U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Report No. 50-440/90022(DRP)

Docket No. 50-440 License No. NPF-58

Licensee: Cleveland Electric Illuminating Company
Post Office Box 5000
Cleveland, OH 44101

Facility Name: Perry Nuclear Power Plant

Inspection At: Perry Site, Perry, Ohio

Inspection Conducted: November 17, 1990, through January 7, 1991

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Reactor Projects Section 3B

1/24/91
Date

Inspection Summary

Inspection on November 17, 1990, through January 7, 1991 (Report No. 50-440/90022(DRP))

Areas Inspected: Routine unannounced safety inspection by resident and regional inspectors of reactor startup from a refueling outage; monthly surveillance observations; monthly maintenance observations; operational safety verification; evaluation of licensee self-assessment capability; engineered safety feature walkdown; onsite followup of events; and a plant status meeting.
Results: Of the eight areas inspected, one violation was identified in the area of operational safety verification (Paragraph 4.b.). That violation concerned the failure to maintain the reactor coolant temperature within the range specified while in Operational Condition 5. The violation was receiving appropriate licensee management attention at the close of the report period.

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For this report period, the area of plant operations was considered adequate based on the inspectors' observations of plant evolutions and response to events; however, licensee management needed to increase efforts to reduce personnel errors. The area of maintenance and surveillance was considered a weakness due to failure of licensee personnel to perform an intended adjustment to main steam isolation valves (MSIV) which contributed to the failure of three MSIVs to slow-close during reactor startup evolutions.

In general, the inspectors found the areas of security and emergency preparedness to be a strength, based on routine observations. The area of radiological controls was considered adequate; however, continued licensee management attention appears warranted to improve housekeeping in general and improve radiological practices at entries to contaminated areas. The inspectors noted that senior licensee management personnel were addressing the concerns in this area.

DETAILS

1. Persons Contacted

   a. Cleveland Electric Illuminating Company (CEI)

   #M. Lyster, Vice President, Nuclear-Perry
   *#R. Stratman, General Manager, Perry Nuclear Power Plant (PNPP)
   M. Gmyrek, Operations Manager (PNPP)
   #M. Cohen, Manager, Maintenance Department (PNPP)
   #V. Higaki, Manager, Outage Planning Section (PNPP)
   #D. Cobb, Operations Superintendent (PNPP)
   #S. Kensicki, Director, Perry Nuclear Engineering Department (PNED)
   *#V. Concel, Manager, Technical Section, (PNED)
   *#F. Stead, Director, Perry Nuclear Support Department (PNSD)
   *#H. Hegrat, Compliance Engineer (PNSD)
   *#R. Newkirk, Manager, Licensing and Compliance Section (PNSD)
   *#E. Riley, Director, Perry Nuclear Assurance Department (PNAD)
   * W. Coleman, Manager, Perry Nuclear Assurance Department (PNAD)
   * K. Russell, Shift Supervisor, Perry Nuclear Power Plant (PNPP)
   * W. Wright, Acting Manager, Instrumentation and Controls Section (PNPP)
   * J. Eppich, Manager, Mechanical Design Section (PNPP)

   b. U. S. Nuclear Regulatory Commission

   #C. Paperiello, Deputy Regional Administrator, RIII
   #R. Knop, Branch Chief, DRP3, RIII
   *#R. Lanksbury, Section Chief, DRP3B, RIII
   #J. Hannon, Director, Project Directorate III-3, NRR
   #R. Hall, Project Manager, NRR
   * P. Hiland, Senior Resident Inspector, RIII
   * G. O'Dwyer, Resident Inspector, RIII
   * R. Roton, Reactor Engineer, RIII

   * Denotes those attending the exit meeting held on January 7, 1991.

   # Denotes those attending the Plant Status meeting on November 28, 1990.

2. Monthly Surveillance Observation (61726)

   For the below listed surveillance activities the inspectors verified one or more of the following: testing was performed in accordance with procedures; test instrumentation was calibrated; limiting conditions for operation were met; removal and restoration of the affected components were properly accomplished; test results conformed with technical specifications, procedure requirements were reviewed by personnel other than the individual directing the test; and any deficiencies identified during the testing were properly reviewed and resolved by appropriate management personnel.

