 3C

9/4/90
Date

Inspection Summary

Inspection on June 28 through August 24, 1990 (Report No. 50-331/90014(DRP))

Areas Inspected: Routine, unannounced inspection by the resident inspectors and regional based inspectors of followup; licensee event reports followup; followup of events; operational safety; maintenance; surveillance; temporary instruction (SIMS II.K.3.19 closed); refueling; installation and testing of modifications; local leak rate testing and report review.

Results: The reactor was shutdown for a planned 69 day refueling outage on June 28, 1990, and remained shutdown throughout the period. On July 9, 1990, a Notification of Unusual Event was declared due to the loss of a vital bus resulting from improper switchyard relay testing (section 4.b). Numerous engineered safeguard feature (ESF) actuations were received this period as a result of maintenance or modification activity errors (section 4.a). Inadequate control and supervision of contractor personnel has been the cause for numerous maintenance deficiencies, especially those associated with drywell activities. The licensee issued Corrective Action Report (CAR) 90-01 on August 13, 1990, to solicit corrective action for these deficiencies. An unresolved item was issued to followup on procedural non-conformances associated with these deficiencies (section 6.b). The Main Steam Isolation Valves (MSIVs) experienced the lowest leakage rate in more than a decade; however, the results were influenced by the mid cycle testing and repairs conducted last September. The feedwater check valves soft seat did not remain effective for one fuel cycle as the elastomer exhibited excessive compression set. The licensee has subsequently removed the soft seat material from the check valves. Housekeeping practices have declined this period (section 5.b). The majority of outage maintenance has been completed, and the licensee estimates that startup will occur in early September.

DETAILS

1. Persons Contacted

R. Anderson, Testing and Surveillance Supervisor
R. Anderson, Assistant Operations Supervisor
+J. Axline, Technical Support Engineer
+P. Bessette, Senior Licensing Engineer
J. Bjorseth, Maintenance Engineering Supervisor
+A. Browning, Acting Manager, Nuclear Licensing
V. Crew, Technical Support Engineer
+M. Deinhammer, Maintenance Engineering
J. Dvorsky, Component Engineering
D. Englehardt, Security Supervisor
D. Fowler, Operations Shift Supervisor
H. Giorgio, Radiation Protection Supervisor
*+R. Hannen, Plant Superintendent, Nuclear
* M. Huting, Quality Control Supervisor
D. Kerr, Fire Marshal
+B. Klotz, Quality Engineering
B. Lacy, Manager, Design Engineering
+R. McGee, Technical Support Engineer
C. Mick, Operations Supervisor
W. Miller, Supervising Engineer, Analysis Engineering
* K. Peveler, Corporate Quality Assurance Manager
* J. Probst, Technical Support Engineer
*+K. Putnam, Technical Support Supervisor
* C. Rushworth, Licensing Engineer
+R. Salmon, Technical Services Superintendent
* K. Schneider, Quality Control
+N. Sikka, Engineering
+E. Sorenson, Maintenance Engineering
S. Swails, Training Superintendent
* G. Van Middlesworth, Assistant Plant Superintendent, Operations
* D. Wilson, Outage Manager
+R. Woodward, Maintenance Engineering
K. Young, Assistant Plant Superintendent, Radiation Protection/Security

NUTECH

C. Johns, Design Engineer

U. S. Nuclear Regulatory Commission (NRC)

*+M. Parker, Senior Resident Inspector
* C. Miller, Resident Inspector
D. Butler, Reactor Inspector
+F. Maura, Reactor Inspector

In addition, the inspector interviewed other licensee personnel including operations shift supervisors, control room operators, engineering personnel, and contractor personnel (representing the licensee).

+Denotes those present at the exit interview on July 27, 1990.

*Denotes those present at the exit interview on August 28, 1990.

