U. S. NUCLEAR REGULATORY COMMISSION
REGION 1

Report Nos. 50-334/90-18 License: DPR-66
50-412/90-18 NPF-73

Licensee: Duquesne Light Company
One Oxford Center
301 Grant Street
Pittsburgh, PA 15279

Facility Name: Beaver Valley Power Station, Units 1 and 2

Location: Shippingport, Pennsylvania

Dates: July 28 - September 7, 1990

Inspectors: J. E. Beall, Senior Resident Inspector
P. R. Wilson, Resident Inspector

Approved by: William H. Ruland 10/2/90
William Ruland, Chief Date
Reactor Projects Section No. 4B

Inspection Summary

This inspection report documents routine and reactive inspections during day and backshift hours of station activities including: plant operations; radiological protection; surveillance and maintenance; emergency preparedness; security; engineering and technical support; and safety assessment/quality verification.

Results

Overall the facility was operated safely. One violation was identified concerning the failure to follow procedures which resulted in a Unit 2 Engineered Safety Feature actuation (Detail 2.3.5). Another violation was identified concerning the failure of the Onsite Safety Committee to review a Unit 1 modification that affected nuclear safety prior to installation (Detail 8.2). One unresolved item was identified concerning a potential design deficiency with respect to the containment isolation dampers (Detail 7). Licensee actions concerning a Site Alert resulting from a partial discharge of carbon dioxide in the Unit 2 West Cable Vault were reviewed (Detail 5). The licensee's Site Management Walkdown Program was found to be a notable strength (Detail 4.3). The licensee's activities concerning the adequacy of a temporary modification concerning a Unit 1 Auxiliary Feedwater pressure switch were reviewed and no significant deficiencies were identified (Detail 8.3). One previous NRC open item was reviewed and closed.

DETAILS

1. Summary of Facility Activities

At the beginning of the period, Unit 1 was holding power at approximately 30 percent power pending steam generator water chemistry improvement and Unit 2 was operating at full power. On July 29, Unit 1 power was raised to 100 percent. On August 2, Unit 1 was shut down due to an inoperable battery bank and charger (See Detail 2.3.1). On August 3, Unit 1 returned to power operation holding power at 30 percent pending steam generator water chemistry improvement. From August 5 until September 2, Unit 1 operated at approximately 100 percent power. On September 2, Unit 1 reduced power to approximately 65 percent due to low demand. On September 4, Unit 1 returned to full power operation and remained at that power level for the remainder of the period.

On August 13, Unit 2 power was lowered to approximately 28 percent to permit repairs of a main feedwater regulating valve. Unit 2 returned to full power operation on August 15 and remained at that level until August 20, when reactor power started end of cycle coastdown. On August 24, power was lowered to approximately 52 percent and then further reduced to approximately 40 percent on August 31 due to low demand. On September 4, Unit 2 was shut down for the second refueling outage. On September 5, Unit 2 entered Cold Shutdown (Mode 5) and remained in Mode 5 for the rest of the period.

2. Plant Operations

2.1 Operational Safety Verification

The inspectors observed plant operation and verified that the plant was operated safely and in accordance with licensee procedures and regulatory requirements. Regular tours were conducted on the following plant areas:

-- Control Room
-- Auxiliary Buildings
-- Switchgear Areas
-- Access Control Points
-- Protected Area Fence Line
-- Spent Fuel Building
-- Turbine Buildings
-- Safeguard Areas
-- Service Buildings
-- Diesel Generator Buildings
-- Intake Structure
-- Yard Areas
-- Containment Penetration Areas

During the course of the inspection, discussions were conducted with operators concerning knowledge of recent changes to procedures, facility configuration and plant conditions. The inspector verified adherence to approved procedures for ongoing activities observed. Shift turnovers were witnessed and staffing requirements confirmed. The inspectors found that control room access was properly controlled and a professional atmosphere was maintained.

Inspector comments or questions resulting from these reviews were resolved by licensee personnel.

Control room instruments and plant computer indications were observed for correlation between channels and for conformance with Technical Specification (TS) requirements. Operability of engineered safety features, other safety related systems and onsite and offsite power sources were verified. The inspectors observed various alarm conditions and confirmed that operator response was in accordance with plant operating procedures. Compliance with TS and implementation of appropriate action statements for equipment out of service was inspected. Logs and records were reviewed to determine if entries were accurate and identified equipment status or deficiencies. These records included operating logs, turnover sheets, system safety tags, and the jumper and lifted lead book. The inspector also examined the condition of various fire protection, meteorological, and seismic monitoring systems.