Surveillance Test No.	Activity
SVI-E31-T1405-A	MSL High Flow Channel A Response Time for 1E31-N086A, 1E31-N088A
SVI-D17-T0040A-D	Main Steam Line Radiation Monitor Channel D Functional for 1D17-K610D
SVI-C51-T0027A	APRM A Trips Channel Functional

No Violations or Deviations were identified.

3. Monthly Maintenance Observation (62703)

Station maintenance activities of safety-related systems and components listed below were observed and/or reviewed to ascertain that activities were conducted in accordance with approved procedures, regulatory guides, 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 and fire prevention controls were implemented.

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

The following specific maintenance activities were observed:

W. O.	Subject
89-7179	Calibration of Division I Diesel Generator day tank level switches 1R45-N0120A, 1R45-N0140A, and 1R45N0150A.
90-5697	Functional test of Local Power Range Monitor (LPRM) connection insulation resistance for LPRMs 5A-40-25 and 5B-48-25.
90-4066	Addition of about 4 gallons of lubricating oil to reactor recirculation pump "A".
90-6193	Repair of reactor recirculation pump breaker "4B".

No Violations or Deviations were identified.

4. Operational Safety Verification (71707)

a. General

The inspectors observed control room operations, reviewed applicable logs, and conducted discussions with control room operators during this inspection period. The inspectors verified the operability of selected emergency systems, reviewed tagout records, and verified tracking of Limiting Conditions for Operation associated with affected components. Tours of the intermediate, auxiliary, reactor, and turbine buildings 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 certain pieces of equipment in need of maintenance. The inspectors by observation and direct interview verified that the physical security plan was being implemented in accordance with the station security plan.

The inspectors observed plant housekeeping, general plant cleanliness conditions, and verified implementation of radiation protection controls.

b. Inadvertent Heatup of Reactor Coolant Temperature

On November 19, 1990, at about 4:00 p.m., while the plant was in Operational Condition 5 (refueling), on-coming operations personnel found that reactor coolant temperature had increased to 121 degrees F during the previous shift (8:00 a.m. to 4:00 p.m.) which was outside the 75 to 85 degree F range specified by Section 5.7, of System Operating Instruction (SOI)-E12, "Residual Heat Removal System (Unit 1)." During the day shift, with residual heat removal (RHR) loop "B" in the shutdown cooling mode, licensed plant operators (ROs) bypassed shutdown cooling flow around the "B" RHR heat exchanger (HX), effectively stopping the removal of decay heat from the core. The operators were not aware of the increase in the reactor coolant temperature, as they had been monitoring coolant temperature on indicators that were not representative of temperature in the reactor pressure vessel (RPV). The operators were using the RHR inlet to HX "B" and RPV bottom head drain temperatures for indication of reactor coolant temperature. Since there was no flow through the RHR "B" heat exchanger, the RHR inlet to HX "B" temperature indication was not representative. The RPV bottom head drain temperature indication was not representative of the temperature in the core due to the normal cooling water flow into the core through the control rod drive mechanisms and thermal stratification in the bottom of the RPV.

Discussions with licensee management indicated that at about 12:00 p.m. (EST), on November 19, 1990, the operators on duty discovered that the "B" reactor recirculation loop temperature (displayed on a back panel strip chart) and the RPV temperature displayed on the emergency response information system (ERIS) indicated about 100 degrees F. These readings were considered erroneous by the on-shift operators and were not discussed with on-shift supervision. After finding that the reactor coolant temperature was about 121 degrees F, the on-coming operators re-established shutdown cooling flow through the RHR "B" heat exchanger and the reactor coolant temperature was returned to about 80 degrees F within 30 minutes.