2. Followup (92701) (92702)

a. (Closed) Violation (331/87004-07): Surveillance Test Procedure No. 42B015 (Rev. 13), "CSCS Trip System Bus Power Monitors Functional Test", placed both trains of the HPCI/RCIC steam leak detection system into tests at the same time. The inspector reviewed the current revision (Rev. 3 of a new surveillance test procedure format) of Procedure No. 42B015 and electrical drawings associated with the steam leak detection system. The review determined the licensee was adequately performing a functional test of the HPCI/RCIC steam leak detection system bus power monitor circuitry. The inspectors have no further concerns regarding this item, therefore, this item is closed.

b. (Closed) Violation (331/87015-01): Corrective actions taken to preclude repetition of excessive leakage through containment isolation valve CV-4311 were ineffective. Following its last failure in 1987, the licensee machined the disc to correct seating angle. Since then, acceptable as-found leakage rates were obtained in 1988 (5950 sccm) and 1990 (1350 sccm). This item is considered closed.

c. (Open) Unresolved Item (331/88022-01): Account for possible pressure drop between local leak rate testing instrumentation and penetration to ensure penetration is at required test pressure. The licensee stated that additional testing will be performed using 200 ft. of 3/8" tubing to obtain pressure drop vs. flow rate data. This item remains open pending the inspector's review of the licensee's data, and of the local leak rate testing procedure modifications incorporating the results obtained.

d. (Closed) Violation (331/88022-02(DRS)): Inadequate safety evaluation to justify the use of a soft seat material on the feedwater isolation check valves. The inspector reviewed the "draft" final safety evaluation dated November 11, 1989, for Design Change Package (DCP) 1422, and determined that it addressed all areas of concern. The licensee will include the final safety evaluation in DCP 1422.

A review of the local leak rate test results showed that the "A" inboard check valve V-14-3 failed its as-found test with a leak rate of 87,750 sccm. Upon disassembly, the soft seat (Parker Compound E 692-75) was found approximately 0.008" below the surface of the hard seal all around the valve. The soft seat original protrusion was 0.020". An inspection of the ring showed no wear marks, but the material, an ethylene propylene elastomer, exhibited a significant loss of resiliency after one plant operating cycle. The "B" inboard feedwater check valve V-14-1 passed its local leak rate test with 7150 sccm. Its soft seat showed erosion or cratering marks on the lower 180° of the valve circumference while the top 180° looked normal. Cracking of the elastomer was also noted. The protrusion was only 0.005" at the top, zero at 90° and negative in the eroded area. The material exhibited the same loss of resiliency as the "A" inboard valve soft seat.

Based on these observations, the licensee has determined that the soft seats do not remain effective for a full fuel cycle and is returning the feedwater inboard check valve to service without the soft seat. The licensee is aware that removal of the soft seats constitute a design change requiring a new safety evaluation to justify the operability of these valves to perform their containment isolation function without the component which was supposed to correct the leakage problem. The licensee will also address the issue of surveillance testing/preventive maintenance frequency based on past component performance for the inboard feedwater check valves.

3. Licensee Event Reports Followup (92700) (90712)

Through direct observations, discussions with licensee personnel, and review of records, the following event reports were reviewed to determine that reportability requirements were fulfilled, immediate corrective action was accomplished, and corrective action to prevent recurrence had been accomplished in accordance with technical specifications.

(Closed) Licensee Event Report No. 89-013 (331/89013 LL): MSIVs have unacceptable leakage during mid cycle testing. The MSIVs were tested during mid fuel cycle in order to determine the effectiveness of the licensee's corrective actions taken during the last five years to reduce the MSIVs leakage rate to acceptable values (less than 11.5 scfh). (Refer to Inspection Reports No. 50-331/87015; 50-331/88022; and 50-331/89018.) Two of the valves failed their test in the hot condition: "C" outboard at 25.4 scfh and "B" outboard at greater than 173.7 scfh. During cold testing, following repairs, the "B" inboard failed with 37.9 scfh. Its original hot test results were 9.4 scfh. The three valves were disassembled, their alignment improved and the valves were returned to service to complete the fuel cycle.

