U. S. NUCLEAR REGULATORY COMMISSION

REGION III

Report Nos. 50-237/92009(DRP); 50-249/92009(DRP)

Docket Nos. 50-237; 50-249 License Nos. DPR-19; DPR-25

Licensee: Commonwealth Edison Company

Facility Name: Dresden Nuclear Power Station, Units 2 and 3

Inspection At: Dresden Site, Morris, IL

Inspection Conducted: March 23 through May 1, 1992

Inspectors: W. Rogers
M. Peck
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A. Madison
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Approved By: R C Knop 5/18/92
R. C. Knop, Chief, Date
Projects Section 1B

Inspection Summary

Inspection from March 23 through May 1, 1992 (Report Nos. 50-237/92009(DRP); 50-249/92009(DRP)).

Areas Inspected: A special unannounced safety inspection was conducted by resident, headquarters, and regional inspectors to review previously identified items regarding (1) the failure to close of a reactor recirculation discharge valve, (2) flooding concerns on the diesel generator cooling water pumps, and (3) recurrences of previous violations.

Results: Five apparent violations were identified: Inoperability of the low pressure coolant injection system (Section 3), Failure to have procedures as required by Part 21, (Section 4), Deficiencies in the corporate quality assurance control of the motor operated valve program (Section 5), Inoperability of the Unit 2/3 diesel generator under cribhouse flooding conditions (Section 6), and Deficiencies in the corrective action process resulting in failure to take corrective actions on a condition adverse to quality and in repeats of previous violations (Sections 6-9).

DETAILS

1. Persons Contacted

Commonwealth Edison

[1]K. Graesser, General Manager, Boiling Water Reactor (BWR) Operations
[1]C. Schroeder, Station Manager
[2]B. Adams, Regulatory Assurance, Engineering and Construction (ENC)
[1,2]S. Berg, Assistant Superintendent - Production
[1,2]E. Carrol, Regulatory Assurance
[2]C. Collins, Site Engineer, Nuclear Engineering Department (NED)
[1]L. Gerner, Technical Superintendent
[2]D. Hoffman, Nuclear Quality Programs
[1]D. Karjala, Performance Improvement Director
[1]J. Kish, On-Site Nuclear Safety
[1]J. Kowtowski, Production Superintendent
[1]W. Morgan, Corporate Nuclear Operations
[2]H. Mulderink, BWR Motor Operated Valve Coordinator, NED
[1]K. Peterman, Procedure Manager
[1,2]R. Radtke, Regulatory Assurance Supervisor
[2]R. Ralph, Assistant Technical Staff Supervisor
[2]R. Rybak, Mechanical and Structural Design Supervisor, NED
[2]T. Schuester, Nuclear Licensing Supervisor
[2]G. Smith, Assistant Superintendent - Operations
[1]M. Strait, Technical Staff Supervisor
[2]D. Taylor, Regulatory Assurance Supervisor, ENC
[1]B. Viehl, NED Site Supervisor

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[1]B. Burgess, Chief, Section 1B
[1,2]A. Hsia, Acting Chief, Section 1B
[2]W. Rogers, Senior Resident Inspector
[1,2]K. Shembarger, Reactor Engineer

[1]Present at the exit held by A. Madison on April 16, 1992.
[2]Present at the exit held by P. Lougheed on May 1, 1992.

The inspectors also talked with and interviewed other licensee employees throughout the course of the inspection period.

2. Action on Previously Identified Items

A. (Closed) Unresolved Item (237/90021-01(DRP)) "Reactor Recirculation Discharge Valve 2-202-5A Failure to Close." As discussed in inspection report 237/91036(DRP), this unresolved item had three issues which required review: (1) Completion and review of the licensee's 10 CFR Part 21 evaluation, (2) further NRC review of the licensee's condition adverse to quality system, and (3) further NRC review of the licensee's calculational

controls. These reviews have been completed and the conclusions are discussed in Sections 4, 5, and 10 of this report. Therefore, this item is considered closed.

