U. S. NUCLEAR REGULATORY COMMISSION
REGION I

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License Nos.	DPR-66 NPF-73
Licensee:	Duquesne Light Company One Oxford Center 301 Grant Street Pittsburgh, PA 15279
Facility:	Beaver Valley Power Station, Units 1 and 2
Location:	Shippingport, Pennsylvania
Inspection Period:	March 15 - April 18, 1992
Inspectors:	L. W. Rossbach, Senior Resident Inspector P. P. Sena, Resident Inspector

Approved by: [signature] acting for — May 8, 1992
John Rogge, Chief — Date
Reactor Projects Section No. 4B

Inspection Summary

This inspection report documents core and regional initiative inspections during day and backshift hours of station activities in the areas of: plant operations; radiological protection; surveillance and maintenance; security; engineering and technical support; and safety assessment/quality verification.

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* The NRC manual inspection procedure (IP) or temporary instruction (TI) is listed for each applicable report section.

EXECUTIVE SUMMARY
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Plant Operations

Overall, the units were operated safely. Unit 1 reduced power for two days in a safe and controlled manner in order to accomplish several maintenance and calibration activities of minor safety significance. Unit 2 refueling operations were conducted safely, in accordance with detailed procedures, and in a controlled manner. The licensee was sensitive to reductions in coolant inventory and the potential effects on decay heat removal.

Radiological Protection

Several individuals received unnecessary exposures from a contaminated bucket but there were no overexposures.

Maintenance and Surveillance

Temporary fire stops were installed in two Unit 2 containment penetrations and remained there during core alterations. Air was found leaking from one of these penetrations, an apparent Technical Specification violation. The exhaust from this area was monitored and showed that there was no release. This event, as it occurred, was of minor safety consequence and effective immediate corrective actions were taken. However, this event created the potential for an unfiltered release during a spent fuel handling accident. An enforcement conference is planned to discuss this issue. A self-identified, non-cited violation involving failure to verify the position of a valve was inspected.

Engineering and Technical Support

Several ISI findings by the licensee indicated that the licensee is doing a detailed and critical self-assessment in response to previous violations. Three sequencing relays in both Unit 2 emergency diesel generator sequencing circuits were found to be inoperable. These relays and circuits had been modified during the previous refueling outage. Potential safety implications existed due to this event as the ability of both diesels to properly sequence safety-related loads during accident conditions was degraded. An enforcement conference is planned to discuss this issue.

Safety Assessment/Quality Verification

Good safety perspective was demonstrated by site management in the planning and control of modification activities involving the use of a temporary cooling water supply to a Unit 2 component cooling water heat exchanger.

<u>DETAILS</u>

1.0 SUMMARY OF FACILITY ACTIVITIES

1.1 Licensee Activities

Unit 1 operated at full power throughout this inspection period except for manual power reductions on April 13 and 14 due to secondary water chemistry limits and an out of calibration analog rod position indication. These issues are discussed in Section 2.2.

Unit 2 remained in the cycle III-IV refueling outage throughout this inspection period. Defueling began on March 25. During routine testing on March 30, three sequencing relays for an emergency diesel generator were found to be inoperable. This issue is discussed in Section 6.2. Refueling began on April 8. On April 9, air was found leaking from a containment penetration through which temporary cables had been routed for the outage. This event is discussed in Sections 2.4 and 4.3. At the end of this inspection period, Unit 2 was in Mode 6 as preparations were being made to tension the reactor vessel head.

1.2 NRC Staff Activities

This inspection assessed the adequacy of licensee activities for reactor safety, safeguards, and radiation protection. The inspectors made this assessment by reviewing information on a sampling basis. Information was obtained through actual observation of licensee activities, interviews with licensee personnel, and documentation reviews.

Inspections were conducted on both normal and backshift hours: 30 hours of direct inspection were conducted on backshift; 10 hours were conducted on deep backshift. The times of backshift hours were adjusted weekly to assure randomness.

An inspection of engineering and technical support activities was conducted from March 16 to 27, 1992 (NRC inspection report 50-334/92-07; 50-412/92-05).

An inspection of physical security activities was conducted from March 16 to 19, 1992 (NRC inspection report 50-334/92-08; 50-412/92-06).

An inspection was conducted from April 13 to 17, 1992, to assess the radiation protection program and occupational exposure controls during the Unit 2 outage (NRC inspection report 50-334/92-06; 50-412/92-03).

