U. S. NUCLEAR REGULATORY COMMISSION

REGION III

Report No. 50-440/92011(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: May 27 through June 15, 1992

Inspectors: W. Stearns
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Approved By: R. D. Lanksbury, Chief 7/2/92
Reactor Projects Section 3B Date

Inspection Summary

Inspection on May 27 through June 15, 1992 (Report No. 50-440/92011(DRP))
Areas Inspected: Special inspection by the Perry senior resident inspector and Region III inspectors of four events occurring during the Perry 1992 refueling outage. The purpose of this inspection was to evaluate the root cause(s) for three loss of water inventory events and failure of licensee core verification to identify a misaligned fuel bundle. The three unexpected loss of water inventory events included drainage of the suppression pool from 18.3 feet (5.58m) to 18.0 feet (5.49m); drainage from the circulating water (CW) system of 18,000 gallons (68,130L); and drainage from the reactor vessel from 217 inches (551cm) above top of active fuel (TAF) to 202 inches (513cm) above TAF.

Results:

The suppression pool drainage event resulted from the mechanical failure of a manual operator on a normally open maintenance isolation valve. The inspectors concluded that the licensee's administrative controls were adequate and control room personnel responded promptly to the unexpected drainage. Adherence to administrative controls and prompt response by control room personnel minimized the significance of this event.

The remaining events directly resulted from personnel errors. Existing administrative controls, if followed, would have prevented the three occurrences.

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The unexpected drainage from the circulating water system was due to the failure of control room supervisors to maintain system configuration control during maintenance. In addition, significant troubleshooting and corrective maintenance was performed under verbal authorization without full compliance with administrative controls. The inspectors concluded that verbal authorization was given in order to expedite the work effort as the licensee was completing the refueling outage. Response by control room operators to the actual drainage from the circulating water system was in accordance with plant procedures.

The unexpected drainage from the reactor vessel was due to failure of the control room unit supervisor to properly authorize instrument venting instructions. A violation with two examples of failure to follow administrative procedures was identified. Response by control room personnel to this event was good. Immediate recognition of the cause for the unexpected reactor vessel drainage and actions taken to control and isolate the drain path minimized the effect of the original personnel error.

The failure to identify the misaligned fuel bundle was due to personnel error (inattention to detail) during the independent review of the core verification data package. Two non-cited violations (NCVs) concerning the proper implementation of the core verification instruction were identified; however, these violations are not being cited because the criteria specified in Section VII.B of the "General Statement of Policy and Procedures for NRC Enforcement Actions," (Enforcement Policy, 10 CFR Part 2, Appendix C, (1992)), were satisfied.

DETAILS

1. Persons Contacted

   a. Cleveland Electric Illuminating Company

      R. Stratman, General Manager, Perry Nuclear Power Plant (PNPP)
      #*K. Donovan, Manager, Licensing and Compliance
      #*M. Gmyrek, Operations Manager, (PNPP)
      #*L. Teichman, Planning Unit Supervisor, Perry Maintenance Section (PNPP)
      *F. Stead, Director, Perry Nuclear Support Department (PNSD)
      #*H. Hegrat, Compliance Engineer, (PNSD)
      #*B. Walrath, Manager, Performance Engineering Section (PNED)
      #*P. Bordley, Reactor Engineer, Performance Engineering Section, PNED
      #*V. Concel, Manager, Technical Section, PNED
      *J. Eppich, Manager, Mechanical Design Section, PNED
      *D. Conran, Compliance Engineer, PNSD
      #*W. Coleman, Manager, Quality Assurance Section
      *D. Cobb, Superintendent, Plant Operations, PNPP
      #*W. Wright, Manager, Instrumentation and Control
      #J. Perry, QA Evaluator, PNSD
      #R. Gaston, Compliance Engineer, PNSD
      #W. Kanda, Manager, Electrical Design Section, PNED

   b. U. S. Nuclear Regulatory Commission

      #*P. Hiland, Senior Resident Inspector, RIII
      #A. Vegel, Resident Inspector, RIII
      *W. Stearns, Resident Inspector, RIII
      *M. Khanna, Intern, RIII

      *Denotes those attending the entrance meeting held on June 1, 1992.