During the 1990 refueling outage, the licensee tested the MSIVs with satisfactory results except for the "A" main steam line MSIVs which experienced a combined leak rate of 33.4 scfh. Individual valve (inboard/outboard) results were not obtained. The "A" inboard MSIV had experienced stem galling on April 22, 1990, when the valve would not fully open following fast closure testing. The valve was declared inoperable, reclosed, and the "A" outboard MSIV was closed to comply with the Technical Specifications. The licensee stated that the stem galling which took place on April 22, 1990, was the root cause of the "A" main steam line MSIVs excessive leakage rate experienced during the 1990 refueling outage.

As stated in Inspection Report No. 50-331/89018, major modifications to the MSIVs are being performed during this refueling outage. At the time of the inspection, work on the new guide pads (four per valve) was in progress. One goal of the modification was to obtain a true valve bore profile of 15.550" to 15.555". However, due to the deviations in the existing base profile, and in order to maintain a minimum satellite ribs and pads thickness of greater than 3/32", the final bore diameter varies between valves as follows:

1 valve 15.550/15.555"
2 valves 15.530/15.535"
5 valves 15.510/15.515"

As a result, each valve's disc/piston assembly will be machined to its specific bore plus a diametrical clearance of 0.015" to 0.018". Valve bonnets will be machined to match the bore dimension plus a diametrical clearance of 0.003" to 0.014". The licensee is confident that the modifications in progress will resolve the MSIVs leakage problem. The next as-found leak rate will be performed at the next refueling outage.

4. Followup of Events (93702)

During the inspection period, the licensee experienced several events, some of which required prompt notification of the NRC pursuant to 10 CFR 50.72. The inspectors pursued the events onsite with licensee and/or other NRC officials. In each case, the inspectors verified that the notification was correct and timely, if appropriate, that the licensee was taking prompt and appropriate actions, that activities were conducted within regulatory requirements, and that corrective actions would prevent future recurrence. The specific events are as follows:

July 9, 1990 - Notification of Unusual Event/With Scram and Engineered Safety Features Actuations due to loss of Essential Bus Power.

July 9, 1990 - Engineered Safety Features Actuation (PCIS - Group 3 Isolation) due to Modifications work in steam leak detection cabinet.

July 20, 1990 - Engineered Safety Features Actuation (PCIS - Group 3 Isolation) due to personnel error.

July 22, 1990 - Engineered Safety Features Actuation (PCIS - Group 3 Isolation) due to failure to communicate with control room. (Subsequently determined to be non-reportable.)

July 26, 1990 - Engineered Safety Features Actuation (PCIS - Group 3 Isolation) due to personnel error. (Subsequently determined to be non-reportable.)

July 26, 1990 - Engineered Safety Features Actuation (PCIS - Group 3 Isolation) due to personnel error. (Subsequently determined to be non-reportable.)

August 22, 1990 - Engineered Safety Features Actuation (PCIS - Group 3 Isolation) due to personnel error. (Subsequently determined to be non-reportable.)

August 23, 1990 - Reactor Scram Signal with no rod motion due to error in filling transmitter reference leg.

a. ESF Actuations

Due to the high number of ESF actuations occurring in a relatively short time, the resident inspectors discussed with plant management their action being taken to prevent further recurrences of these events. The licensee agreed that these events were primarily personnel errors; however, they were complicated due to the tight quarters in the back panel of the control room and the numerous terminations required to complete one specific design modification. These actuations have been associated with a design modification to install an isolation mimic panel in the control room. This has necessitated performing the modification on an energized panel. After discussions with both NRR and AEOD, the inspectors informed the licensee that they could take action to remove unnecessary systems from service to reduce or eliminate unnecessary ESF actuations. As a result of this action should a partial/spurious actuation occur, this would not be considered an ESF actuation for reportability as a result of the action to remove the system from service. The licensee has subsequently taken this approach on selective equipment where appropriate to reduce unnecessary ESF actuations.