Technical Specification 6.8.1.a. required that written procedures be established, implemented, and maintained as recommended in Appendix A of Regulatory Guide 1.33, Revision 2, February 1978. Section 4.e of Appendix A of Regulatory Guide 1.33 recommended procedures for the operation of "Shutdown Cooling." Failure of the licensee to maintain reactor coolant temperature within the specified range while in Operational Condition 5 is a violation of Technical Specification 6.8.1.a. (440/90022-01(DRP)).

c. Scram Solenoid Pilot Valves 10 CFR 21 Notification

On December 11, 1990, the licensee reported to NRC Region III that contrary to design criteria, six of forty-one installed scram solenoid pilot valves (SSPV) had failed. The licensee sent a written report to the Director, Office of Nuclear Reactor Regulation on December 14, 1990, (letter: PY-CEI/NRR-1281L). The SSPV was used in the control rod drive system to cause the insertion of control rods into the reactor core. By design, the SSPV changed position upon de-energization, which permitted the venting of air pressure from the scram valve actuators. This in turn opened the scram valves which exerts a differential pressure across the control rod drive mechanism causing control rod insertion. The following data was provided regarding the valves:

Model EP-139
GE Part No. 922D138P001
ASCO Part No. HV 176-816-1
Perry Stock Code 1579947
Perry was furnished 70 SSPV's on P.O. S-121462
  Material Receipt MR-96472, Lot No. F61191A.

Only forty-one of the seventy valves received at Perry were installed during the then ongoing refueling outage. All valves from the identified lot were removed and replaced. Efforts to determine the cause of the failures included the return of four of the six failed valves to GE/ASCO for analysis. Independently, two of the six failed valves were also evaluated by a laboratory contracted by the licensee. In both cases, no conclusive evidence was found that identified the probable cause of the malfunction. In all cases, the failures could not be duplicated in the laboratory. Concurrent with the licensee's investigation, it was determined that the Hope Creek Station received thirty valves from the same lot (F61191A). The Hope Creek Station was notified by Perry of this situation. This accounted for all valves in this lot, for a total of one hundred. The licensee submitted a written 10 CFR 21 report.

d. Inadvertent Containment Evacuation Alarm

On December 19, 1990, at about 9:30 a.m. (EST), while the plant was in cold shutdown, the Containment Evacuation Alarm was inadvertently initiated during the performance of Periodic Test Instruction (PTI)-D17-P1680, "Containment Atmosphere Radiation Monitor 1D17-K680 Calibration," because instrumentation and control (I&C) personnel lifted the "panel-side" wires off of terminal AA-8 in Panel 866 instead of the "field-side" wires as delineated in the procedure. The containment evacuation alarm sounded when the gaseous channel trip test pushbutton was depressed as directed by step 5.3.3 of the PTI and operations personnel entered Off Normal Instruction (ONI)-D17, Revision 4, "High Radiation Levels within Plant (Unit 1)." Twelve people were evacuated from containment and Health Physics performed surveys that indicated no unexpected levels of radiation existed in containment. ONI-D17 was exited and normal access to containment was restored at about 10:00 a.m. (EST).

e. Drywell Closeout

The licensee conducted "closeout" inspections of the drywell in accordance with Integrated Operating Instruction (IOI)-1, "Reactor Startup from Cold Shutdown," Attachment 2, Revision 5, in preparation for reactor startup after the second refueling outage. The inspectors accompanied a supervising operator (a licensed reactor operator) and two quality assurance representatives on two of the closeout inspections.

During the inspection on December 19, 1990, several deficiencies were identified including: small amounts of debris; unremoved scaffolding, lead shielding, air samplers, and temporary sensors on reactor recirculation pumps; floor drain sump covers were missing; a loose thermocouple; and a broken ground cable.

During the inspection on December 20, several more deficiencies were identified including nuts missing on the drywell personnel airlock door mechanical interlocks and loose bolts on three of the four inboard Main Steam Isolation Valve (MSIV) stem leak detection sightglass covers. Work orders were written as necessary. The inspectors noted that the individuals were conscientious in conducting the inspections. The licensee resolved all of the identified deficiencies prior to reactor startup.