Plant housekeeping controls were monitored, including control and storage of flammable material and other potential safety hazards. The inspector conducted detailed walkdowns of accessible areas of both Unit 1 and Unit 2. General housekeeping at both units was good.

2.2 Engineered Safety Features System Walkdown

The operability of selected engineered safety feature systems was verified by performing detailed walkdowns of the accessible portions of the systems. The systems inspected during this period include the Emergency Diesel Generators, Safety Injection Auxiliary Feed and Recirculation Spray systems. The inspectors confirmed that system components were in the required alignments, instrumentation was valved-in with appropriate calibration dates, as-built prints reflected the as-installed systems and the overall conditions observed were satisfactory.

2.3 Followup of Events Occurring During the Inspection Period

During the inspection period, the inspectors provided onsite coverage and followup of events. Plant parameters, performance of safety systems, and licensee actions were reviewed. The inspectors confirmed that the required notifications were made to NRC. The following events were reviewed:

2.3.1 Unit 1 Shutdown Due to Failed Battery Charger

At approximately 5:19 p.m. on August 1, 1990, while Unit 1 was operating at 100 percent, control room operators received a No. 2 Battery Charger failure alarm. Subsequent investigation by the licensee found that the A phase input lug to the No. 2 Battery Charger AC input breaker had broken off. The charger was then declared inoperable. Unit 1 Technical Specification (TS) 3.8.2.3.b required that with one full capacity battery charger inoperable, the associated battery bank be demonstrated to be operable within one hour. At approximately 6:06 p.m., the No. 2 battery was declared inoperable due to pilot cell and total voltage being below the limits required by the TS. With one required battery bank inoperable, TS 3.8.2.3.a required the battery to be restored within two hours or be in Hot Standby (Mode 3) within the next six hours. At approximately 8:05 p.m., a controlled Unit 1 shutdown was initiated.

The failed input breaker was a Class 1E Westinghouse 50 amp, 3 phase, 480 volt Model No. EHB 3050. A qualified 50 amp breaker was unavailable. The licensee located a qualified 40 amp breaker (Westinghouse Model No. HFB-3040L) in a Unit 2 spare battery charger. The licensee's Engineering Department performed a technical analysis (Technical Evaluation Report No. 5689) and concluded that the 40 amp breaker would provide adequate protection and would not trip open on the expected charger full load.

At approximately 12:15 a.m. on August 2, 1990, the 40 amp breaker was installed. The No. 2 charger was then placed in service; however, it was found that the charger output voltage could not be adjusted. At approximately 1:52 am, the Unit 1 shut down to Hot Standby (Mode 3) was completed. Further investigation by the licensee found a burned open collector resistor on the battery charter output. The resistor was subsequently replaced and the battery charger was placed in service. By 1:30 p.m., the No. 2 battery was recharged. At approximately 7:25 p.m. the battery and its associated charger were declared operable following the Onsite Safety Committee review of the 10 CFR 50.59 safety evaluation for the 40 amp breaker. On August 3, Unit 1 returned to power operation.

The licensee subsequently performed a root cause analysis to determine the cause of the resistor and 50 amp breaker failure. The resistor failure was attributed to normal end of life event. The phase A lug failure of the 50 amp input breaker could not be definitively determined. The licensee hypothesized that the lug had been inadvertently bent and weakened while performing charger testing during the last refueling outage. When the charger front panel was removed (breaker was attached to panel) while trouble shooting the charger failure alarm, the lug broke off.

The inspector reviewed the licensee's activities associated with the shutdown and repairs to the No. 2 Battery Charger. The inspector identified concerns with the review process for the substitution of the 50 amp input with a 40 amp breaker (see Detail 8.2). Other than these concerns, the inspector found that procedures and TS requirements were properly followed.

2.3.2 Control Room Emergency Bottled Air Pressurization System Actuation

On August 15, 1990, while both units were operating at approximately 100 percent power, the Control Room Emergency Bottled Air Pressurization System (CREBAPS) automatically initiated due to a failed control room radiation monitor. The CREBAPS is designed to provide a source of pressurization for the combined Unit 1 and Unit 2 control room in the event of high outside radioactivity, high control room atmosphere radioactivity or a significant outside chloride leak. The system maintains the control rooms at a positive pressure for approximately one hour.

A Unit 1 control room atmosphere radiation monitor (RM-IRM-218A) failed high causing an automatic initiation of CREBAPS. Control room operators promptly isolated the CREBAPS air bottles after verifying a high radiation condition did not exist. This action exceeded the requirements of both unit's Technical Specification (TS) 3.7.7.1, "Control Room Emergency Habitability Systems," and TS 3.0.3 was entered. The failed radiation monitor's output to CREBAPS was disconnected and actuation signal was reset. Approximately 39 minutes after the actuation, the CREBAPS air bottles were unisolated and TS 3.0.3 was exited. The radiation monitor was repaired and returned to service.