      #Denotes those attending the exit meeting held on June 15, 1992.

2. Overview

   Toward the end of the licensee's third refueling outage, three unexpected loss of water inventory events occurred over a 5-day period. On May 23, 1992, the suppression pool water level was drained from 18.3 to 18.0 feet (5.58 to 5.49m). On May 26, approximately 18,000 gallons (68,130L) of water were drained from the circulating water system. On May 26, reactor vessel water level was drained from 217 to 202 inches (551 to 513cm) above the top of active fuel (TAF). In addition, on May 27, after an initial plant startup, a supplemental licensee review of the core verification videotape identified an improperly seated fuel bundle.

In response to these events, Region III management directed that this special safety inspection be conducted. The purpose of this inspection was to identify the root cause(s) for each of the four events and to evaluate the licensee's control of work activities during plant startup to determine whether expediting work efforts contributed to the event occurrences.

The inspection of the four events, documented in the following paragraphs, was accomplished through review of plant procedures, operating logs, work orders, chart recordings, videotapes, historical records, and direct field observation of equipment. Interviews with personnel knowledgeable of the events were also conducted. Attachment 1 to this inspection report lists the individuals interviewed by the inspectors.

3. Suppression Pool Drain Down

Initial Plant Condition

At the time of this event, the reactor plant was in Operational Condition 4, COLD SHUTDOWN. Train "A" of the residual heat removal (RHR-A) system was out of service and partially drained as a result of required corrective maintenance on RHR-A heat exchanger bypass valve 1E12-F048A. That corrective maintenance activity was being implemented in accordance with Work Order 92-2194, initiated May 20, 1992. Since repairs required valve disassembly, the associated tag-out (92-1331) had established an isolation boundary in the RHR-A piping system to permit draining of the 1E12-F048A valve.

Concurrent with the above activity, Work Order 92-2273, initiated May 23, was approved to commence work at about 3:30 p.m. on May 23. That work order was an emergent work activity to verify proper limit switch setpoints on the RHR-A suppression pool suction valve 1E12-F004A. The associated tag-out (92-1349) for Work Order 92-2273 had established the necessary administrative controls to permit limit switch adjustment on the suppression pool suction valve.

Suppression Pool Drainage

At 11:43 p.m. on May 23, the radwaste supervising operator (RSO) reported to the main control room that high auxiliary building sump levels were present. Shortly after that report, a low suppression pool level alarm at 18.0 feet (5.49m) was received in the main control room. Control room operators, aware of limit switch work activities, directed that the RHR-A suppression pool suction valve (1E12-F004A) be closed. After valve closure, suppression pool water level remained stable at 18.0 feet (5.49m). The excess water in the auxiliary building sumps resulted in backing up the auxiliary building floor drains onto the lower floor elevation. As the water drained down, sludge remains were left on the floor.

Licensee's Immediate Corrective Action

After the unexpected draining of the suppression pool was stabilized, plant auxiliary operators were instructed to verify boundary valves on the RHR-A system tag-out. Within a few minutes operators reported that a manual 18-inch (45.7cm) maintenance isolation valve, 1E12-F029A, just downstream of the RHR-A pump discharge, was not fully closed as required by the attached red tag. Operators were able to close the 1E12-F029A valve an additional 4 to 6 inches (10.2 to 15.2cm). The lower elevation of the auxiliary building was controlled as a contaminated area until cleanup was completed on May 24. Subsequent investigation identified that maintenance isolation valve 1E12-F029A had a broken mechanical operator. The valve operator was repaired by replacing the gear operator, stem nut, yoke adapter, and both bearing assemblies. The licensee initiated Condition Report (CR) 92-161 to document the investigation into this event and to document corrective actions taken.