B. (Closed) Unresolved Item (237/92005-01) "Failure to Take Adequate Corrective Actions to Prevent Recurrence of Previous Violations." The causes of the recurrences of these violations are discussed in Section 7 of this report. Based on the conclusions reached there, this item is considered closed.

3. Reactor Recirculation Discharge Valve Failure to Close Resulting in the Low Pressure Coolant Injection System Being Inoperable

A. Background: The low pressure coolant injection (LPCI) system is one of several emergency core cooling systems (ECCS) that operate following a loss of coolant accident (LOCA). It takes suction from the torus and returns water to the reactor vessel through the reactor recirculation injection line. In order to ensure that LPCI water is injected into the reactor vessel, the appropriate reactor recirculation loop discharge valve is required to close.

B. Description of Event: As discussed in special inspection report 50-237/91036(DRP), during the fall 1990 Unit 2 refueling outage, an incorrect torque switch setting was established for the 2A reactor recirculation discharge valve (2-202-5A) motor operator due to misinterpretation of the zero point on a Liberty Technologies Valve Operation Test Evaluation System (VOTES) trace. After completion of refueling activities, Unit 2 resumed operation on January 4, 1991.

On August 6, 1991, operators attempted to close valve 2-202-5A prior to restarting the reactor recirculation pump following maintenance. The operators then discovered that the discharge valve torque switch was incorrectly set. The torque switch setting was corrected on August 10, 1991.

C. Analysis of Root Cause: As discussed in special inspection report 50-237/91036(DRP), the root cause of the valve being inoperable was the misinterpretation of the VOTES trace, due to valve stem anomalies that were not apparent to the valve testers. Because of limitations in the VOTES testing program, including the methodology, training and software, the person performing the valve testing was not equipped to properly evaluate the reactor recirculation discharge valve data.

D. Assessment of Safety Significance: If a LOCA had occurred between January 4, 1991, and August 10, 1991, the 2A reactor recirculation discharge valve would not have closed. Because the valve was unable to close, the LPCI system may not have been able to achieve its intended safety function. Technical Specification 3.5.A.5 allows continued operation for a maximum of seven days, with LPCI inoperable for any reason. This seven day period was exceeded.

This is considered an apparent violation of Technical Specification 3.5.A.5 (237/92009-01(DRP)).

4. 10 CFR Part 21 Procedural Deficiencies

A. Background: During the review of the failure of the reactor recirculation valve, the NRC identified an unresolved item regarding the licensee's Part 21 evaluation process.

10 CFR Part 21.21 requires a licensee to have procedures to evaluate deviations and to ensure that appropriate licensee management is made aware of any defects found during the evaluations, and that appropriate notifications are made once a defect is identified.

B. Description of Inspection Findings: Three separate screenings for Part 21 applicability were performed on the reactor recirculation valve failure. On August 10, 1991, the Operations Engineer initially screened the valve failure to close and concluded a Part 21 evaluation was not required. On August 12, 1991, either the technical staff supervisor, or an assistant technical staff supervisor, completed a reportability screening of the event and again concluded a Part 21 evaluation was not required. On September 4, 1991, the On-Site Review Committee reviewed the event investigation and proposed corrective actions. Their investigation concluded the incorrect torque switch setting was the result of an inappropriate zeroing of the VOTES trace and that a Part 21 evaluation was not required. In all three reviews, the only documentation for Part 21 screening was a checklist box marked "no."

On January 14, 1992, the NRC completed a special inspection which determined that the root cause of the event was VOTES process limitations. The NRC further concluded that the licensee's initial LER was incorrect when it concluded that an improved version of the VOTES software would have identified the condition. The licensee began a Part 21 evaluation, and on February 5, 1992, they determined that a defect existed associated with the VOTES process and issued a Part 21 notification.

C. Analysis of Root Cause: The inspectors identified the following causes as contributing to the licensee's failure to identify that a Part 21 evaluation was required:

Technical staff management appeared to be aware of Part 21 requirements when it came to hardware issues. However, they did not have the necessary procedural guidance or background to address Part 21 issues in the areas of software, methodology, or training.