The inspector determined that the event was of minor safety significance. Prompt action by the control room operators prevented the unnecessary depressurization of the CREBAPS air bottles which, if depressurized, would have resulted in the shutdown of both units.

2.3.3 Carbon Dioxide Discharge into Cable Vault

On August 30, 1990, at about 10:00 am, an inadvertent partial carbon dioxide discharge occurred in the Unit 2 West Cable Vault. Due to a measured degradation in oxygen level in one space, the licensee declared an Alert in accordance with the Emergency Plan at 11:10 am. The affected areas were purged, adequate oxygen was confirmed, and the Alert was terminated at 1:10 pm. No areas outside the plant were affected. For additional information see Section 5.

2.3.4 Unit 1 Steam Generator Blowdown Line Isolation

On August 31, 1990, while operating at 100 percent power, the three Steam Generator (SG) blowdown line containment isolation valves (TV-BD-100A, B, C), and the three SG blowdown sample line containment isolation valves shut due to the failure of pressure switch PS-FW-157-3. The pressure switch, which senses the turbine drive auxiliary feedwater pump discharge pressure, is designed to trip shut the SG blowdown and blowdown sample isolation valves when the auxiliary feedpump starts. The licensee found that of the pressure switch had a loose wire and some internal corrosion. The switch was subsequently replaced and the SG blowdown and SG blowdown sample valves were re-opened. Since the above valves perform Engineered Safety Feature function, the licensee made the appropriate NRC notifications for the event.

The inspector reviewed the event and questioned if the failed pressure switch was required to be environmentally qualified. The licensee stated that the switch was not required to be environmentally qualified and provided the inspector with the appropriate justification. The inspector determined the event to be of minor safety significance and had no further questions.

2.3.5 Unit 2 Letdown Isolation Due to Failure to Follow Procedure

On September 2, 1990, the Unit 2 Chemical and Volume Control System normal letdown line inadvertently isolated during the performance of Operating Surveillance Test (OST) 2.1.11D, "Safeguards Protection System Train A CIA Go Test." The normal letdown isolation valves provide an Engineered Safety Feature isolating the normal letdown line on a containment isolation signal. At the time of the event, Unit 2 was operating at approximately 40 percent power.

The OST tests several safeguards protection system relays. Step 3 of the surveillance tested relay K605A. Actuation of this relay causes the Containment Instrument Air System isolation valves IAC*MOV133 and IAC*MOV130 to shut. To protect the containment instrument air compressors (located outside containment), the OST required that the station instrument air backup supply valve IAC- MOV131 be opened and then the OST directed the containment air compressors to be shut down. Following the actuation of the relay and verification that the containment isolation valves associated with the K605A relay shut as required, the OST then directed the operator to reset the relay, open IAC*MOV130 and IAC*MOV133, restart a containment air compressor and then shut IAC*MOV131. Step 6 of the OST tested relay K614A/614XA. One of the valves which receives an automatic closure signal when this relay is actuated is 2CCP*MOV118, which isolates cooling water to the containment air compressors. Therefore, Step 6 directed the operators to open No. 2IAC-MOV131 and shut down the running air compressor prior to actuating the relay.

Prior to starting the OST, the control room operators, in consultation with the shift foreman (second Senior Reactor Operator on shift), decided, in order to minimize the start-stop cycles on the containment air compressors, that containment air compressors would not be restarted and that the containment instrument air containment isolation valves would not be re-opened as required by Step 3 of the OST. The operators mistakenly believed that the containment instrument air loads would be supplied by the normal instrument air system via IAC-MOV131. This valve, however, was located upstream of containment isolation valve IAC*MOV130, and therefore no air could be supplied to containment instrument air loads as long as IAC*MOV130 was shut.

Approximately 25 minutes after the containment instrument air containment isolation valves were shut in Step 3 of the OST, normal letdown valves 2CHS*AOV200A and 2CHS-AOV200C (letdown orifice isolation valves) drifted shut, isolating the letdown line due to low air pressure. The control room operators promptly responded by establishing Excess Letdown and re-established containment instrument air pressure, and then re-established normal letdown. The required NRC notification was made.