Inspectors Root Cause Evaluation

The drain down of the suppression pool to the auxiliary building drain sumps occurred when RHR-A suppression pool suction valve 1E12-F004A was manually cycled open in order to adjust limit switches. The existing tag-out for concurrent work on the RHR-A heat exchanger bypass valve, 1E12-F048A, was reviewed prior to work release on the suppression pool suction valve. With the 18-inch (45.7cm) manual maintenance isolation valve (1E12-F029A) closed, no effect from cycling the suppression pool suction valve was expected. However, the 18-inch (45.7cm) manual maintenance isolation valve (1E12-F029A) was in fact open about 4 to 6 inches (10.2 to 15.2cm). That valve position allowed suppression pool water to drain through the suction valve (1E12-F004A), through the idle RHR-A pump, through the 18-inch (45.7cm) maintenance isolation valve, and into the RHR-A piping that was drained and maintained in a drained condition by tagged open drain valves. The draining of suppression pool water through that flow path was beyond the capacity of the auxiliary building drain sumps; therefore, some floor drains in the lower elevation of the auxiliary building backed up.

The reason the 18-inch (45.7cm) manual maintenance isolation valve was not placed in the fully closed position was a mechanical failure at the drive gear to housing bushing interface. The housing bushing provided the transition piece between the valve's stem protector and the upper housing of the manual operator. When initially closed and independently verified on May 21, plant operators turned the manual handwheel (18 turns/thread) until no further movement occurred. What had actually occurred during that attempted valve closure was a binding of the bushing to gear drive after the valve had been operated about 70 percent of its full travel. The 18-inch (45.7cm) manual maintenance isolation valve (1E12-F029A) did not have a valve position indicator.

The root cause for this event was that the stem protector pipe bushing on valve 1E12-F029A was screwed too far into the gear reducer against the rotating member (installed in 1985). The rotating member threaded the bushing farther into the operator. This caused the operator to bind and gave indication that the valve was shut, even though the valve was not fully seated. When fully closed on May 23, the top of the valve housing cracked.

Inspectors' Conclusions

This event was due to a mechanical failure. The mechanical failure was an isolated occurrence in that no previous maintenance history identified the type of failure experienced on the 1E12-F029A valve. Licensee controls were in place and in accordance with station procedures. Had there not been a mechanical failure of the 18-inch (45.7cm) maintenance isolation valve, 1E12-F029A, the controls would have been adequate for the work activity.

No violations or deviations were identified.

4. Unexpected Drainage of Circulating Water

Initial Plant Conditions

At the time of this event, the reactor plant was in Operational Condition 4, COLD SHUTDOWN. The circulating water (CW) system was in a secured status with portions of the system drained for repairs to a failed flow instrument (annubar) located inside the CW system piping. Following corrective maintenance, the CW system was being filled with water as a prerequisite to starting the CW system. At Perry the CW system was a closed loop cooling water system which provided the normal heat sink for the main condenser (main turbines) and two auxiliary condensers (feedwater turbines). Makeup water to the CW system was provided by the service water (SW) system which received its water supply from Lake Erie.

The CW system fill was being accomplished in accordance with System Operating Instruction (SOI)-N71, "Circulating Water/Condenser Mechanical Cleaning System," revision 4, through Temporary Change Notice-7.

Circulating Water System Drainage

Following shift turnover at about 8:00 a.m. on May 26, the day shift control room operators continued with performing SOI-N71. At 8:38 a.m., CW Pump "A" was started. At 8:47 a.m., CW Pump "B" was started. With two of the three CW pumps operating, adequate system pressure was available to properly vent system piping.

At about 9:00 a.m. control room operators were informed that an excessive amount of water was being sent to the radwaste system from the turbine power complex (TPC) sump pump. In response to that notification, control room operators secured the operating CW pumps. About 18,000 gallons (68,130L) of water from the CW system had been drained to the radwaste collection facilities.

Operator and Plant Response

The immediate response by plant operators was to secure both operating CW pumps. Following building walkdowns, the most likely reason for CW system drainage during the fill and vent evolution was believed to be a stuck open automatic vent valve. At about 10:00 a.m., with personnel positioned at accessible locations in the turbine building, CW Pumps "A" and "B" were restarted. At that time, a plant operator identified the source of water drainage to be an open 16-inch (40.6cm) CW system water box drain valve (1N71-F532A). That drain valve was closed and no further unexpected system drainage occurred.

Licensee's Immediate Corrective Action

Licensee CR 92-165 was initiated to document the licensee's investigation of the cause for this event and the corrective action taken. In addition to the immediate operator action taken, as discussed above, plant personnel responsible for procedural changes were requested to add a reference in the applicable section of the filling instruction to ensure the 16-inch (40.6cm) drain valves were closed.