An inspection of the radioactive liquid and gaseous effluent control programs was conducted from April 13 to 17, 1992 (NRC inspection report 50-334/92-10; 50-412/92-09).

2.0 PLANT OPERATIONS (IP 71707, 71710, 93702, 60710)

2.1 Operational Safety Verification

Using applicable drawings and check-off lists, the inspectors independently verified safety system operability by performing control panel and field walkdowns of the following systems: emergency diesel generators; reactor vessel level indicating system; fire protection/service water system interface; and decay heat removal. These system were properly aligned. 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 of the following plant areas:

- Control Room
- Auxiliary Buildings
- Switchgear Areas
- Access Control Points
- Protected Areas
- Spent Fuel
- Diesel Generator Buildings
- Safeguard Areas
- Service Buildings
- Turbine 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 inspectors 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. Inspectors' 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 inspectors 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 inspectors conducted detailed walkdowns of accessible areas of both Unit 1 and Unit 2. Housekeeping at both units was adequate.

2.2 Unit 1 Power Reduction Due to Secondary Water Chemistry and Rod Position Indication

On April 13, 1992, a manual power reduction was initiated due to secondary water chemistry limits increasing to chemistry program action levels. Power was reduced to 29% in accordanc with procedures due to an increase in steam generator cation conductivity. Due to indication of main condenser tube leakage, the A waterbox was isolated to prevent further circulating water in-leakage. Licensee inspection of the main condenser revealed 19 damaged tubes which were subsequently plugged. The cause of the tube degradations was water impingement due to inadvertent actuation of the heater drain tank high-level dump on April 12. Baffle plates are installed in the main condenser to minimize the effects of water impingement on the tubes. The adequacy of the installed baffle plates is under review by the licensee. The heater drain tank high-level dump valve, which failed in the midposition due to instrument air leaks within the controller, has been repaired. Condenser hotwell feed and bleed operations, combined with steam generator blowdown, successfully reduced secondary chemistry to within acceptable limits.

Following water chemistry cleanup, the licensee initiated power escalation on April 14. However, when at 98% power, the licensee entered Technical Specification (TS) 3.1.3.1, "Moveable Control Assemblies," due to indication of misaligned rod. Bank D control rod H2 position indicated greater than 12 steps from its group step counter demand position. Specifically, the analog rod position instrument (ARPI) channel indicated 220 steps while the demand position was at 202 steps. In accordance with the TS action statement, the licensee reduced power to 71% within one hour. The TS limiting condition for operation was subsequently exited following the power reduction as rod H2 indicated proper alignment with demand position due to the inward rod movement. Troubleshooting indicated the need for recalibration of the ARPI for rod H2. An actual rod misalignment never existed. The plant was returned to 100% power on April 15 following successful calibration of the ARPI.

The inspectors concluded that both power reductions were of minor safety significance. Operations personnel responded to the equipment failures in a safe and controlled manner. The prompt power reduction and isolation of the A waterbox contributed to minimizing the severity of the secondary chemistry degradation.

2.3 Unit 2 Refueling Activities

The inspectors observed the licensee's preparations for entering Mode 6 and subsequent refueling activities in order to verify that the refueling operations were conducted safely and in accordance with technical specification requirements. The observed activities included: draining the reactor coolant system (RCS); reactor vessel head removal; rod cluster control assembly unlatching; and fuel offload.

On March 20, 1992, prior to detensioning the reactor head closure studs, the RCS was drained to two feet below the reactor vessel flange. Although the RCS was not drained to a reduced inventory condition (three feet below the reactor vessel flange) or to midloop, the licensee demonstrated good sensitivity to the reduction in coolant inventory and the potential effects on the residual heat removal system. Two independent tygon hose level gauges were installed in containment as a backup means of indication to the permanently installed RCS level gage and level transmitter. The inspectors performed a walkdown of the temporary level gauges and verified the system was properly aligned, filled, and vented. The draining evolution was performed in a deliberate and cautious manner without complication. The period in which the RCS was partially drained was held to a minimum as the licensee filled the refueling cavity following the reactor head removal. During core offload, the inspectors noted that all fuel movement activities were properly controlled by the refueling Senior Reactor Operator (SRO). Continuous communication was maintained between refueling personnel in containment, the spent fuel building, and the control room as the status of each fuel assembly was altered. In general, refueling activities were performed in accordance with detailed procedures, in a controlled manner, by knowledgeable personnel.