The inspector reviewed the event and found that the cause of the letdown line isolation was the failure to follow the OST as written. If the control room operators had re-established the Containment Air System pressure as required by Steps 3.o and 3.p, the event would have been avoided. The inspector also questioned why the operators did not initiate an Operating Manual Change Notice (OMCN) to revise the procedure, deleting Steps 3.o and 3.p even though the control room operators had made a deliberate decision to deviate from the procedure as written prior to performance of the procedure. If an OMCN had been prepared, the required Nuclear Shift Supervisor review of the OMCN might have prevented the event.

The inspector also found that weaknesses in the OST procedure contributed to the event. The procedure lacked caution statements prior to the steps isolating the Containment Instrument Air System, reminding the operator that the system would be lost. Also, the valve description for IAC*MOV131 led the operators to mistakenly believe that Containment Instrument Air System would remain pressurized via the normal Instrument Air System. The procedure described the valve as the "Instrument Air System backup to 21AC-TK23" (Containment Instrument Air System receiver tank). Licensee evaluation of potential corrective actions was still in progress at the end of the inspection period.

The licensee had been issued three separate Notices of Violation (Level 4) for events caused by failure to follow procedures during surveillance testing in the last two years (see Inspection Reports 50-334/89-04; 50-412/89-04, 50-334/89-12; 50-412/89-13, and 50-334/89-23; 50-412/89-22). While the inspector found that the isolation of the normal letdown line had only minor safety significance, the failure to follow the OST as written is a Violation (50-412/90-18-01).

2.3.6 Unit 2 Letdown Isolation Due to Low Pressurizer Level

On September 4, 1990, while Unit 2 was in Hot Standby (Mode 3), an inadvertent Chemical and Volume Control System normal letdown line automatic isolation occurred due to low pressurizer level. At the time of the event, a cooldown to Hot Shutdown (Mode 4) was in progress. The control room operators stopped two of the three operating reactor coolant pumps in accordance with the cooldown procedure, increasing the cooldown rate. The control room operator became distracted from monitoring pressurizer level by 2B steam generator level control concerns. The 2B steam generator level had been decreasing and the operator took action to prevent an automatic start of the auxiliary feedwater pumps on low SG water level. Due to the reactor coolant cooldown, pressurizer level decreased to 14 percent and the letdown line automatically isolated as designed. Pressurizer level was promptly restored to its program level and normal letdown flow was re-established. The required NRC notification was made.

The inspector reviewed the event and determined it to be of minor safety significance. The licensee root cause analysis of the event was in progress at the end of the period.

2.3.7 Unit 2 Containment Purge Duct Isolation

On September 6, 1990, while Unit 2 was in Cold Shutdown (Mode 5), the containment purge supply and exhaust duct inside containment isolation dampers automatically shut due to high radiation signal during a containment atmosphere purging evolution. Control room operators observed that the inside containment purge exhaust duct isolation damper 2HVR*MOD23B failed to completely shut and immediately took action to completely close the damper.

There are two Unit 2 radiation monitors which generate isolation signals to the containment purge line isolation dampers. HVC*RQ104A provides the isolation signal for the outside containment isolation dampers and HVC*RQ104B provides the isolation signal to the inside containment dampers. Both radiation monitors sample the containment exhaust line duct. The licensee determined that the high radiation signal generated by HVC*RQ104B was caused by an electrical spike due to a lightning strike during a severe thunderstorm. Containment air samples indicated airborne activity levels were normal. Containment atmosphere purge was subsequently re-established.

The licensee determined that the cause of HCV*MOD23B not fully closing was due to the isolation signal from HCV*RQ104B not being of sufficient duration to allow the motor operated damper to fully close. The licensee performed surveillance tests for the damper and radiation monitor. No deficiencies were identified.

The inspector reviewed the event and determined it to be of minor safety significance. The inspector had no further questions with respect the operational aspects of the event. An additional design concern involving the damper closure logic is discussed in Detail 7.

2.3.8 Event Performance Assessment

The inspector reviewed the licensee's performance during the above events. The inspector found with the exception of the letdown line isolation described in Detail 2.3.5 that the control room operators' responses to each event were good and performed in accordance with approved site procedures. The licensee's corrective actions appeared to adequately address the root causes of the events. The unit letdown line isolation described in Detail 2.3.5 was of particular concern because the surveillance procedure was deliberately not followed and the operators involved did not attempt to formally revise the procedure as required.

3. Radiological Controls

Posting and control of radiation and high radiation areas were inspected. Radiation Work Permit compliance and use of personnel monitoring devices were checked. Conditions of step-off pads, disposal of protective clothing, radiation control job coverage, area monitor operability and calibration (portable and permanent) and personnel frisking were observed on a sampling basis.

There were no notable observations.