Inspectors Root Cause Evaluation

On May 25, 1992, at about 10:15 a.m., with the reactor plant in Operational Condition 4, COLD SHUTDOWN, and the CW system running in recirculation, a "loud banging noise" was reported coming from the main condenser. As documented in the Unit Log, troubleshooting efforts between 10:15 a.m. and 11:45 a.m. identified the location of the noise to be in the "A"/"B" train split upstream of the condenser inlet isolation valves.

At 12:00 noon a work request was initiated to generate the necessary work order to perform an inspection of the CW pipe internals. In parallel with that effort, control room personnel initiated actions to drain the CW system piping. SOI-N71, Section 7.6 provided instructions for partial main condenser isolation during operation. The shift supervisor directed that the CW system be drained in accordance with SOI-N71, Section 7.6.

At the time of shift turnover (4:00 p.m.) from day shift to swing shift on May 25, the CW system was still draining, the work order (92-2295) was prepared but not approved to work, and the associated system tag-out was prepared but not authorized to hang. At some time during swing shift on May 25, the cause for the loud banging was identified to be an "annubar" for CW flow instrument 1N71-N230A. Since it was not required for normal plant operation, the annubar was permanently removed from the CW system. The annubar was originally installed to obtain data during startup testing.

At the time of shift turnover (12:00 midnight), all field work was complete and the CW system was reported in a "secured status." With the work order completed, the midshift unit supervisor (SRO) and the work group supervisor authorized the tag-out clearance. About 4:00 a.m. on May 26 the unit supervisor initiated the system restoration in accordance with SOI-N71, Section 7.1, "Circulating Water System Fill."

As noted above, the day shift operating crew commenced a system drain-down in accordance with SOI-N71, Section 7.6, on May 25. That section of the SOI was written assuming the plant was in operation and included the instruction to throttle open drain valve 1N71-F532A (Step 7.6.5.c). However, the corresponding instruction to close the throttled open drain valve did not appear until Section 7.7, "Filling and Returning an Isolated Main Condenser Section to Service," Step 7.7.2. Similar to Section 7.6, Section 7.7 of the SOI was written assuming the plant was in operation. Since the midshift operating crew was informed that the CW system was in a "secured status" and no deviations to plant instruction were documented, they entered SOI-N71 at the intended section for filling the CW system from a "secured status." As a result, the opened drain throttle valve, 1N71-F532A, remained in the as-left position as instructed by Section 7.6 of SOI-N71 until water pressure, with two CW pumps in operation, caused the associated water box drain tank to overflow.

The root cause for this event was the failure of the "day shift" shift supervisor on May 25 to effectively maintain configuration control over the CW system.

The inspectors review of work control processes identified the following deviations from established Perry Administrative Procedures (PAPs):

a. PAP-0201, "Conduct of Operations," Section 6.5, "Procedural Compliance," Item 6.5.2.3 required in part that infrequent deviations from system operating instructions (SOIs) be approved by the shift supervisor and be documented in the plant log.

   Contrary to that requirement, on May 25, the shift supervisor approved a deviation to SOI-N71, Section 7.6 without documenting the deviation in the plant log.

b. PAP-0905, "Work Order Process," Section 6.11, "Troubleshooting Log," required in part that if immediate corrective action was required a Troubleshooting Log be prepared and troubleshooting or corrective actions taken be documented.

   Contrary to that requirement, between 12:00 noon and 10:23 p.m. on May 25, significant troubleshooting efforts were performed under verbal authorization without initiating a Troubleshooting Log.

c. PAP-0516, "Confined Space Entry and Industrial Hygiene Sampling," Section 6.1.1 required in part that when a confined space was opened for any reason, the work supervisor shall ensure the space is tagged. Additionally, the confined space tag-out shall be a "Red Tag" tag-out due to personnel safety concerns.

Contrary to that requirement, on May 25, entry into the CW system piping was made under a confined space permit without the associated tag-out in place.