2.4 Air Discovered Leaking from Temporary Containment Penetration

Between March 19 and March 22, 1992, the spare Unit 2 containment penetrations 8D and 11E were opened up so that temporary cables could be run into the containment building. Steam generator eddy current cables were run through penetration 8D. Video cables and a 480V power cable were run through penetration 11E. The penetrations were then sealed at one end with fire retardant fiber and tape. This work was completed on March 22 under work request .325. This maintenance activity and a discussion of the inappropriateness of these seals is presented in Section 4.3.

On March 23, the containment integrity surveillance was completed. Core alterations began at 2005 on March 23 with commencement of the reactor head lift. Fuel offload began at 1004 on March 25 and was completed at 1358 on March 27. The cables were removed from penetration 8D and its normally installed blind flange was replaced around April 7. The containment integrity surveillance was completed again on April 8. Refueling began at 2050 on April 8. At 0356 on April 9, the backshift planning coordinator discovered air leaking from penetration 11E and notified the control room.

Refueling operations were stopped shortly after the control room was notified. Additional initial corrective actions consisted of removing the temporary seal and cables, replacing the normally installed blind flange, and doing a type B leak test on penetrations 8D and 11E. There were no other temporary containment penetration seals. Refueling then resumed and was completed on April 11. This event was properly reported per 10 CFR 50.72. The licensee is studying this event to determine long-term corrective actions.

These penetrations are located in the east cable vault. Ventilation from this area is exhausted through the supplementary leak collection and release system. Exhaust from this system is continuously monitored by radiation monitors HVS-RQ101A and RQ101B. Normal background levels were indicated by these radiation monitors during the refueling operations. During the refueling operations on April 8 and April 9, this exhaust was unfiltered. Operator action would have been required to switch to a filtered exhaust. The inspectors concluded that this event, as it occurred, was of minor safety significance. However, this event created the potential for an unfiltered release from a spent fuel handling accident.

3.0 RADIOLOGICAL CONTROLS (IP 71707)

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.

On March 26, a containment airlock door operator noticed that his dosimetry indicated that he had received a small unexpected exposure of 10 mRem. Health physics personnel investigated and discovered dose rates of 200 mRem on contact from a bucket being used as a seat outside the airlock. The licensee promptly removed the bucket and began an investigation. Subsequent dose assessments determined that a number of airlock operators received unnecessary exposures from the bucket but none received an over exposure. An NRC health physics inspector followed up on this event and also inspected other occupational exposure controls and the radiation protection program. That inspection is reported in inspection report 50-334/92-06; 50-412/92-03.

4.0 MAINTENANCE AND SURVEILLANCE (IP 61726, 62703, 71707)

4.1 Maintenance Observation

The inspectors 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;
- 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; and
- equipment was properly tested and returned to service.

Maintenance activities reviewed included:

MWR 907316	2SWS-MOV107A Replacement
MWR 008574	Sequencer Relay 862-EGSBA Testing
MWR 008576	Sequencer Relay 762-EGSBA Testing
MWR 008902	Analog Rod Position Instrument (Rod H2) Calibration

There were no notable observations.

4.2 Surveillance Observations

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:

OST 1.2.1	Nuclear Instrument Power Range Functional Test
OST 2.49.3	Refueling Operations Prerequisites
OST 2.47.3	Containment Integrity Checklist for Refueling
2BVT 1.11.3	SI Accumulator Discharge Check Valves Full Stroke Test
2BVT 1.4.7.4	Containment Electrical Penetrations Type B Leak Test
2BVT 1.4.7.7	Containment Isolation Valve Leakage Test Connection Verification

Beaver Valley Test (BVT) 1.11.3 is performed every refueling outage as the test results in the blowdown of the safety injection accumulators into the reactor coolant system (RCS). The inspectors noted good control and coordination of the infrequently performed surveillance by testing and operations personnel. Adequate precautions were taken to prevent the potential injection of nitrogen from the accumulators in the RCS. All accumulator discharge check valves achieved full-open stroke as determined by accumulator flow rates.

There were no other notable observations.

4.3 Inadequate Temporary Containment Penetration Installation

As described in Section 2.4, temporary cables were run through spare Unit 2 containment penetrations 8D and 11E to support refueling outage work. The penetrations were then sealed at one end with fire retardant fiber and tape. This work was completed on March 22, 1991, under ork request 1825. During refueling on April 9, the backshift planning coordinator discovered air leaking from penetration 11E.