The inspectors noted that additional management attention was focused on the closeout inspections; however, licensee management should continue to emphasize post-maintenance restoration activities which were the cause of many of the identified deficiencies.

f. Licensee Use of Overtime

The inspectors reviewed licensee practices and programs for the use of overtime by departments other than operations personnel. The use of overtime for the operations personnel had been inspected in Inspection Reports No. 440/89022(DRP) dated September 11, 1989, and

No. 440/88012(DRP) dated September 14, 1988. Perry Administrative Procedure (PAP)-110, Revision 3, "Shift Staffing and Overtime," provided the manning and overtime requirements for all modes of operation. That procedure applied to personnel who were involved in the "hands on" performance of safety-related functions. That procedure was not applicable to system engineers and other personnel not physically involved with safety-related functions. The licensee specified that the PAP-110 guidelines were used in providing overtime criteria for system engineers even though not required by procedures. The inspectors discussed with licensee management a recent event at another Region III facility where an engineer in charge of testing, and in his twenty-fourth consecutive hour of work, gave misdirection that resulted in reactor coolant system water being sprayed through an open vent valve. Consequently, the inspectors emphasized the importance of monitoring overtime for all licensee staff involved in directing and overseeing safety-related activities and the need to adhere to the PAP-110 guidelines. The inspectors reviewed the employee overtime and absence records (Form TK-5) for the radiation protection, instrumentation and control, engineering, and maintenance sections with particular emphasis on the time period of the 1990 refueling outage. The inspectors found instances where individuals apparently exceeded the PAP-110 administrative guidelines for total hours worked; however, the respective section supervisors provided overtime guideline information which showed that the employee overtime and absence hours were actually within the licensee policy guidelines. The information included: allotted time for lunch breaks during overtime; time allotted for shift turnover; and the work activity performed was not considered safety-related. The licensee demonstrated that overtime deviation requests had been approved when an individual's hours of work exceeded the administrative limits.

The inspectors interviewed section managers and the respective record keepers about the reasons plant personnel had worked overtime and why no discrepancies were identified. The inspectors were informed that on October 1, 1990, the Maintenance Section implemented an informal method of tracking regular and overtime hours of each individual to check hours prior to making overtime assignments. In order to avoid exceeding the guidelines of PAP-110, maintenance personnel hours of work were tracked and overtime deviation requests written regardless of whether the maintenance activity was categorized as safety-related or not.

No Deviations were identified; however, one Violation was identified.

5. <u>Engineered Safety Feature (ESF) Walkdown</u> (71710)

During this inspection period, the inspectors performed a detailed walkdown of the accessible portions of train "B" of the emergency closed cooling (ECC) system. The system walkdown was conducted using Valve Lineup Instruction (VLI)-P42, and the controlled piping and instrumentation diagrams (P&IDs) for the ECC system.

During the walkdown, the licensee identified the "B" train as operable. The inspectors took into account that during the walkdown the "B" train was in various modes of operation and therefore in various valve lineups.

During the system walkdown, the inspectors directly observed equipment conditions to verify that hangers and supports were made up properly; appropriate levels of cleanliness were being maintained; piping insulation, heaters, and air circulation systems were installed and operational; valves in the system were installed in accordance with applicable P&IDs and did not exhibit gross packing leakage, bent stems, missing handwheels, or improper labeling; and major system components were properly labeled and exhibited no leakage. The inspectors verified that instrumentation associated with the system was properly installed, functioning, and that significant process parameter values were consistent with normal expected values. By direct visual observation or observation of remote position indication, the inspectors verified that valves in the system flow path were in the correct positions as required by the various modes of operation that were required; power was available to the valves; valves required to be locked in position were locked; and pipe caps and blank flanges were installed as required.

No Violations or Deviations were identified.

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

a. General

The inspectors performed onsite followup activities for events which occurred during the inspection period. Followup inspection included one or more of the following: reviews of operating logs, procedures, and condition reports; direct observation of licensee actions; and interviews of licensee personnel. For each event, the inspectors reviewed one or more of the following: the sequence of actions; the functioning of safety systems required by plant conditions; licensee actions to verify consistency with plant procedures and license conditions; and verification of the nature of the event. Additionally, in some cases, the inspectors verified that licensee investigation had identified root causes of equipment malfunctions and/or personnel errors and were taking or had taken appropriate corrective actions. Details of the events and licensee corrective actions noted during the inspector's followup are provided in paragraph b. below.

b. Details

(1) Unexpected Residual Heat Removal "A" Shutdown Cooling System Isolation

On November 16, 1990, at about 10:30 p.m., while the reactor was in cold shutdown, an unexpected isolation of the residual heat removal (RHR) "A" shutdown cooling (SDC) system occurred. Control room operators promptly suspended performance of a work order in which instrumentation and control (I&C) technicians were replacing a control relay in the SDC isolation logic circuitry, removed the isolation signal and restarted the RHR "A" SDC system.