In reviewing these events with the licensee it was subsequently determined that three of the above PCIS Group III isolations were not reportable (July 22 and July 26, 1990) as a result of the licensee's actions to remove the system from service.

b. Loss of Essential Bus Power

On July 9, 1990, a loss of standby transformer power caused a temporary loss of power to both 4160 volt essential busses (1A3 and 1A4). The reactor was shutdown with all fuel removed from the reactor vessel, the "A" EDG out of service for overhaul, the "A" 125V battery disconnected from its Division I bus for testing, and alternate power to 1A3 and 1A4 from the startup transformer secured.

While performing relay testing in the switchyard, a technician failed to insert a relay logic block, and caused Oil Cooled Breaker (OCB) "M" (supply breaker to the standby transformer) and other breakers to trip. This caused both 1A3 and 1A4 to lose power. A subsequent full RPS scram signal and PCIS Group 1 through 5 isolation signals were received. No rod motion occurred due to refueling tagouts on the Hydraulic Control Units. Isolation of some PCIS valves (such as MSIVs and RWCU) were also blocked due to maintenance tagouts. Bus 1A4 was reenergized immediately by the "B" EDG. Bus 1A3 remained deenergized about forty minutes until the licensee determined that OCB "M" did not trip on a fault, and had restored power to the Division 1 125 d.c. bus to regain breaker control power. Busses 1A3 and 1A4 were then reenergized from the standby transformer.

No violations or deviations were identified in this area.

5. Operational Safety Verification (71707) (71710)

The inspectors observed control room operations, reviewed applicable logs and conducted discussions with control room operators during the inspection. The inspectors verified the operability of selected emergency systems, reviewed tagout records and verified proper return to service of affected components. Tours of the reactor building and turbine building were conducted to observe plant equipment conditions, including potential fire hazards, fluid leaks, and excessive vibrations and to verify that maintenance requests had been initiated for equipment in need of maintenance. It was observed that the Plant Superintendent, Assistant Plant Superintendent of Operations, and the Operations Supervisor were well informed on the overall status of the plant and that they made frequent visits to the control room and regularly toured the plant. The inspector by observation and direct interview verified that the physical security plan was being implemented in accordance with the station security plan.

The inspector observed plant housekeeping/cleanliness conditions and verified implementation of radiation protection controls. During the inspection, the inspector walked down the accessible portions of the Residual Heat Removal - Shutdown Cooling System to verify operability by comparing system lineup with plant drawings, as-built configuration or present valve lineup lists; observing equipment conditions that could degrade performance; and verified that instrumentation was properly valved, functioning and calibrated.

These reviews and observations were conducted to verify that facility operations were in conformance with the requirements established under technical specifications, 10 CFR, and administrative procedures.

a. River Water Conditions

Due to the unseasonably high rainfall in the area this spring and summer, river water flow and level have increased significantly. During the past winter, the licensee has had to supplement river water, as required by their water use permit, with water from Pleasant Creek Reservoir. At that time river water flow was averaging about 400 scfs. However, during this spring and summer, the rainfall has resulted in a significant increase in river water flow rates (10 to 100 fold increase). This river water flow has resulted in elevated river water levels. The licensee has continued to monitor river water levels due to local flood warnings.

The average natural grade around the plant varies from 746' to 750' elevation. The plant finished grade is 757' elevation. Technical

Specifications specifies an action level of 753' elevation for additional monitoring of river conditions due to the potential of site flooding.