4. Maintenance and Surveillance

4.1. Maintenance Observation

The inspector reviewed selected maintenance activities to assure that:

-- The activity did not violate Technical Specification Limiting Conditions for Operation and that redundant components were operable;

-- required approvals and releases had been obtained prior to commencing work;

-- procedures used for the task were adequate and work was within the skills of the trade;

-- activities were accomplished by qualified personnel;

-- where necessary, radiological and fire preventive controls were adequate and implemented;

-- QC hold points were established where required and observed;

-- equipment was properly tested and returned to service.

Maintenance activities reviewed included:

MWR 903789 - Repair Unit 1 Waste Gas Compressor GW-C-1B

MWR 906027 - Repair Service Water Vacuum Breaker SWS-487

MWR 901200 - Uncouple, Install Temporary Lube Oil Pump and Recouple Unit 1 Turbine Driven Auxiliary Feedwater Pump

MWR 901330 - Replace Pressure Switch 1PS-MS-111

There were no notable observations.

4.2 Surveillance Observation

The inspectors witnessed/reviewed selected surveillance tests to determine whether properly approved procedures were in use, details were adequate, test instrumentation was properly calibrated and used, Technical Specifications were satisfied, testing was performed by qualified personnel and test results satisfied acceptance criteria or were properly dispositioned. The following surveillance testing activities were reviewed:

Unit 1

OST 1.30.6 Reactor Plant River Water Pump 1C Test

OST 1.2.1 Nuclear Power Range Channel Functional Test

OST 1.24.2 Motor Driven Auxiliary Feed Pump Test (1FW-P-3A)

Unit 2

OST 2.1.11C Safeguards Protection System Train A CIB/Spray Actuation Test

OST 2.13.3 Recirculation Spray Pump (2RSS*P21A) Dry Test

OST 2.36.2 Emergency Diesel Generator (2EGS*EG2-2 Monthly Test)

OST 2.24.3 Motor Driven Auxiliary Feed Pump (2FWE*P23B) Test

There were no notable observations.

4.3 Site Management Walkdown

The inspector noted continued significant improvements in general housekeeping and the material condition of plant components. The inspector reviewed one licensee initiative, walkdowns performed by licensee management, which has contributed to these improvements.

In June 1989, the licensee implemented a program (Site Administrative Procedure (SAP)-58 "Plant Inspection Program") by which site managers were assigned to perform inspections of the facility approximately every three weeks. The site was divided into several inspection zones. The managers were assigned to different inspection zones each inspection. The inspections were separated into the following four different categories: Material Condition, Industrial Safety, Housekeeping, and Radiological Protection. Each inspection concentrated on one category to allow an in-depth look at one specific aspect of plant performance. The inspector accompanied site managers during an inspection and observed that other deficiencies outside the category selected for the inspection were not overlooked.

The inspector found that inspection deficiencies were being resolved in a timely manner and there was only a small backlog of items awaiting resolution. For each identified material condition deficiency, a maintenance work request was generated, given the appropriate priority and entered into the maintenance tracking system.

The inspection program effectiveness was clearly evident by the observed improvements at both units. The inspector found the inspection program to be fully implemented and constituted a notable licensee strength.

5. Emergency Preparedness

The inspector monitored licensee activities during and following the Alert which was declared for Unit 2 on August 30, 1990 (see also Detail 2.3.2).

5.1 Inadvertent Carbon Dioxide Discharge

Routine fire protection system testing was in progress with the three carbon dioxide tanks isolated (manual valves 2FPD-HCV-202A, 202B and 203 for tanks 2FPD-TK-22, 23, and 24 respectively). Two tanks are normally available with the third in reserve. At about 10:00 a.m., annunciators indicated that a carbon dioxide discharge had occurred; this was confirmed by phone reports from affected areas. Search and rescue procedures were implemented and all personnel were verified clear of the affected areas. All potentially involved areas were tested for adequate oxygen levels, ventilation purging was conducted where necessary, and affected areas were monitored until normal oxygen levels had been restored.

The cause of the carbon dioxide release was determined to be a leak through 2FPD-HCV-202B. The release was terminated when timers automatically closed the downstream valves as part of the test. The reserve tank was placed on line, full system operability was restored, and maintenance was scheduled for 2FPD-HCV-202B.

5.2 Emergency Plan Implementation

The control room staff recognized that the carbon dioxide release required an evaluation of emergency action levels when oxygen levels in affected areas were measured to be below the normal value of about 20 percent. The licensee declared an Alert at about 11:10 a.m. when one area was determined to be about 15 percent oxygen. The inspector was notified following the apparent carbon dioxide discharge and entered the control room shortly after the Alert was declared. The inspector monitored the control room staff response and recovery from the discharge throughout the rest of the event.