The licensee reported this event to the NRC Operations Center via the Emergency Notification System (ENS) about 1:00 a.m., on November 17; however, the preliminary root cause determination was incorrect. The actual cause for this event was identified to have been incorrect written instructions in the work order. The work order, incorrectly, directed that a jumper (that was preventing the SDC isolation) be removed before the isolation signal was removed.

Licensee Event Report (LER) 440/90032 was issued on December 14, 1990, detailing this event occurrence, root cause, and corrective actions taken to prevent recurrence. The inspectors will perform a followup review of that LER after completion of licensee corrective actions.

(2) Unplanned Initiation of Train B of the Control Room Emergency Ventilation System

On November 20, 1990, at about 2:45 a.m., while the reactor was in cold shutdown, an unplanned actuation of train "B" of the control room ventilation system in the emergency recirculation mode occurred while personnel were restoring temporary power. Train "A" was already running in the emergency recirculation mode. At about 4:00 a.m., the licensee reported this event as an unplanned engineered safety feature actuation in accordance with 10 CFR 50.72(b)2(ii). The inspectors will review the forthcoming LER.

(3) Combined Leakage Rate Greater Than 0.60 La

On November 22, 1990, at about 4:20 p.m., while the reactor was in cold shutdown, licensee personnel determined that the primary containment leakage rate exceeded the 0.60 La combined leakage rate limit specified in Technical Specification 3.6.1.2.b. This was identified when the 42-inch outboard containment isolation purge valve (M14-F40) of the containment vessel and drywell purge system was determined to leak 11,750 standard cubic centimeter per minute (sccm) which exceeded the 5,011 sccm limit specified in Technical Specification 4.6.1.8.4. This event was documented in licensee Condition Report (CR) 90407.

The licensee reported this event to the NRC Operations Center via the ENS at about 7:45 p.m. on November 22, 1990, in accordance with 10 CFR 50.72(b)2(i) and (iii). The inspectors will review the forthcoming LER during a future inspection period.

(4) Unusual Event Due to Loss of Offsite Communication

On December 4, 1990, at about 2:00 a.m., while the reactor was in cold shutdown for a refueling outage, the licensee declared an Unusual Event (in accordance with Emergency Plant Instruction (EPI A.1(I)1.1)) due to a loss of the offsite communication networks: the private branch exchange (PBX) and the off premise exchange (OPX). The licensee notified Alltel (local telephone company) regarding the PBX system and the system was restored at about 3:00 a.m. At about 4:00 a.m., the licensee exited the Unusual Event after re-establishing the plant's offsite communication capability. At about 6:00 p.m., the OPX system was restored to service. The root cause for that failure was determined to be microwave equipment misaligned due to inclement weather.

The licensee informed the NRC operations center of this event via the ENS at about 2:30 a.m. This event did not satisfy any of the criteria of 10 CFR 50.73. Therefore, no LER will be forthcoming and the inspectors have no concerns or questions about this event.

(5) Control Room Emergency Ventilation System Design Deficiency

On December 4, 1990, at about 4:00 p.m., while the reactor was in cold shutdown for a refueling outage, an evaluation (performed by the licensee's Architect Engineer) identified a design deficiency. That evaluation concluded that the control room emergency ventilation system (M26) could be rendered inoperable if a loss of coolant accident (LOCA) and a seismic event (SE) occurred concurrently. The evaluation indicated that a substantial loss of inventory from the safety-related chilled water system (P47, a necessary support system) could result from a guillotine break of a non-safety portion of the P47 system, specifically the piping at the non-safety cooling coils for the controlled access and miscellaneous equipment area ventilation system (M21) and the computer room ventilation system (M27). The evaluation determined that in about 12 seconds the inventory loss would cause the level in the P47 expansion tanks to drop low enough to cause an isolation signal to be sent to valves which would isolate the M21 and M27 systems in about 30 seconds. The evaluation found that enough inventory would be lost to render the chilled water system incapable of supporting the control room emergency ventilation system and that system was declared inoperable at 4:00 p.m. In cold shutdown without the M26 system, the Technical Specifications prohibit core alterations, moving of irradiated fuel in the fuel handling building and containment, operations with a potential for draining the core, and entering the Startup mode. The licensee planned to modify the M26 system and its operation so that it could be declared operable prior to plant restart. The inspectors will review the forthcoming LER for this event.