On August 1, 1990, after significant rainfall up river, the river crested at 742.5' elevation at the pump house intake structure. The river water flow was recorded at 44,900 scfs. While this has resulted in localized flooding both upstream and downstream, the plant site has experienced no significant impact.

b. Housekeeping

The inspectors noted a significant decline in housekeeping practices this period. While housekeeping is generally an increased problem during outages, the inspectors noted that plant conditions had degraded beyond what was acceptable for a safe work environment. After the inspectors met with the licensee to discuss their concerns, the licensee held a short drywell work stoppage to correct housekeeping problems and increase supervisory attention in the drywell. However, temporary services routing, unguarded missing deck grates, equipment stowage, and general cluttering with trash and personal protective equipment continued to be a problem in the drywell. Other problems outside the drywell included unattended solvent and rags left in a battery room, and sandblasting grit scattered throughout the northwest corner room, including on the "B" side ECCS pumps. The inspectors will continue to follow the licensee's corrective actions in this area, as it affects not only personnel but also equipment safety.

No violations or deviations were identified in this area.

6. Monthly Maintenance Observation (62703)

Station maintenance activities of safety related systems and components listed below were observed/reviewed to ascertain that they were conducted in accordance with approved procedures, regulatory guides and industry codes or standards and in conformance with technical specifications.

The following items were considered during this review: the limiting conditions for operation were met while components or systems were removed from service; approvals were obtained prior to initiating the work; activities were accomplished using approved procedures and were inspected as applicable; functional testing and/or calibrations were performed prior to returning components or systems to service; quality control records were maintained; activities were accomplished by qualified personnel; parts and materials used were properly certified; radiological controls were implemented; and, fire prevention controls were implemented.

Work requests were reviewed to determine status of outstanding jobs and to assure that priority is assigned to safety related equipment maintenance which may affect system performance.

The following maintenance activities were observed/reviewed:

- "A" EDG overhaul
- MSIV overhaul and modification
- Reactor Recirculation Pump Motor overhaul
- Station battery repairs
- Feedwater checkvalve overhaul (V-14-001 and V-14-003)
- Residual Heat Removal (RHR) Heat Exchanger (IE-201A) inspection
- "A" and "B" EDG ESW Heat Exchanger inspections
- Control Rod Dive insert/withdraw line repair
- Motor operated valve repairs and testing
- PSV Accumulator check valve repair (V-14-09, V-14-14, V-14-15, V-14-16)
- Lead Test Assembly (LTA) Fuel Bundle inspections
- Secondary Containment repairs

Following completion of maintenance on the emergency diesel generators, the inspector verified that these systems had been returned to service properly.

a. Heat Exchanger Inspections

During the outage, the inspectors observed several heat exchanger inspections (RHR heat exchanger, "A" EDG ESW heat exchanger, "B" EDG ESW heat exchanger, and drywell coolers) as part of the licensee's actions associated with Generic Letter (GL) 89-13, Service Water System Problems Affecting Safety-Related Equipment. Both EDG heat exchangers on the ESW system were found to be in good condition with no tube plugging. The RHR heat exchanger having the same water source (river water) as the ESW was initially observed with no tube plugging; however, during eddy current testing, some tube plugging was evident at the tube bends. The plugging appeared to be caused by a buildup of silt from the river water. During drywell cooler inspections, a buildup of rust was evident, resulting in the need for mechanical rodding of cooler tubes and flushing. The inspectors will continue to observe the licensee's actions associated with GL 89-13.

b. Refueling Outage

The licensee commenced the 1990 refueling on June 28, 1990. A newly formed Outage Management group has been responsible for planning and overall outage management control. General Electric has been given responsibility for all drywell work including quality control. Major outage projects include Control Rod Drive (CRD) bundle repair of fifty lines, reactor recirculation pump "A" and "B" rotating assembly replacement and motor repair, MSIV overhaul and operator replacement.