Dresden Administrative Procedure (DAP) 2-8, "Deviation Reporting," provided the basic definition of a deviation and noted that existence of a deviation would require further evaluation by the corporate engineering staff. However, it did not provide sufficient guidance to allow site personnel to identify software, methodology, or training Part 21 issues.

The licensee did not have a formal training program on Part 21 requirements for onsite personnel. Training was mainly on-the-job exposure to non-conforming conditions, usually hardware discrepancies.

D. Assessment of Safety Significance:

Because of the above factors, while there were three screenings, no evaluation for Part 21 applicability was performed until after the NRC inspection questioned the acceptability of not performing such an evaluation. When the evaluation was performed, a deviation was identified. This deviation was further evaluated and a Part 21 defect was discovered. The initial inadequate screenings could have resulted in this defect not being identified and reported.

The failure to have appropriate procedures addressing how to identify and evaluate a deviation is considered an apparent violation of 10 CFR 21.21 (237/92009-02(DRP)).

5. Corporate QA Program

A. Background: The corporate based nuclear engineering department (NED) implemented a motor operated valve (MOV) testing program to meet the requirements of Generic Letter 85-03 and NRC Generic Letter 89-10. They contracted with an architect/engineering (A/E) firm to reestablish the design basis for MOVs, under the A/Es accepted quality assurance (QA) program. The A/E provided to NED calculated valve thrust values to be used with the VOTES software. NED then transmitted these thrust values to the site where they were used to determine required MOV torque switch settings.

Following the reactor recirculation valve failure, the NED MOV group reanalyzed all 39 safety-related VOTES tests performed during the December 1990 refueling outage. During this reanalysis, five valves were identified to have thrust values outside the design specification. One valve was reported to the NRC as being non-conforming in a response to Generic Letter 89-10. For the other four valves, the design specification was revised and retransmitted to the station to be incorporated into the next testing effort.

B. Description of Inspection Findings: During follow up inspection efforts as to the root cause of the reactor recirculation valve failure to close, NRC found that the licensee did not have any

procedures covering their activities on the MOV program. NRC identified that NED routinely altered controlled calculations in order to reflect actual field data such as torque switch tolerance, valve lubrication history, measurement equipment uncertainty and spring pack capacity, or to remove design margin in order to bring non-conforming valves into conformance. These adjustments were performed by pen and ink markups, without justification of the new values and without any formal review of either the assumptions or the final conclusions. In some cases, the final thrust window values issued by NED to the station were outside of the thrust windows established by the Bechtel calculations. NED stated that these adjustments were acceptable because the new maximum values were still below the valve or actuator calculated structural limit. Specific examples of adjustments include valves 2-1001-01B, 3-1301-1, 3-1001-05A, and 3-3702; however numerous other examples exist.

NRC also identified that, due to the method by which the licensee incorporated their quality assurance program, NED did not have an overall procedure addressing handling of conditions adverse to quality. Instead the governing procedure for the work being done was to provide instructions for dealing with nonconforming conditions. However, in the case of the MOV program, because no procedure governed program activities, there was no guidance within NED on how to deal with conditions adverse to quality. Therefore, when NED reevaluated safety-related VOTES tests results following discovery of the incorrect 2-202-5A torque switch setting and discovered that 17 of the 37 required rezeroing and that 4 valves had thrust values outside the target window, there was no mechanism to document these findings, to ensure corrective actions, or to prevent recurrence. The four valves mentioned above were: 2-1001-2A, 2-1001-2B, 2-1501-5C, and 2-1501-32A.

At NRC's request, the licensee's nuclear quality programs group (NQP) performed an audit specifically of the MOV program within engineering during the week of March 23-27, 1992. The audit results confirmed that the same deficiencies existed within the NED MOV program as identified by the inspectors and an audit action item was generated.