Inspectors' Conclusion

The inspectors noted that the CW system at Perry was not a safety-related system. However, the administrative procedures governing the work activities conducted on May 25 were also applicable to safety-related activities. During this inspection period, the inspectors discussed the above failures to follow administrative procedures with the General Plant Manager. The inspectors concluded that work activities on the CW system between 12:00 noon and 10:23 p.m. on May 25 were conducted under verbal authorization in an attempt to expedite the corrective action process. As discussed above, verbal communications were not adequate to ensure that proper administrative controls were adhered to.

No violations were identified. Three examples of failure to follow administrative procedures during work activities on nonsafety-related system components were identified.

5. Loss of Reactor Water Level

Initial Plant Condition

At the time of this event, the reactor plant was in Operational Condition 4, COLD SHUTDOWN. Reactor vessel level band was to be maintained 200 to 260 inches (508 to 660cm) above the top of active fuel (TAF). Reactor coolant temperature band was 150 to 190°F (65.6 to 87.8°C). Residual heat removal Train "B" (RHR-B) was providing shutdown cooling. Residual heat removal Train "A" (RHR-A) was inoperable due to corrective maintenance on its associated heat exchangers bypass valve 1E12-F048A. The reactor water cleanup system was aligned to drain to the main condenser for reactor water level control.

RHR-A was to be returned to service following a post maintenance pump and valve operability test. That test was to be conducted in accordance with Surveillance Instruction (SVI) E12-T2001, "RHR A Pump and Valve Operability Test," revision 7, through Temporary Change Notice-7. At 2:25 p.m., the unit supervisor (SRO) authorized the start of prerequisites for SVI-E12-T2001. At 2:42 p.m., RHR-A was placed in standby. As required by Section 4.10 of SVI E12-T2001, the operating crew directed instrument and control (I&C) personnel to fill and vent RHR flow transmitters sometime between 2:25 and 3:25 p.m.

Unexpected Reactor Level Drainage

At 3:25 p.m., I&C technicians began to fill and vent RHR-B (operating in shutdown cooling) flow transmitter 1E12-N052B. That effort resulted in a low flow signal generation and the automatic opening of the RHR-B pump minimum flow valve 1E12-F064B. With the minimum flow valve open and RHR-B operating in its shutdown cooling lineup, a bypass of the intended system flow occurred. RHR-B was aligned to remove coolant from the reactor vessel via the open 1E12-F006B shutdown cooling suction valve, and to discharge coolant through its associated heat exchangers back to the reactor vessel via the feedwater system. With the RHR-B minimum flow valve (1E12-F064B) open, a portion of the pump's discharge was directed to the suppression pool and not returned to the reactor vessel. Over a 2 minute period, reactor water level dropped from 217 inches (551cm) above TAF to 202 inches (513cm) above TAF.

Operator and Plant Response

Control room alarm "Possible Reactor Siphon Via RHR-F006B and F064B," located on main control room panel H13-P601, annunciated as designed at the time the RHR-B minimum flow valve (1E12-F064B) opened. In response, control room personnel made preparations to secure RHR-B, secured the reactor water cleanup system drain path to the main condenser, and placed the RHR-B minimum flow valve control switch (spring return to AUTO) in the close position. Concurrently, the unit supervisor paged the I&C technicians performing the fill and vent evolution on the RHR system and requested that they secure the evolution. Prior to actually securing the RHR-B train, I&C technicians restored the RHR-B flow transmitter. Reactor water level remained stable at 202 inches (513cm) above TAF.

Licensee's Immediate Corrective Action

Immediate corrective actions performed by control room personnel were effective and in accordance with plant alarm response instructions. The decision to maintain RHR-B operating in a shutdown cooling alignment appeared reasonable and maintained reactor water level within the prescribed band of 200 to 260 inches (508 to 660cm). Following identification of the fill and vent evolution being performed on RHR-B, the unit supervisor directed that evolution be suspended. The Operations Manager issued a Daily Instruction (ref. DI dated 5/26 sheets 2 and 3) which reiterated the philosophy of "Control not Speed."

The licensee initiated CR 92-162, dated May 26, 1992, to document the investigation of the root cause for this event. In addition, the General Plant Manager directed that a human performance enhancement system (HPES) evaluation be performed concurrent with the CR investigation.