The resident inspectors have noted repeated instances of inadequate work control by contractor personnel, some of which are identified below:

> Control Rod Drive (CRD) 10-07 withdraw line was mistakenly cut without a freeze seal, resulting in gross leakage into the reactor building.
>
> CRD 26-43 insert line was partially cut on the wrong side of a coupling. Rework included weld repairing the cut and making the proper cut on the insert line.
>
> Reactor Recirculation Discharge Valve MO-4627 was installed without proper torquing of the bonnet bolts. When operability testing was performed the yoke turned before the test was stopped. Further review showed that MO-4628 was also not torqued properly. Since this and other steps were missed by contract personnel, Iowa Electric electricians were directed to reinspect and rework the valves as needed.
>
> Heat numbers for 7 CRD repair lines were scribed with the wrong heat numbers. The mistake was identified before they were installed and the lines were rescribed correctly. An inspection of the installed lines showed no other problems.
>
> An Automatic Depressurization System (ADS) nitrogen check valve (V-14-14) was worked without a maintenance order due to personnel selecting the wrong valve. When the mistake was discovered, field personnel worked on another ADS nitrogen check valve (V-14-15) without a work order to correct the problem.
>
> A reactor recirculation pump motor was over filled with oil. This caused the oil to overflow into the motor windings. The motor was subsequently removed and sent to an offsite facility for rework.

The inspectors met several times with licensee management concerning outage work control. Many of the problems noted involved drywell work which was controlled and inspected for quality by contractor personnel. After some attempts to improve supervision of work in the drywell did not improve overall work quality, the licensee's Quality Assurance organization issued a CAR to the Outage Management organization to solicit corrective actions.

Solutions listed on the CAR involved licensee craft oversight of drywell activities, licensee quality control involvement and realignment of contractor resources to increase supervision effectiveness. The inspectors observed drywell maintenance following the August 16, 1990, implementation of the corrective actions to determine their effectiveness. Some problems with work supervision and communication were noted and expressed to licensee management on August 23, 1990. The licensee, after further investigation, instituted a policy to require supervision present for all drywell maintenance activities. The majority of drywell work had been completed by August 23, 1990; however, the inspectors will continue to monitor contractor control and followup on corrective actions taken by the licensee. The procedural non-conformances resulting from these maintenance items will be followed by the inspectors as an unresolved item (331/90014-01(DRP)).

No violations or deviations were identified in this area.

7. Monthly Surveillance Observation (61726)

The inspectors observed technical specifications required surveillance testing and verified that testing was performed in accordance with adequate procedures, that test instrumentation was calibrated, that limiting conditions for operation were met, that removal and restoration of the affected components were accomplished, that test results conformed with technical specifications and procedure requirements and were reviewed by personnel other than the individual directing the test, and that any deficiencies identified during the testing were properly reviewed and resolved by appropriate management personnel.

The inspector also witnessed portions of the following test activities:

STP-41A014	-	Steam Line High Radiation Instrument Calibration
STP-42C005-(S/D)	-	Control Rod Block Actuation - Source Range Monitor Trip Functional/Calibration Test (shutdown)
STP-43B001	-	Nuclear Instrument Response to Control Rod Motion and Control Rod Coupling Integrity Check
STP-47A005	-	Containment Isolation Valve Leak Tightness Test - Type C Penetrations (V-14-001, V-14-003, CV1412, CV1413, CV1415, CV1416, CV1418, CV1420, CV1421, MO-4441, and MO-2312)
STP-47C001-CY	-	Secondary Containment Integrity
STP-48A002	-	Standby Diesel Generators and Emergency Service Water System Automatic Actuation Test
STP-48A006-SP	-	Cyclic Special Performance Discharge Test
STPNS-55001	-	CRD Friction Testing
SPTP-163-B	-	Heat Exchanger Performance Monitoring

"A" EDG Over Speed Test

"A" EDG Electric Current Test

The inspector witnessed portions of the Main Steam Line (MSL) radiation monitor surveillance. Workers performing the MSL radiation monitor test used good ALARA practices and good technique to prevent unnecessary exposure to the radioactive test source. The inspector notified the licensee that the surveillance procedure did not contain a precise acceptance criteria in that no formula was given to calculate the 30% allowable difference between the survey instrument and the radiation monitor. The licensee has agreed to revise this procedure.