At about 7:30 p.m., on December 4, 1990, the licensee reported the inoperability of the control room emergency system as a condition that alone could have prevented the fulfillment of the safety function of a system needed to mitigate the consequences of an accident in accordance with 10 CFR 50.72b(2)(iii)d.

(6) Inadvertent Start of "B" RHR Pump During Surveillance

On December 9, 1990, at about 6:30 a.m., while the reactor was in cold shutdown, the "B" RHR pump unexpectedly received three auto-start signals. At the time of this occurrence, plant instrumentation and control (I&C) personnel were in the process of performing Surveillance Procedure (SVI)-E12-T5368, "ECCS/LPCI Pump B Start Time Delay Relay Channel Functional/ Calibration for 1E12A-F70B." The first auto-start of the "B" RHR pump occurred at 6:39 a.m. Upon receiving the pump start signal, plant operators made the determination that plant conditions were inappropriate for injection; therefore, the pump start signal was overridden. Approximately 34 seconds later, the second auto-start of the "B" RHR pump occurred while plant operators were investigating the first pump start. Again, the operators secured the "B" RHR pump and held the pump switch in the off position. Approximately one minute after the second auto-start of the "B" RHR pump, a third auto-start signal was received. However, due to the operator holding the pump switch in the off position, the pump did not start.

Subsequent investigation into the event revealed that during the performance of SVI-E12-T5368, instrumentation and control (I&C) personnel had opened link MMA-5 in lieu of the procedure specified link MMA-4. Since the incorrect link was in the open position, the pump starts occurred during the testing of relay 1E12A-F070B.

The licensee reported this event to the NRC Operation Center via the ENS at about 9:10 a.m., December 9, 1990, in accordance with 10 CFR 50.72(b)(2)ii. This condition was also documented in licensee's Condition Report 90-437. The inspectors will review LER 90037 during a future inspection period.

(7) Inadvertent Reactor Water Cleanup (RWCU) System Isolation

On December 18, 1990, at about 1:00 a.m., while the reactor was in cold shutdown, a Division 1 outboard containment isolation of the reactor water cleanup (RWCU) system occurred. Operators verified system valves isolated properly and suspended performance of Work Order 90-6164 which instrumentation and control (I&C) personnel were attempting to determine the cause of RWCU outboard isolation valves not opening when required by System Operating Instruction (SOI)-G33, "Reactor Water Cleanup System." The RWCU isolation occurred about the time that I&C

personnel were replacing a broken wire and two fuses blew, which apparently caused the RWCU isolation. I&C personnel determined that the wire that was being relugged was verified de-energized at both ends, no arcing was observed, and tools did not show signs of shorting to ground. I&C personnel replaced the fuses and operations personnel restored the RWCU system. Preliminary licensee investigation could not determine a root cause for the two blown fuses. The inspectors will review the forthcoming LER during a future inspection.

At about 4:00 a.m., on December 18, the licensee reported the event to the NRC operations center via ENS as an unplanned engineered safety feature actuation in accordance with 10 CFR 50.72(b)2(ii).