Inspectors Root Cause Evaluation

The unexpected decrease in reactor water level began with an attempt to fill and vent RHR-B flow transmitter 1E12-N052B. The operating crew had directed the fill and vent evolution as a prerequisite to performing a post maintenance pump and valve operability test on RHR-A. In accordance with Instrument and Controls Section Administrative Procedure (IAP)-0503, "Plant Instrument Calibration and Maintenance," revision 4, an I&C Instrument Valve Lineup/Fill-Vent sheet was prepared and presented to the day shift unit supervisor for authorization at about 2:30 p.m. on May 26. As prepared, that Fill-Vent sheet identified all RHR flow transmitters (nine) and was not limited to the RHR-A train and its associated flow transmitters (two). The day shift unit supervisor provided written authorization by signature on the presented Fill-Vent sheet. Although some verbal communications occurred concerning the specific order or number of instruments to be filled and vented, no modifications were made to the authorized Fill-Vent sheet. With written authorization to fill and vent all nine instruments identified on the Fill-Vent sheet, I&C technicians commenced that work assignment on the RHR-C train instruments, with no problems identified (RHR-C was in standby mode only and does not have a "shutdown cooling" mode). Following the completion of the RHR-C instruments, the I&C technicians initiated the fill and vent on RHR-B flow transmitter 1E12-N052B at which time they were directed by the unit supervisor to immediately return that instrument to service.

The root cause of this event was the failure of the day shift unit supervisor to properly provide written authorization to fill and vent only the RHR-A flow transmitters. In addition, a contributing factor was the lack of awareness of plant conditions by I&C personnel.

Technical Specification (TS) 6.8.1.a required that written procedures and instructions be established, implemented, and maintained as recommended in Appendix A of Regulatory Guide 1.33, revision 2, February 1978. Appendix A, Item 4, required instructions for filling, venting, startup, and changing modes of operation of the emergency core cooling systems. The inspectors review of work control processes identified the following deviations from established PAPs:

a. PAP-0205, "Operability of Plant Systems", section 6.5.3, required in part that necessary modifications to a lineup or checklist be documented.

   Contrary to that requirement, the I&C Instrument Valve Lineup/Fill-Vent sheet, dated May 26, 1992, for nine flow transmitter instruments associated with the RHR system was authorized by the unit supervisor on May 26, 1992, without documenting required modifications for the existing plant conditions. This is a Violation (440/92011-01A(DRP)).

b. PAP-0201, "Conduct of Operations," revision 8, provided instruction and guidance to ensure that plant operations were conducted in a safe manner. PAP-0201, Section 6.4.2, required in part that operation of mechanisms and apparatuses shall only be accomplished with the knowledge and consent of the licensed operator "at the controls." On May 26, 1992, the day shift unit supervisor authorized I&C technicians to perform a fill and vent evolution on nine flow transmitter instruments associated with the RHR system. Instrument Maintenance Instruction (IMI)-E2-1, "Instrument Valve Line-Ups," revision 1, was the implementing instruction used to perform the authorized fill and vent evolution. IMI-E2-1, Section 4.2, required in part that the supervising operator (RO) be informed of any interlocks that may be received during the performance of that instruction.

Contrary to the above requirements, on May 26, 1992, the supervising operator "at the controls" was not informed of the authorization or planned performance of fill and vent evolutions on the nine flow transmitters associated with the RHR system and that the RHR-B minimum flow valve would open. This is a Violation (440/92011-01B(DRP)).

Inspectors' Conclusion

The root cause for this event was the failure of licensed control room personnel to ensure proper communication of requested work activities. The administrative controls were available and, if properly used, would have prevented the event from occurring. In addition to the procedural violations discussed above, the inspectors noted that I&C personnel involved in the preparation and implementation of the RHR flow instrument Fill-Vent sheet were unaware of existing plant conditions. Awareness of system or component status during maintenance on associated instrumentation is fundamental knowledge required for controlled evolutions.

One violation with two examples of failure to follow procedures was identified. No deviations were identified.

6 Identification of Misaligned Fuel Bundle

Background

At the completion of fuel load activities during the Perry 1992 refueling outage, core verification was completed on April 19, 1992, with no discrepancies noted. The reactor vessel was reassembled and plant startup commenced on May 26.