No violations or deviations were identified in this area.

8. Temporary Instruction (TI)

(Closed) Temporary Instruction 2515/065 - TMI Action plan followup for item II.K.3.19. This item is identified in NUREG 0737 as applicable to non jet pump boiling water reactors. Since Duane Arnold Energy Center is a jet pump plant, this item does not apply and is considered closed.

9. Refueling Activities (60710)

The inspectors reviewed the licensee's refueling practices during core off load and reload phases. The review included verification of personnel qualifications, procedure acceptability and adherence, neutron monitoring system and other support system acceptability, actual fuel transfer, and bundle position verification activities from the refuel bridge and remotely.

The inspectors concluded that refueling activities were conducted adequately to ensure the safety and proper location of the fuel was maintained. The following comments apply:

General Electric contractors were used to move the fuel for the first time at DAEC. The crew was experienced and worked well with Iowa Electric operators. Reload sequencing was complicated by repairs on the CRD withdrawal and insert lines. Fuel could not be loaded into a cell without the rod inserted, and a rod could not be inserted before work was complete on its insert and withdraw lines. Refueling was delayed while waiting for CRD repairs, and due to reduced pool clarity. The reactor water cleanup system was not operable during refueling operations due to a system leak; therefore, a temporary cleanup system was used to improve water clarity.

During the refueling process, a roll of duct tape was dropped from the refueling bridge into the reactor vessel. The licensee initially followed the roll visually as it travelled into the shroud annulus region of the vessel near the inlet of jet pump number eleven. It subsequently disappeared and could not be located in the shroud annulus region. The licensee postulated that the roll went through the jet pump and down into the lower plenum region. The licensee chose not to search for the roll in the lower plenum due to the difficulty of search and retrieval in that General Electric subsequently performed a safety analysis evaluating the potential for fuel bundle blockage, interference with control rod operation, and increased corrosion of vessel internals. The analysis determined that the tape material would disintegrate without adverse affects on vessel internals if a controlled heat up procedure was used during startup.

During an inspection of GE fuel lead test assemblies (LTAs), a GE inspection crew noticed a crack indication on the channel of LTA number LYA477. Subsequent inspection determined the crack to be about 7 inches in length, originating from a fastener tack weld near the bottom of the channel. LYA477 and three other LTAs with a similar channel design were removed from service pending further inspection of the damage.

10. Installation and Testing of Modifications (37828)

The inspectors obtained a copy of the licensee's list of Design Change Packages (DCPs) that were being implemented during the current Cycle 10/11 refueling outage. The inspectors selectively chose two minor modifications for detailed review. This review was to provide for the selective inspections of ongoing plant modification activities. The DCPs reviewed and results obtained from the inspections are as follows:

a. DCP-1467, Main Steam Isolation Valves (MSIVs) Solenoid Status Indication.

This modification resulted as part of the corrective actions identified in LER 89-008 concerning a reactor scram that occurred on March 5, 1989. The scram occurred following an MSIV closure during calibration of the main steam line radiation monitors. The "B" MSIV d.c. solenoid coil had failed without the operators knowledge previous to the calibration activity. During the calibration activity, the a.c. solenoid coil was deenergized, per plant procedures, resulting in the "B" MSIV closure and subsequent reactor scram. The a.c. and d.c. solenoid coils work in combination to control a three way valve that normally supplies nitrogen to the MSIV actuator to hold it open. Both solenoids must be deenergized for the MSIV to close. Failure of the d.c. solenoid coil was not readily detectable as no direct indication of solenoid status existed.

DCP-1467 provided in-line resistors for each of the MSIVs a.c. and d.c. solenoid power supplies and associated LEDs wired in parallel with the voltage dropping resistors. The LEDs were installed on panels IC41 and IC42 as applicable. All installation activities were confined to the above mentioned control panels and provides direct indication of MSIV a.c./d.c. solenoid operation in the control room.