The installation procedure for the temporary cables and penetration seals, procedure 2CMP-75-SG, "Cable Install-1E," Issue 4, Revision 2, was reviewed and approved by the Onsite Safety Committee (OSC). Step VII B2 states, "Install temporary fire stops and seals per PIP M16.3 at the containment penetrations." Plant installation process standard PIP M16.3, "Fire Stops and Seals," specifies an injected foam or rubber type 3-hour rated fire barrier certified to -1/4 inch water gauge differential air pressure and 2 inches hydrostatic pressure for these penetration seals. Such a seal would probably not have leaked. The licensee does not have in-house capability to install the type seals described in PIP M16.3 and a vendor had not been retained for this work. Everyone involved in the installation, acceptance, and surveillance of the seals for penetrations 8D and 11E focussed on the words, "temporary fire stops" rather than "per PIP M16.3." The actual seal installed was a temporary fire seal as described in VI C5 of Section 4.23 of the site maintenance manual. Those involved were also following past practices since a temporary fire seal was used to seal these penetrations in previous outages. Those involved include the installers and supervisor, a fire protection engineer who accepted the seal, a QC inspector who observed its installation, and a test engineer who did the containment integrity surveillance. Based on the above, the inspectors considered the root cause of the air leak to be the lack of installation details in the installation procedure 2 CMP-75-SG.

Even if the seals had been installed per PIP M16.3, they would not be in strict compliance with Technical Specification 3.9.4, since technical specifications only permit the use of blind flanges or valves to establish containment integrity. The use of the temporary fire stops installed from March 22 to April 9 to establish containment integrity during core alterations is an apparent violation of Technical Specification 3.9.4.

4.4 Missed Surveillance

On April 2, 1992, a quality assurance audit determined that a Unit 1 technical specification surveillance had not been performed. Technical Specification 4.7.4.1.b requires that each reactor plant river water (RPRW) valve servicing safety-related equipment not locked, sealed, or otherwise secured in its position, be verified in its correct position. Operational surveillance test (OST) 1.30-13B, "Reactor Plant River Water System B-Header Valve Position Verification," Revision 1, is performed at least every 31 days to meet the technical specification requirement. However, RW-645, reactor plant component cooling (CCR) heat exchanger outlet to blowdown, was deleted from OST 1.30-13B on July 15, 1991. The licensee immediately verified RW-645 to be in its correct open position and revised OST

1.30-13B to include the aforementioned valve. RW-645 was originally a normally locked open valve; thus its position was not required to be verified per OST 1.30-13B and was accordingly deleted. However, at an indeterminate date, the lock was removed and the normal valve position was changed to "open" vice "locked open" without updating OST 1.30-13B. The inspectors concluded that the failure to verify the position of RW-645 to be of minor safety significance. The river water flow through the CCR heat exchanger, as well as CCR system temperature, was maintained throughout the period in question. The failure to verify the position of RW-645 in accordance with Technical Specification 4.7.4.1.b is a violation; however, the violation is not being cited because the licensee's efforts in identifying and correcting the violation meet the criteria specified in Section VII.B of the Enforcement Policy.

4.5 Steam Generator Eddy Current Examinations

The Unit 2 steam generator examinations consisted of bobbin coil examination of 100 percent of the system generator tubes. Distorted signals were further examined with a rotating pancake probe. Three tubes were plugged due to indications greater than allowable. Eight tubes were plugged due to denting. Eleven tubes are now plugged in the A steam generator, seven tubes in the B steam generator, and fifteen tubes in the C steam generator. There are 3,378 tubes in each generator. The total number of tubes plugged is very low and indicates good steam generator performance.

5.0 SECURITY (IP 71707)

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 maintained and persons in Vital Areas were 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.

6.0 ENGINEERING AND TECHNICAL SUPPORT (IP 37700, 37828, 71707)

6.1 Unit 1 Missed Inservice Inspection (ISI)

On October 8, 1991, the licensee was issued a Notice of Violation (50-334/91-14-03), in part, for failing to take prompt and adequate corrective action for a quality assurance auditor's finding that identified a longitudinal pipe weld on the low-head safety injection system not included in the ISI program. In response, the licensee initiated a self assessment of the ISI program in order to prevent recurrence. This self assessment subsequently identified that various Class 3 equipment supports, including the auxiliary feedwater, quench spray, and river water pumps, were not examined during the first 10-year inspection interval as required by ASME Section XI. Article IWD 2600 (c) requires, in part, that supports and hangers for components whose structural integrity is relied upon to withstand design loads when the system function is required, be visually examined to detect any loss of support capability.