C. Analysis of Root Cause: NED considered the A/E thrust values to constitute the design basis. Therefore, NED did not prepare any procedures to address QA requirements for the MOV program, and did not believe that generic procedures, such as QE 51.D "Controlled Analysis Originated by Nuclear Engineering Department", applied to their activities. NED did not consider the alterations being done to the controlled MOV calculations to constitute design changes, because the design basis was not being altered.

NED also failed to recognize that placing corrective action requirements in a program procedure rather than a generic corrective actions procedure could result in situations where

procedural guidance on handling of conditions adverse to quality did not exist.

These attitudes appeared to be due to a deficiency in the licensee's QA manual, which implements the NRC-approved QA Topical Report. The procedures in the QA manual applied only to the operating stations, and to "construction" activities. Activities affecting quality performed by corporate engineering for operating units were not addressed in the QA manual. The licensee considered the activities of the corporate engineering staff to fall under the "construction" portion of the QA manual and overlooked the transition from engineering support of construction activities to engineering support of operational activities.

D. Assessment of Safety Significance: NRC determined that the licensee failed to incorporate the motor operated valve program activities into procedures specific to that activity. This is contrary to the requirements of 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings" which requires that activities affecting quality be prescribed by documented procedures, instructions, or drawings appropriate to the circumstance (237/249-92009-03a(DRP)).

Because no program procedures existed, and because the licensee concluded that the overall program requirements specified in procedure QE 51.D did not apply, when changes to the design were made the licensee did not ensure that the requirements of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," were met. Criterion III requires that design changes be subject to the design control measures commensurate with those applied to the original design and be approved by the organization that performed the original design (237/249-92009-03b(DRP)).

Because neither program specific nor generic procedures contained information on dealing with a condition adverse to quality, when valve thrust values outside the design specification were found, they were not identified as a condition adverse to quality, and corrective actions were not promptly taken. This is contrary to the requirements of 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action" which requires that measures be established to assure that conditions adverse to quality are promptly identified and corrected, and, for significant conditions adverse to quality, that the cause of the condition is determined and corrective action taken to preclude repetition (237/249-92009-03c(DRP)).

The violations were considered to constitute a serious deficiency in the licensee's control of their quality assurance program in the area of motor operated valve testing. This is considered an apparent violation of 10 CFR Part 50, Appendix B, Criterion II "Quality Assurance Program", with the previously discussed failures to meet the requirements of Criteria III, V, and XVI as examples (237/249-92009-03(DRP)).

6. Diesel Generator Flooding

A. Background: The emergency diesel generators provide electrical power to safety related components in the event of a loss of offsite power. The diesels are cooled by water supplied via the diesel generator cooling water pumps, which are located in the cribhouse below ground level. If a cooling water pump fails, then the diesel generator will overheat within three to ten minutes, depending on load.

The diesel generator cooling water pumps are located at the bottom of the cribhouse at elevation 495' adjacent to the circulating water pumps. Approximately 30 seconds following a circulating water pump seal failure, the diesel generator cooling water pumps would be submerged. Based upon a calculation which does not take into account volume displaced by piping and equipment, the maximum time between the seal failure and complete submergence of the diesel generator cooling water junction boxes and transfer switch is approximately three minutes, providing power to the circulating water pump motors is not terminated before then. If power to the circulating water pump motors is removed before the three minutes, or if offsite power is lost concurrently with the seal failure, then the diesel generator cooling water pumps themselves would be submerged. The remainder of the cribhouse would flood only to the river level. As the junction boxes and transfer switch are located above the maximum river level, they would not be affected. As long as the pumps, and the power feeds at the pump motor were submergence qualified, the pumps would be available to provide cooling water to the emergency diesel generators.