The licensee notified the emergency response organization, and subsequently activated the Technical Support Center, the Operations Support Center, the Radiological Operations Center, and placed the Emergency Operations Facility on Standby. The on-site facilities were activated at approximately 11:50 a.m. The Media Center was not activated; media functions were performed at the corporate offices in Pittsburgh.

Off-site activation also took placed. The Pennsylvania Emergency Management Agency and the Ohio Emergency Management Agency were fully activated. West Virginia activated key personnel. Two of the three primary counties were activated. The third primary county (Hancock, West Virginia), had a limited activation and all backup counties and affected municipalities were placed on standby.

The event was terminated at about 1:10 p.m. with the restoration of normal oxygen levels in all areas. Termination was discussed with, and agreed to, by all affected agencies.

5.3 Emergency Preparedness Plan Review

The declaration of an Alert was consistent with a conservative interpretation of the licensee's Emergency Preparedness Plan (Issue 8, Rev. 6) for a toxic gas release (Tab 18). Lacking specific guidance for carbon dioxide or oxygen levels, the control room staff elected to characterize the reduced (about 15 percent) oxygen levels as equivalent to the presence of toxic gas.

The Plan's classification criteria in Tab 18 were based on the guidance provided in NUREG-0654, "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants." The NUREG includes example initiating conditions and one example for an Alert is the "experienced or projected [e]ntry into facility environs of uncontrolled toxic or flammable gases."

NUREG-0654 considers an Alert to be appropriate in response to events which involve an actual or potential substantial degradation of the level of plant safety. The Alert is, intended, among other things, to assure that offsite emergency organizations are readily available for such tasks as offsite radiation monitoring. Other examples in the NUREG of initiating conditions for Alert are severe loss of fuel cladding, rapid gross failure of one steam generator with loss of offsite power, and any tornado striking the facility.

The inspector told the licensee that the discharge of carbon dioxide as designed from a permanently installed fire protection system did not appear to represent a substantial degradation of plant safety, especially in the absence of a fire, normally associated with an Alert. The licensee acknowledged the inspector's concern and stated that the Emergency Plan would be reviewed.

5.4 Summary

The operators promptly identified that a carbon dioxide discharge had occurred despite a correct valve lineup to preclude that possibility. Measures to ensure personnel safety were taken rapidly and efficiently. It is the inspector's opinion that the operators' declaration of an Alert was consistent with a broad interpretation of the current wording of the Emergency Preparedness Plan. Following the Alert declaration, all required notifications were made promptly and the licensee consulted with, and obtained the agreement of, all offsite groups prior to termination. The inspector identified the treatment within the Plan of carbon dioxide discharge by installed fire protection systems as an area for further review.

6. Security

Implementation of the Physical Security Plan was observed in various plant areas with regard to the following:

-- Protected Area and Vital Area barriers were well maintained and not compromised;

-- Isolation zones were clear;

-- Personnel and vehicles entering and packages being delivered to the Protected Area were properly searched and access control was in accordance with approved licensee procedures;

-- Persons granted access to the site were badged to indicate whether they have unescorted access or escorted authorization;

-- Security access controls to Vital Areas were being maintained and that persons in Vital Areas were properly authorized;

-- Security posts were adequately staffed and equipped, security personnel were alert and knowledgeable regarding position requirements, and that written procedures were available; and

-- Adequate illumination was maintained.

There were no noteworthy observations.

7. Engineering and Technical Support

During the September 6, 1990 Unit 2 Containment purge duct isolation, one of the dampers did not go fully closed automatically (See Detail 2.3.7). Unit 2 is committed to IEEE 279-1971, "Criteria for Protection Systems for Nuclear Power Generating Stations." The IEEE Standard requires that protection systems be designed so that "once intitiated, a protective action at the system level shall go to completion" (paragraph 4.16).

The isolation dampers provide Containment isolation which is within the scope of the IEEE Standard. The dampers are normally de-energized and locked shut during plant operation. The dampers are open during refueling including fuel handling activities such that they would provide protection during a hypothetical fuel handling accident. The IEEE Standard scope includes actuation of protection features following a serious reactor incident.

The isolation dampers' design did not include a seal-in contact or similar feature. The omission of a seal-in feature is a potential deficiency with respect to the commitment to IEEE 279-1971. This item is Unresolved (50-412/90-18-02) pending further reviw of the applicability of IEEE 279-1971 design requirements to the normally de-energized, locked shut Containment isolation dampers.