The licensee initiated immediate corrective action and performed the required visual inspection of all accessible equipment supports that were excluded from the exam schedule. Additionally, the inspectors conducted a walkdown of the component supports in question. No unacceptable conditions were identified. The river water pumps (WR-P-1A, 1B, 1C) seismic support rings were not inspected at this time as these supports are normally submerged. However, the structural integrity of these components was previously verified by the licensee during post maintenance and engineering activities. The inspectors reviewed MWR 003824, 004265, and 883145 which detailed the previous work performed on the river water pumps. Specifically, the seismic support rings were rebuilt and subsequently inspected in October 1991 for pump 1A, November 1991 for pump 1B, and September 1988 for pump 1C. The inspectors concluded these inspections met the requirements of ASME Section XI and that these river water pump supports are structurally sound. In addition, the licensee has committed to performing the visual inspections when the river water intake bays are drained for scheduled maintenance.

The licensee's ISI self assessment is continuing. All equipment supports that were excluded from the exam schedule have been identified. The inspection deficiencies were promptly corrected within 24 hours. The inspectors considered the findings by the licensee to be indicative of a detailed and critical self assessment.

6.2 Unit 2 Diesel Generator Sequencer Relay Failures

The emergency diesel generators (EDG) are designed to start automatically upon receipt of a safety injection signal (SIS) or emergency bus undervoltage signal. In order for the diesel generator output breaker to auto-close onto the emergency bus, an undervoltage condition must exist. Once the diesel output breaker shuts, a sequencer energizes to automatically start various loads in specific steps. A six step loading sequence is utilized to permit inrush currents to subside prior to starting the next block of loads. During the performance of routine testing on March 30, 1992, per 1/2 RCP-30, "Calibration of Timing Relays, Type ATC," three relays in the 2-1 EDG sequencer were found to be inoperable. The 2-2 diesel sequencer is of identical design and the licensee conservatively declared the same relays also inoperable. The sequencer degradation was reported by the licensee via the emergency notification system per 10 CFR 50.72.

The sequencer components are solid state relays manufactured by Automatic Timing and Controls Company (ATC), model number 365A. The relays in question for the 2-1 sequencer are designated 162-EGSAAX1, 762-EGSAA, and 862EGSAA. Sequencer relay 162-EGSAAX1 normally functions to permit starting of auxiliary feedwater pump 2FWE-P-23A and quench spray pump 2QSS-P-21A during step four of the diesel sequencing. Step four occurs between 15 and 17 seconds after the diesel output breaker auto-closes. This start signal is removed after 2 seconds. In order for the auxiliary feedwater pump to start, an SIS, or containment isolation signal phase B (CIB), or pump auto-start signal (i.e., low-low steam generator level) must be present coincident with the sequencer start signal. In order for the quench spray pump to start, a CIB signal must be present coincident with the sequencer start signal. The failure of relay 162-EGSAAX1 results in the auxiliary feedwater pump and quench spray pump not auto-starting during step four of the start sequence. Sequencing relays 762-EGSAA and 862-EGSAA function to reset the sequencer to step zero following an SIS or CIB. The load sequence is terminated then reinitiated from step one if an SIS or CIB were to occur during the load sequence for loss of onsite power. Any load which has been loaded will remain running; however, recycling the sequencer assures loading for the event is completed in the proper order. The failure of the reset relays results in the load sequencer continuing through to the end of its cycle (step six) instead of being reset to step one upon the SIS or CIB. The combined effect of the three relay failures in both the 2-1 and 2-2 EDG sequencers is that both train A and B auxiliary feedwater and quench spray pumps would not automatically start, if called upon, during step four of the sequencing. However, a seal-in feature exists in both sequencers which will permit these components to auto-start when the sequencer times out at 60 seconds.