(8) High Pressure Core Spray (HPCS) System Inoperability

On December 28, 1990, at about 12:05 a.m., while the reactor vessel pressure was about 160 pounds per square inch gage (psig) during a reactor startup, the licensee declared the high pressure core spray (HPCS) system inoperable because a reactor water level instrument was indicating about 23 inches lower than the other reactor water level instruments. These instruments were designed to initiate a HPCS system auto-start signal if reactor water level dropped to a low level (level 2, which was 129 inches above top of active fuel). At 11:20 p.m., on December 27, operations personnel were decreasing reactor pressure when a half-scram occurred on a level-3 (177 inches above top of active fuel) signal from reactor protection system (RPS) channel "D" due to a trip on the "D" reactor water level trip unit. The other three reactor water level instruments indicated normal level; therefore, operations personnel declared the associated detector inoperable. Operations personnel found that the transmitters for the HPCS level-2 auto-start signal also indicated low; therefore, plant operators declared the transmitters and the HPCS system inoperable in accordance with Technical Specifications. Licensee personnel found the root valve for the reference leg of the "D" instruments was closed instead of being open as required. Operations personnel believed that the root valve had been either not fully closed, or its seat had been leaking by because the "D" water level instruments had tracked closely to the other three channels during the previous reactor pressure changes of reactor startup. Also, after this rapid pressure change had caused the "D" instrument readings to deflect significantly away from the other three channels, the "D" readings drifted back into agreement with the other three channels. The root valve was opened and the instruments and the HPCS system were declared operable at about 8:40 a.m. Preliminary investigation by the licensee revealed that the root valve had been checked and independently verified open on about December 13, 1990, by Valve Lineup Instruction (VLI)-B21, "Nuclear Steam Supply

System (Unit 1)," Revision 3. The licensee was continuing to investigate the cause for this event. The inspectors will review the forthcoming LER on this event in a future inspection period.

At about 3:00 a.m., on December 28, the licensee notified the NRC Operations Center of this event via the ENS as a condition that could have prevented a safety function needed to mitigate the consequences of an accident in accordance with 10 CFR 50.72(b)2(iii)d.

(9) Unplanned Reactor Water Cleanup System Isolation

On January 1, 1991, at about 7:30 p.m., while the reactor was in the Startup mode with all rods in, the reactor water cleanup (RWCU) system isolated on a "high differential flow" signal. The reactor operator had started the "A" RWCU pump in order to shift from using the "B" pump to provide RWCU system flow. When the reactor operator shutdown the "B" pump, system flow coasted down and the operator started the "A" pump but the "high differential flow" timer had timed out and the RWCU system isolated. Investigation found that the operator had failed to realize that the discharge valve for the "A" pump was white-tagged closed (which indicated valve operation by operations personnel only). The RWCU system was restored and it was found that the "A" pump had been deadheaded for about 160 seconds. The inspectors will review the forthcoming LER for this event.

The licensee notified the NRC operations center of this event via the ENS at about 11:00 p.m. on January 1, 1991.

(10) Unplanned Reactor Protection System Actuation

On January 1, 1991, at about 9:00 p.m., while the reactor was in the Startup mode with all rods fully inserted, an unplanned reactor protection system actuation signal was generated because the turbine stop valves (TSV's) went closed for unknown reasons. The reactor operators placed the reactor mode switch in "Shutdown" as required by Off-Normal Instruction (ONI)-C71-1, Revision 1, "Reactor Scram (Unit 1)," which placed the plant in cold shutdown. The next day, at about 10:00 a.m., licensee management concluded that during the simultaneous performance of two surveillance instructions, a reactor operator depressed the wrong pushbutton and inadvertently closed the TSV's. The reactor operator did not recall depressing the wrong pushbutton (the "close valves" pushbutton on the speed set control for the main turbine). Licensee management reviewed the event and concluded that no problems existed with the surveillance instructions or the TSV control circuitry. Operations personnel placed the reactor mode switch in "Startup" at about 11:30 a.m. (EST), on January 2, 1991, and began withdrawing control rods. The inspectors observed plant startup as the operators brought the reactor critical at about 2:00 p.m. The inspectors will review the forthcoming LER on this event in a future inspection period.

At about 11:50 p.m., on January 1, 1991, the licensee reported this event to the NRC operations center via ENS as an unplanned engineered safety feature actuation in accordance with 10 CFR 50.72(b)2(ii).