B. Description of Events: In June 1972 Quad Cities had an internal flooding event due to circulating water boot seal failure. In August 1972, the Atomic Energy Commission (AEC) required the licensee to evaluate the potential for a similar failure at Dresden. The event to be evaluated was the catastrophic failure of a circulating water pump discharge expansion joint. The circulating pump motors were to be assumed unaffected by the flood, becasue they were above the maximum flood level, and to continue to operate, resulting in the continued discharge of water into the cribhouse pit at a predicted flow rate of approximately 250,000 gallons per minute. The loss of circulating water would cause a loss of condenser vacuum, which would result in turbine and reactor trip. Upon the reactor trip, offsite power was assumed to be lost, in accordance with design basis scenarios. In subsequent correspondence, the licensee noted that the diesel generator cooling water pumps were susceptible to internal flooding. The licensee committed to install submersible pumps with canned motors so that the pumps would not be lost during the internal flood. A maximum flood level to 517' elevation was discussed in the licensee's response to the AEC questions. The pumps were installed in 1973; however, electrical junction boxes located at approximately the 510' elevation were not waterproofed.

In November 1986, because of 10 CFR Part 50, Appendix R concerns, a transfer switch was installed for the 2/3 diesel generator cooling water pump. This transfer switch would select an appropriate electrical power feed for the pump in the event of a fire. As part of this modification, the cable to the pump motor at elevation 492' was replaced. The personnel responsible for the modification overlooked the requirement for the pumps and their power feeds to be submergence qualified. Therefore, when they replaced the cable, they failed to reseal the motor connections against water intrusion.

On several occasions in 1989, 1990, and 1991, contractor employees of the licensee identified to the licensee that the 1972 commitment to the AEC did not appear to have been met.

In June 1991, the power feeds to the Unit 3 diesel generator cooling water pump were sealed against water intrusion.

In December 1991, the licensee identified that the issue had not been handled in accordance with their procedural requirements and the power feeds to the Unit 2 and 2/3 diesel generator cooling water pumps were sealed. This included resealing the Unit 2/3 power feed that was unsealed in 1986.

In January 1992, the licensee identified that the transfer switch connections for the 2/3 pump had not been sealed in December. They were promptly sealed.

In April 1992, while going through the licensee's correspondence on this issue, the NRC questioned whether the electric cables were submergence qualified. The licensee determined that the cables were submergence qualified through a document review and operability determination.

Also in 1992, the licensee determined that the scenario of a condenser boot seal failure with a subsequent loss of offsite power was not within their design basis. They concluded that only design basis accidents, defined in Chapter 14 of the final safety analysis report, coupled with a coincident loss of offsite power, were within the design basis. They acknowledged that they had not met an NRC commitment, but concluded that the diesel generators would have been operable under all design basis scenarios. These conclusions overlooked the 1986 transfer switch modification to the Unit 2/3 cooling water pump, which resulted in the electrical power feed going into the motor at the 495' elevation not being resealed following cable replacement.

C. Analysis of Root Cause: The initial failure to seal the power feed boxes against water intrusion appeared to be due to an oversight as to the maximum flood level which could occur following a circulating water seal failure.

The 1986 failure to reseal the Unit 2/3 power feed at the pump motor following the transfer switch modification was due to overlooking the requirement for submergence qualification during the safety evaluation process. The safety evaluation failed to address the potential for water intrusion into the diesel generator cooling water pump resulting in loss of that pump with subsequent loss of the associated diesel generator. Because of the unsealed power feeds, the Unit 2/3 diesel generator would have been inoperable under following a circulating water line break with concurrent loss of offsite power.

The 1989-1991 failures to take corrective actions were due to the licensee assuming that the junction boxes and cables were submergence qualified, without verifying the assumption.

D. Assessment of Safety Significance: Betwen November 1986 and December 1991, a circulating water pump seal failure coupled with a concurrent loss of offsite power would have resulted in the Unit 2/3 diesel generator cooling water pump being inoperable due to water intrusion into the electrical feeds. The inoperability of the diesel generator cooling water pump would have resulted in the Unit 2/3 emergency diesel generators overheating when called upon. If a single failure of another diesel generator was assumed to occur, for any reason, then one unit could have been in a station blackout condition. This would have severely affected the plant's capability to achieve shutdown. The root cause of these feeds being unsealed was an inadequate safety evaluation which failed to account for pump submergence. This is an apparent violation of 10 CFR 50.59 (237/249-92009-04(DRP).