The resultant effect of the 45 second delay in starting both auxiliary feedwater and quench spray during postulated accident conditions is currently under analysis by the licensee. There were no immediate safety concerns due to the relay failures as the plant was in a refueling outage with all fuel removed from the reactor vessel. The functions normally provided by the sequencer relays were not required for the plant conditions at the time of the discovery. However, a strong possibility exists that the relay failures occurred at an indeterminate date during power operations. An evaluation of the consequences of the postulated relay failures at power is to be addressed by the licensee's safety analysis.

On April 8, 1992, the inspectors observed the as found testing of the identical sequencer relays for EDG 2-2. Relays 162-EGSBAX1 and 762-EGSBA both failed to operate while relay 862-EGSBA operated intermittently. Each relay has two separate internal circuits; a clock circuit and a power supply circuit. Visual inspection of the relay identified a 2 watt carbon resistor for the internal clock circuit as the suspect subpart (thermal degradation of the internal resistor was indicated). These circuits are both designed to be operated on 24 Vdc. Under higher voltage applications, as in the sequencer (125 Vdc nominal, 130 Vdc actual), an external 20 watt voltage compensating resistor is installed in series to reduce the applied voltage to the design voltage. In this design configuration, the relay is normally deenergized with the two internal circuits jumpered together. When energized, the voltage drop across the relay would be approximately 26 Vdc. Due to the jumper, this would be the voltage across both the clock circuit and power supply circuit. However, the six failed relays were installed such that the jumper was removed and the clock circuit was continuously energized. This new installation configuration was per the vendor's verbal recommendation to the licensee in order to further improve the timing accuracy of the relays. However, this installed configuration resulted in a lower than design voltage drop across the external dropping resistor and excessive operating voltage across the clock circuitry. The as-found voltage across the clock circuit for several of the failed relays was subsequently measured between 112.8 - 129 Vdc. Excessive operating voltage across the clock circuit may have caused the internal resistor to have exceeded its 2 watt power rating. The cause of the internal resistor failure in this configuration is still under investigation by the licensee. All failed relays have been replaced with the same ATC 365A model relays under the normally deenergized configuration. Initial licensee testing has verified sequencer operability. EDG automatic testing (2OST - 36.3, 36.4), which simulates a loss of offsite power in conjunction with an SIS, had not yet been performed by the end of the inspection period. The licensee's initial corrective action thus far appears adequate.

The inspectors performed a review of the Design Change Package (DCP) 1545 which replaced the existing ATC 305E electromechanical time delay relays with ATC 365A solid state relays in September 1990. The inspectors had certain observations as follows:

- The 365A relays were commercial grade items and dedicated as Quality Assurance (QA) Category I in accordance with IEEE 323-1974, "IEEE Standard for Qualifying Class IE Equipment for Nuclear Power Generating Stations." Review of the dedication activities performed by Wyle Laboratories, per the licensee's procurement specification, indicates the baseline functional testing was conducted in the original, normally deenergized, installation configuration. In this configuration, with 125 Vdc applied to the relay, approximately 26 Vdc was measured across the relay with the remainder across the external dropping resistor.

- The installation specification specified the wiring of the six relays to be terminated per wiring diagrams 10080-RE-3EQ and 3EX. Review of these diagrams indicated the installation configuration to be such that the clock circuit is continuously energized. No documentation was provided by the vendor to support this change in configuration as the specifics of the relay circuit were considered proprietary information.

- The post modification testing by the licensee was adequ... to verify sequencer operability. Review of the proof and functional testing conducted in September through November 1990 revealed the sequencer relays operated satisfactorily as installed. This in turn suggests the relays had not yet failed.

IEEE 323, Section 6.3.1.3, states, "equipment shall be connected in a manner that simulates its expected installation when in actual use unless an analysis can be performed and justified to show that the equipment's performance would not be altered by other means of connection." However, as previously identified, the qualification installation differed from the actual installation and no analysis was performed to justify the change in configuration. 10 CFR 50, Appendix B, Criterion III, requires in part that measures be established for the selection and review of suitability of application of parts that are essential to the safety related functions of systems and components. The installation of ATC 365A relays for EDG 2-1 and 2-2 sequencers under an unsuitable configuration for its intended application is an apparent violation.

7.0 SAFETY ASSESSMENT AND QUALITY VERIFICATION (IP 40500, 71707, 90712, 91700)

7.1 Review of Written Reports

The inspectors reviewed LERs and other reports submitted to the NRC to verify that the details of the events were clearly reported, including accuracy of the description of cause and adequacy of corrective action. The inspectors 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:

92-02 ESF Actuation - Blowdown Isolation due to Pressure Switch Failure

92-03 ESF Actuation - Inadvertent Closure of Containment Isolation Valves TV-SS-117 A, B, and C

Unit 2:

92-01 Failure to Log Axial Flux Difference

These events were reviewed in inspection report 92-05/04. The inspectors had no additional comments regarding these events.