8. Safety Assessment and Quality Verification

8.1 Review of Written Reports

The inspector reviewed LERs submitted and other written reports to the NRC Region I Office to verify that the details of the events were clearly reported, including accuracy of the description of cause and adequacy of corrective action. The inspector determined whether further information was required from the licensee, whether generic implications were indicated and whether the event warranted onsite followup. The following LERs were reviewed:

Unit 1:

LER 90-010-00 Inadvertent Letdown Isolation Due to Main Steam Isolation Valve Stroke Testing

LER 90-011-00 ESF Actuation - Inadvertent Trip of 1A Reactor Coolant Pump Caused by Relay Testing

LER 90-012-00 Plant Shutdown Due to Failure of No. 2 Battery Charger

Unit 2:

LER 90-008-00 Reactor Trip/Turbine Trip Due to Protection Relay Actuation

The above LERs were reviewed with respect to the requirements of 10 CFR 50.73 and the guidance provided in NUREG 1022. Generally, the LERs were found to be of high quality with good documentation of event analyses, root cause determinations, and corrective actions.

The inspector noted that the Cause of Event section of Unit 1 LER 90-012-00 exhibited weakness in that it did not describe the root cause of a 50 amp breaker failure.

8.2 Onsite Safety Review Committee Modification Review

On August 1, 1990, the Class 1E 480 VAC, 50 amp input breaker to the Unit 1 No. 2 Battery Charger failed. This failure ultimately required a Unit 1 shutdown (see Detail 2.3.1). The licensee was unable to locate a qualified 50 amp replacement breaker; however, a qualified 40 amp breaker was found and installed in the No. 2 Battery Charger following a technical analysis by the licensee's Engineering Department. Unit 1 Technical Specification (TS) 6.5.1.6.c states that "Onsite Safety Review Committee (OSC) shall be responsible for the review of all proposed changes or modifications to the plant systems or equipment that affect nuclear safety" (emphasis added). The No. 2 Battery Charger affects nuclear safety in that it provides power to one of four divisions of vital DC loads, and maintains the No. 2 battery charged. The 40 amp breaker was installed and the No. 2 charger was subsequently used to recharge its associated battery without prior OSC review.

10 CFR 50.59 authorizes licensees to make changes to the facility as described in the facility's safety analysis report, unless it involves a change in the Technical Specifications or an unreviewed safety question. The affected chargers and its associated input breaker were described in the Unit 1 Updated Final Safety Analysis Report (UFSAR), Figure 8.4-2. The UFSAR described the input breaker as a 50 amp breaker and the charger input full load rated amperes was indicated to be 44 amps. The 40 amp breaker was installed and its associated charger was used to recharge the No. 2 battery without the preparation of a 10 CFR 50.59 evaluation to determine if a Technical Specification change was required or if unreviewed safety question existed. It was unclear that the newly installed 40 amp was adequate to prevent the breaker's inadvertent tripping if the full load current described in the USFAR was reached.

The inspector discussed the above concerns with the licensee. The licensee contended that the OSC review of the modification and a 10 CFR 50.59 evaluation were not required to be performed prior to the installation of the modification, but that both were required to be completed prior to declaring the affected system, subsystem or component to be operable.

The inspector found no clear guidance whether performing a 10 CFR 50.59 evaluation after modification but before the component was declared operable was acceptable. However, the 40 amp breaker was used to charge its associated safety related battery before an OSC review was performed. Unit 1 TS 6.5.1.6c clearly required the OSC to review the modification while it was still a proposal, and therefore before use. This is a Violation (50-334/90-18-01).

8.3 Unit 2 AFW Operability Issue

On August 30, 1990, the licensee removed the Unit 1 Turbine Driven Auxiliary Feedwater (TDAFW) pump from service for testing. The testing was a measure taken following receipt of NRC Information Notice No. 90-45, "Overspeed of the Turbine-Driven Auxiliary Feedwater Pumps and Overpressurization of the Associated Piping Systems."

The Notice details recent events at other sites which the TDAFW pump did not trip on overspeed as expected leading to potential overpressurization of piping. Both Units 1 and 2 TDAFW pumps each have a discharge relief valve in addition to a recirculation valve. In the case of Unit 1, these were added after construction via a modification in the 1976-7 time frame. During the testing, pressure switch PS-1MS-111 was found to have failed. The switch senses inlet steam pressure to the TDAFW pump and allows the recirculation valve to open when steam pressure exceeds 400 psig. The valve is air operated and fails shut on loss of the non-safety air system. The apparent purpose of the switch, though not found explicitly stated in the records, is to assure that TDAFW flow under low steam pressure conditions will go to the steam generators and not just be recirculated. At the time of the modification, the system was not a safety system so records were limited with respect to design bases.