(11) Unexpected Closure of Containment Isolation Valves for the Main Steam Drain Lines

On January 6, 1991, at about 1:45 p.m., while reactor power was about 35 percent, an unexpected closure of the containment isolation valves in the main steam drain lines occurred. Operations personnel were placing the "6A" feedwater heater in service in accordance with System Operating Instruction (SOI)-N27, "Feedwater System." At about 1:00 p.m., the unit supervisor declared the main steam line radiation monitors inoperable, entered the Limiting Condition of Operation, took compensatory actions, and attempted to inhibit the isolation signals from the monitors by performing the applicable steps of the surveillance instructions for the radiation monitors (SVI-D17-T40A through D) as directed by System Operating Instruction (SOI)-N27. Licensee personnel incorrectly placed the "Nuclear Steam Supply System Main Steam Line Drain Isolation Logic" test switches in the "Test" position which sent isolation signals to the main steam drain valves.

The root causes were determined to be personnel error and inadequate procedure. Licensee personnel incorrectly determined that placing switches in test would prevent the isolations. In addition, procedural guidance in SOI-N27 was inadequate. The trip signals were removed and the valves were reopened. The inspectors will review in a future inspection period the forthcoming LER on this event.

At about 4:50 p.m., on January 6, 1991, the licensee reported this event to the NRC Operations Center via the ENS as an unplanned engineered safety feature actuation in accordance with 10 CFR 50.72(b)2(ii).

No Violations or Deviations were identified.

7. Plant Startup from Second Refueling Outage (71711)

During the report period, the licensee completed their second refueling outage. The generator was synchronized to the grid at about 12:00 noon, on January 4, 1991, with reactor power at about 14 percent. The inspectors observed the licensee's restart activities which included placement of the reactor mode switch in the "startup" position on December 24, 1990. The reactor achieved 100 percent power on January 6, 1991. The inspectors noted that the licensee's approach to plant startup following the extended outage was cautious and well controlled. Observations of control room activities indicated that plant operators were well briefed on planned events and were provided sufficient time to conduct startup testing and surveillance activities. Because of licensee concerns, as well as NRC concerns, on the relatively large number of personnel errors by licensed plant operators during the refueling outage, the licensee placed a second Senior Reactor Operator (SRO) in the control room to help oversee operations. The second SRO was scheduled to remain in place until attainment of 100% power.

No Violations or Deviations were identified.

8. Evaluation of Licensee Self-Assessment Capability (40500)

During this report period, the inspectors observed the function of the licensee's offsite review committee to evaluate the depth of review by that organization of overall plant performance. The inspectors observed the nuclear safety review committee meeting number-75 conducted on December 12, 1990.

The inspectors reviewed the meeting agenda and discussion topic handouts. Items reviewed included the subcommittee reports prepared by the audit and quality assurance subcommittee; the operations and maintenance subcommittee; the radiological, environmental and chemistry subcommittee; and the engineering subcommittee. The inspectors noted that those subcommittee reports contained current items of interest for the offsite review committee. The inspectors noted by observing the offsite committee meeting held on December 12 that subcommittee reports were presented in a clear manner with opportunity for the committee members to address specific areas of interest or concern.

In addition to the subcommittee reports, the inspectors observed the offsite review committee discussion of proposed changes to the Perry Technical Specifications. The inspectors noted that the offsite committee was provided sufficient information to act on those proposed changes.

The inspectors noted that the offsite committee meeting conducted on December 12 was well formatted with the required quorum of committee members in attendance. In general, the planned agenda was followed with an appropriate level of review. The inspectors concluded that the depth of review for the overall plant performance as discussed at the December 12 meeting was adequate.

No Violations or Deviations were identified.

9. Plant Status Meeting (30702)

NRC Management met with CEI management on November 28, 1990, at the Region III office and discussed: the status of the second refueling outage which started September 7, 1990; repairs and design modifications to the main steam isolation valves (MSIVs) and the hydraulic control unit solenoid pilot valves to prevent future failures; and events of interest since the last plant status meeting of August 7, 1990.

NRC management acknowledged the licensee's plans and current plant status.

10. Exit Interviews

The inspectors met with the licensee representatives denoted in Paragraph 1 throughout the inspection period and on January 7, 1991. The inspector summarized the scope and results of the inspection and discussed the likely content of the inspection report. The licensee did not indicate that any of the information disclosed during the inspection could be considered proprietary in nature.

During the report period, the inspectors attended the following exit interview:

Inspector	Exit Date
RER Team	11-30-90
Kavin Ward	12-12-90