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, re[illegible] determinations, and corrective actions.

7.2 Alternate Fuel Pool Cooling Modificat[illegible]

The inspectors observed the implementation of temporary modification 2-92-008 to the Unit 2 service water (SW) system. Specifically, the fire protection system was utilized as a source of temporary cooling, in place of service water, to reactor plant component cooling water (CCP) heat exchanger 21B which in turn provides for spent fuel pool cooling. This modification was necessary to allow both trains of SW to be removed from service to accomplish repair of cross-connect valves 2SWS-MOV107A through D. Due to excessive leak-by of the valves, the SW trains could not be isolated from each other to perform individual train related work.

The implementation of the modification was designated an "Infrequently Performed Test and Evolution" (IPTE) per Nuclear Group Administrative Procedure 6.23. This designation resulted in a high degree of management involvement in the planning and control of the modification activities. Licensee management exhibited a proper safety perspective during the modification process and provided reasonable assurance that plant safety would not be compromised. Management ensured that: (1) detailed safety assessments were performed; (2) contingency plans were formalized and in place; (3) personnel were properly briefed as to management's expectations; (4) the temporary modification was fully tested prior to removing both trains of SW from service; and (5) staging and equipment were ready to support the modification. The service water headers were secured for approximately 6 hours and the fire protection system was able to maintain fuel pool temperature less than 90° F. Overall, the inspectors considered the licensee's efforts to formalize the responsibilities and requirements of evolutions that meet the IPTE criteria to be an excellent initiative.

8.0 STATUS OF PREVIOUS INSPECTION FINDINGS (IP 71707, 90702, 92701)

The NRC Outstanding Items List was reviewed with cognizant licensee personnel. Items selected by the inspectors 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.

8.1 (Closed) Unresolved Item (50-412/90-20-01): This unresolved item involved an overpressure event which occurred on the C loop of the Unit 2 reactor coolant system (RCS). During this event, the pressure in the isolated C loop exceeded the limits of Technical

Specification (TS) 3.4.9.1. The technical specifications require in part that if RCS temperature and/or pressure limits are exceeded, then the licensee is required to perform an analysis to determine the effects of the out-of-limit condition on the fracture toughness properties of the RCS. The limiting component for this TS is the reactor vessel. However, the inspectors identified a lack of specificity in the technical specifications regarding pressure/temperature limits applicable to isolated reactor coolant loops which may be at a substantially different pressure and/or temperature from the reactor vessel.

In response to the above concern, the licensee submitted a proposed change to Technical Specification Bases 3/4 4.9. The change involves the addition of pressure/temperature limits for a loop isolated from the reactor vessel. The proposed limits were reviewed by the NRC staff and found to be acceptable. The staff concluded that the pressure/temperature limits are conservative and satisfy the requirements of Standard Review Plan 5.3.2, "Pressure/Temperature Limits," and Regulatory Guide 1.99, "Radiation Embrittlement of Reactor Vessel Materials." Accordingly, the change to Technical Specification Basis 3/4 4.9 was issued on March 2, 1991. Additionally, the inspectors performed a review of the licensee's fracture toughness effects analysis. The maximum loop C pressure was calculated to be 890 psig. This value is within the allowable pressure limits at identified temperatures for an isolated loop.

The inspectors had no further questions. This item is closed.

9.0 EXIT MEETING

9.1 Preliminary Inspection Findings Exit

Meetings were held with senior facility management throughout the inspection to discuss the inspection scope and findings. A summary of the findings was further discussed with the licensee at the conclusion of the report period on April 28, 1992.

9.2 Attendance at Exit Meetings Conducted by Region-Based Inspectors

Dates	Subject	Inspection Report No.	Reporting Inspector
3/16-27/92	Engineering	50-334/92-07; 50-412/92-05	Woodard
3/16-19/92	Physical Security	50-334/92-08; 50-412/92-06	Smith
4/13-17/92	Effluents	50-334/92-10, 50-412/92-09	Jang
4/13-17/92	Health Physics	50-334/92-06; 50-412/92-03	Noggle