The failed switch was powered from a vital DC source, was seismically mounted, and was procured as Category 1 (though the master equipment list has it as Category 2). In the absence of clear guidance, the control room staff declared the TDAFW pump not operable due to the failed switch. Unable to replace the switch within the time before a unit shutdown was required due to TDAFW pump inoperability, the licensee implemented interim measures to replace the function of the switch and the pump was declared operable shortly before the shutdown would have been required.

The interim measures consisted of a dedicated operator stationed near the pump with a meter which would indicate when the valve should be opened. The operator would then turn a switch allowing air to the valve to open it. The inspector expressed several concerns with respect to the adequacy of the interim measures as an equivalent replacement to the switch and therefore as suitable for declaring operability.

The switch used vital DC power while both the temporary pressure transducer and the operator's meter used non-safety AC power. The switch was seismically mounted while the operator's equipment was not (e.g., the meter was found on a three foot ladder secured with one strip of electrical tape). No guidance was provided on loss of power to the meter. The inspector also questioned the response time assumed for operator action. The licensee was able to locate a suitable replacement switch, the switch was installed, and the interim measures were terminated on September 5, 1990.

Further review and analysis by the licensee revealed that the switch was not required for TDAFW pump operability and represented an enhancement only. Previous vendor calculations showed that the pump required no recirculation flow for at least 20 minutes. This was due to a small recirculation path always in place via flow through the installed lube oil cooler. Alarms in the control room are provided to alert operators of low AFW flow.

The inspector concluded that the control room operators properly, but conservatively, declared the TDAFW pump not operable in the absence of information available. Licensee actions to replace the function with an operator until a replacement switch could be located and installed were well intended but flawed, and might not have constituted an adequate basis for operability if the switch had been a design requirement. The inspector met with licensee management and discussed the issues including the determination whether equipment was inoperable, operable or operable but degraded.

In summary, the inspector found the licensee's actions to be generally good, especially the conservatism exhibited in the operators' decision to declare AFW not operable in the absence of information. Weaknesses were identified in the licensee followup actions, but no safety significant problems were identified.

9. Status of Previous Inspection Findings

The NRC Outstanding Items List was reviewed with cognizant licensee personnel. Items selected by the inspector were subsequently reviewed through discussions with licensee personnel, documentation reviews and field inspection to determine whether licensee actions specified in the OIs had been satisfactorily completed. The overall status of previously identified inspection findings was reviewed, and planned/completed licensee actions were discussed for the items reported below.

9.1 (Closed) Unresolved Item (50-334/87-07-02): Licensee to resolve Auxiliary Feedwater (AFW) Pump 3A performance characteristics. This item had been previously reviewed (See Inspection Reports 50-334/90-02; 50-412/90-02 and 50-334/90-13; 50-412/90-13) and was left open pending the determination of Total Developed Head (TDH) valve assumed in the Updated Final Safety Analysis Report (UFSAR). The licensee provided the inspector with documentation which indicated that the actual TDH of AFW pump 3A was above the valve assumed in USFAR accident analysis. The inspector had no further questions.

9.2 (Closed) Unit 1 Action Plan Requirement (NUREG 0737) Item III.D.3.4.3, Control Room Habitability: This item required the licensee to perform only necessary modifications to ensure that control rooms were safe and habitable under both toxic gas and radiological releases. As documented in a letter dated February 9, 1982, the NRC staff concluded that no modifications were required for the Unit 1 control room habitability system.

9.3 (Closed) Unit 2 NUREG-0737 Items: The inspector conducted a review of the docket file for TMI items not documented as closed in an NRC inspection report. The following items were determined not to be applicable to Unit 2 because the features or systems were added to the plant program or design during construction:

I.	D.2.3	Plant Safety Parameter Display Console
II.	B.3.1	Post Accident Sampling - Interim System
II.	B.4.2.A	Training for Mitigating Core Damage - Initial
II.	E.1.2.1.A	Auxiliary Feedwater System

The inspector reviewed the items and found no deficiencies; these items are closed.

10. Exit Meeting

10.1 Preliminary Inspection Findings Exit

Periodic meetings were held with senior facility management during the course of this inspection to discuss the inspection scope and findings. A summary of inspection findings was further discussed with the licensee at the conclusion of the report period on September 14, 1990.

10.2 Attendance at Exit Meetings Conducted by Region-Based Inspectors

No exit meetings were conducted during the inspection period by Region-based inspectors.