From review of the DCP, the inspectors determined that the licensee conducted an adequate 10 CFR 50.59 review and that the modification, which was not required to be submitted for approval to the NRC, was in conformance with the requirements of the licensee's Technical Specifications and 10 CFR Part 50, Appendix B, Criterion III, "Design Control". In addition, the inspectors examined the installed hardware and verified that the installation conformed to the construction drawings. Testing of the installed hardware was not scheduled to be performed until later in the outage, after MSIV repair/replacement activities were to be completed. The inspectors reviewed the proposed testing and determined that it should adequately demonstrate proper operation of the newly installed hardware.

b. DCP-1461, HPCI and RCIC Steam Trap Drain Lines Replacement.

The modification was performed to resolve steam leakage and operability problems with the High Pressure Coolant Injection (HPCI) and Reactor Core Isolation Cooling (RCIC) Turbines steam supply line drainpot condensate collection systems. The plant has a history of condensate collection valve packing and bonnet leaks, drainpot level controller problems, and valve seat leakage problems. The modification replaces the existing drainpot systems with a simpler design using thermal dispersion level sensors, new valves designed to minimize packing and seat leakage problems, and more reliable steam traps consisting of bi-metal bar assemblies to control discharge of condensate.

The inspectors reviewed the DCP and determined that the licensee had conducted an adequate 10 CFR 50.59 review, and that the design change complied with the requirements of the licensee's Technical Specifications and 10 CFR Part 50, Appendix B, Criterion III. The inspectors met with applicable responsible plant and contractor personnel to review and discuss a portion of the eight construction work packages utilized to implement the DCP. The inspectors reviewed a sample of worker welding qualifications and compared construction drawings with a sample of material and as-built dimensions associated with installed piping and components. No concerns were identified.

Although the HPCI/RCIC drainpot modification was minor, the construction and testing activities were scheduled to extend throughout the refueling outage. During the inspection, the inspectors reviewed the engineering acceptance and test requirements and the Modification Acceptance Test procedure. The proposed testing activities appeared to have been prepared in accordance with plant procedures, contained the proper review and approvals, and were designed to demonstrate the proper level of performance for the new systems.

No violations or deviations were identified.

11. Containment Local Leak Rate Testing (61720)

Discussions were held with the licensee regarding their plans to leak rate test the CRD penetrations (at least 50) being repaired as a result of the fatigue cracking being experienced. The following issues were discussed:

a. The repair/modification is considered major and must be followed by either a Type A or Type B test (10 CFR 50, Appendix J, Section IV.A.).

b. A type B test using the primary containment as the test chamber is acceptable. The acceptance criteria would be zero leakage on the outside of each CRD penetration using soap bubble, helium leak detection, or other equivalent test methods.

c. The acceptability of the test method used would be based on the sensitivity of the method and equipment to detect any leakage. Calibration procedures and results must be available to demonstrate the acceptability of the equipment/method used.

12. Report Review (90713)

During the inspection period, the inspectors reviewed the licensee's Monthly Operating Report for May, June, and July 1990. The inspectors confirmed that the information provided met the requirements of Technical Specifications 6.11.1.C and Regulatory Guide 1.16.

No violations or deviations were identified in this area.

13. Unresolved Items

Unresolved items are matters about which more information is required in order to ascertain whether they are acceptable items, violations or deviations. An unresolved item disclosed during the inspection is discussed in Paragraph 6.b.

14. Exit Interview (30703)

The inspector met with licensee representatives (denoted in Paragraph 1) on July 27 and August 28, 1990, and informally throughout the inspection period and summarized the scope and findings of the inspection activities. The inspector also discussed the likely information content of the inspection report with regard to documents or processes reviewed by the inspector. The licensee did not identify any such documents or processes as proprietary. The licensee acknowledged the findings of the inspection.