Independent of these activities, General Electric, the fuel vendor, had identified a concern regarding peripheral fuel bundles and a vulnerability with regard to minimum critical power ratio (MCPR) and orientation of the bundles. Because of that concern, the licensee's Quality Assurance (QA) group obtained the core verification videotapes recorded in April and performed a second independent review documented in licensee Surveillance Report 92-194.

Initial Plant Conditions

At the time of this event, the reactor plant was in Operational Condition 2, STARTUP. On May 27, during the performance of QA surveillance 92-194, the licensee identified a misaligned (improperly seated) peripheral fuel bundle. As a result of that discovery, plant startup was terminated and the reactor was returned to Operational Condition 4, COLD SHUTDOWN.

Licensee's Immediate Corrective Action

The potential for misalignment of boiling water reactor (BWR)/6 peripheral fuel bundles was the subject of General Electric Rapid Information Communication Service Information Letter (RICSIL) No. 014, dated December 14, 1987. As stated in RICSIL No. 014, operation with a misaligned peripheral fuel bundle would not cause a safety problem or result in violation of any TS safety limit. However, initial licensee concerns included the possibility of a loose part between the fuel support piece and the fuel assembly or that the intermediate range monitor (IRM) instrument tube, located adjacent to the misaligned bundle, was itself an obstruction. The licensee concluded that an appropriate corrective action was to disassemble the reactor vessel and properly seat the misaligned fuel bundle. As documented in licensee Surveillance Report 92-206, video inspection of the fuel support piece, following the removal of an adjacent bundle, determined that no foreign material was present and the fuel bundle had been misaligned about two inches to one side. A third independent review of the fuel assembly locations was performed as documented in licensee Surveillance Report 92-195.

The licensee initiated CR 92-163, dated May 27, 1992, to document the investigation of the root cause for this event. In addition, the General Plant Manager directed that a human performance enhancement system (HPES) evaluation be performed concurrent with the CR investigation.

Inspectors Root Cause Evaluation

The inspectors reviewed the core verification data package prepared during the implementation of Fuel Accountability Instruction (FTI)-D1, "Core Verification," revision 2. That instruction was performed following completion of fuel load activities on April 19, 1992. In addition, the inspectors reviewed the original videotapes prepared during the core verification process and the subsequent video inspection of the misaligned fuel bundle. The core verification data package indicated all instructional steps had been performed for both the initial verification and the independent verification. One noted discrepancy in the documentation was the failure to document the specific portion of the initial verification performed (step 5.2.5) when multiple individuals were involved.

B 'h methods (initial and independent) used for fuel assembly seating vc ification failed to identify the misaligned fuel bundle when core verification was performed on April 19. The following is the inspectors' evaluation of the initial, or bump test, and independent, or videotape review, methods employed by the licensee.

a. Bump Test

The first method used described in Fuel Accountability Instruction (FTI)-D1 was the "bump" method. That method consisted of lowering the fuel grapple until it was approximately 0.5 inch (1.27 cm) above the fuel assembly bail handles and slowly traversing each row of the core to verify proper seating. If the grapple bumped any of the bail handles, that fuel bundle was probably not properly seated.

In practice, the grapple was lowered and the digital readout obtained for the height. This was done for several bundles until a nominal value was obtained. The grapple was then set at 0.5 inch (1.27 cm) above the nominal bail handle height. The mast was then held by hand to "feel" for vibration from an impact as it was traversed across each row. The maximum height expected for a fuel bundle not properly seated was about 2 inches (5.08 cm) above its "normal" height.

The inspectors noted the following potential weaknesses with the "bump" method:

1) The digital height indication was only accurate to ± 1 inch (2.54 cm).

2) The fuel bridge mast did not stay vertical when the bridge moved, but swayed with motion. This had the effect of raising the grapple above the fuel higher than desired.

3) Fuel bundles are inherently different lengths due to burnup, (i.e., growing in length with life). Exact figures for Perry fuel were not known. Measured expansion at another boiling water reactor was on the order of 0.75 inch (1.9cm).

4) Due to the proximity of the fuel bundles and the shroud at the periphery of the core, it was possible to misidentify contact with the shroud as contact with a bundle.

b. Videotape Review

The second method used by the licensee to determine proper fuel assembly seating also failed to detect the misaligned bundle. That method consisted of reviewing a videotape of each fuel assembly concurrently for location (serial # vs. position) and for proper seating. Specific parameters monitored for proper seating were:

1) Relative height of assemblies
2) Mismated channel spacer buttons
3) Uneven shadows
4) Out-of-focus serial numbers *
5) Channel fastener obstruction

* The inspectors noted during review of the videotape containing the misaligned fuel bundle that the serial numbers remained within focus and by itself would not have provided evidence of the misalignment.

The videotape review of proper fuel bundle seating was performed by plant reactor engineers with the assistance of a licensed senior reactor operator. All reactor engineers were trained as a part of their qualification program by reviewing previously recorded videotapes to perform a mock core verification procedure. The inspectors' review of the videotapes indicated that had sufficient "attention to detail" been practiced with regard to prop seating during the independent verification, the misaligned bundle would have been discovered.

Inspectors' Conclusion

The cause of this event was personnel error (inattention to detail) while performing the videotape review during the independent fuel bundle seating verification. During review of potential factors which could have contributed to this event, the inspectors concluded that neither excessive overtime nor other distractions played a role in impacting the performance of the core verification process.

The inspectors noted there were other missed opportunities to have discovered the misaligned bundle. There were initial reviews of the videotapes for location and orientation verification (seating verification not required) that provided an opportunity for discovery.

Technical Specification 6.8.1.a required that written procedures and instructions be established, implemented, and maintained as recommended in Appendix A of Regulatory Guide 1.33, revision 2, February 1978. Appendix A, Item 2, required procedures for refueling. Fuel Accountability Instruction (FTI)-D1, "Core Verification," revision 2, was the implementing instruction used to independently verify fuel bundle seating on April 19, 1992. Failure of the independent reviewers to properly implement that instruction is a violation. Additionally, failure of personnel performing a portion of the initial verification to properly document the portion performed is a violation. These violations were not cited because the licensee's efforts in identifying and correcting the violations met the criteria specified in Section VII.B of the Enforcement Policy.

Two non-cited violations (NCVs) were identified. No deviations were identified.

7. Items For Which A "Notice of Violation" Will Not Be Issued

During this inspection, certain licensee activities, as described above in Paragraph 6, appeared to be in violation of NRC requirements. However, the licensee identified these violations and they are not being cited because the criteria specified in Section VII.B of the "General Statement of Policy and Procedures for NRC Enforcement Actions," (Enforcement Policy, 10 CFR Part 2, Appendix C, (1992)), were satisfied.

8. Exit Interviews

The inspectors met with the licensee representatives denoted in Paragraph 1 throughout the inspection period and on June 15, 1992. The inspectors 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.

ATTACHMENT #1 TO INSPECTION REPORT 50-440/92011

PERSON INTERVIEWED	TITLE
P. BORDLEY	REACTOR ENGINEER
T. CARROLL	FIELD ENGINEER
J. CAGE	SUPERVISING OPERATOR
D. COBB	OPERATIONS SUPERINTENDENT
D. DI COLA	REACTOR ENGINEER
M. FAHL	SUPERVISING OPERATOR
M. GMYREK	OPERATIONS MANAGER
J. HERMAN Jr.	TAG-OUT GROUP
D. JOHNSON	UNIT SUPERVISOR
R. KEARNEY	UNIT SUPERVISOR
J. McHUGH	SENIOR REACTOR OPERATOR
J. MIKOLIJ	UNIT SUPERVISOR
D. MORGAN	I&C TECHNICIAN
A. OKORN	SHIFT SUPERVISOR
K. PECH	OUTAGE MANAGER
A. RABENOLD	SUPERVISING OPERATOR
D. RICHMOND	PERRY PLANT OPERATOR
J. RINCKEL	REACTOR ENGINEER
R. SMITH	UNIT SUPERVISOR
R. SOCHIA	SHIFT SUPERVISOR
R. STEVENS	I&C SUPERVISOR
B. STETSON	SUPERVISING OPERATOR
R. STIFLER	SHIFT SUPERVISOR
J. TARKOWSKI	SUPERVISING OPERATOR
R. YOUNG	I&C TECHNICIAN