Additionally, the failure to protect the diesel generator cooling water pumps power feeds from flooding, as previously committed to, is a significant condition adverse to quality. The existence of this significant condition adverse to quality was first identified to the licensee in November 1989. However corrective actions were not taken until late 1991 or early 1992. The failure to promptly identify and correct the condition adverse to quality is an apparent violation of 10 CFR Part 50, Appendix B, Criterion XVI "Corrective Actions" (237/249-92009-05a(DRP)).

7. Failure to Follow Dresden Administrative Requirements

A. Background: The Dresden Administrative Procedures (DAPs) delineate requirements for performance of tasks throughout the plant. They are called out in Technical Specification (T/S) 6.3 as required procedures, and must be followed in order to show compliance with the technical specifications.

The NRC previously identified violations in the area of failing to follow DAPs as follows:

1. Between April 1 and August 30, 1990, the licensee failed to maintain the Control Rod Drive Accumulator High Water/Low Pressure Alarm Log in accordance with the requirements of DAP 9-12, "Procedural Adherence Deficiencies." Corrective actions for this violation were to:

   - Issue a weekly list of new and revised Dresden Operating Procedures and selected DAP procedures to all licensed operators to ensure that they are aware of procedure changes which altered their day-to-day routines.

   - Hold tailgate sessions on the requirements of DAP 9-12.

   - Develop a list and summary of 66 DAPs, and present them during tailgate meetings to all station personnel.

2. On December 27, 1990, the licensee committed to perform training on various DAPs as a corrective action to an escalated violation on failure to perform a safety evaluation required by 10 CFR 50.59. The wording used in the corrective actions to this violation was identical to that of the corrective actions for the violation above.

B. Description of Event: The corrective actions to these violations were ineffective as attested by the following recurrences:

1. On March 7, 1992, the licensee performed Dresden Operating Surveillance (DOS) 6600-03. During performance of the procedure, the 2/3 Diesel Generator vent fan and fuel oil transfer pump failed to transfer to Unit 3 power when expected. However, the licensee failed to complete DAP Form 9-11A, Procedural Comment Supplement, as required by DAP 9-11 "Procedure Usage and Adherence," when the failures occurred.

2. Also on March 7, 1992, during performance of the same procedure as above, several steps were performed locally by station personnel. The test leader, stationed in the control room, signed off on these steps without directly observing the steps being performed and without including the initials of the persons actually performing the steps as required by DAP 9-11.

3. On April 1, 1992, the standby liquid control storage tank air sparge inlet valve 3-1101-36 was identified to be open and unlocked although DOP 0040-M4 required it to be locked closed and independently verified in the locked closed position. When the valve was manipulated on March 20, 1992, personnel did not use either an approved procedure or outage checklist, nor was an operator in continuous attendance, as required by DAP 7-14, "Control and Criteria for Locked Equipment and Valves."

C. Analysis of Root Cause: During the inspections, personnel involved in the repeated violations repeatedly stated that they were not aware of the administrative requirements, and none of the personnel recalled training on administrative procedures. The following reasons for the lack of awareness were identified:

- The DAP summaries provided to the departments to be covered in the tailgate meetings were brief, and not all departments covered all the DAPs when the tailgates were held. Some department heads stated that DAP training was partially given, to those present at the time, and other department heads did not remember the extent of the training sessions.

- Documentation did not exist for a substantial portion of station personnel that should have received DAP training.

- Previously identified problems with procedural inadequacies and revision timeliness have discouraged personnel from using the DAPs.

D. Assessment of Safety Significance: While the individual recurrences of the failure to follow DAP requirements were relatively minor, the underlying cause, personnel not being aware of administrative requirements, is more significant. Additionally, the fact that corrective actions to previously issued violations, including one at Severity Level III, did not preclude repetition of personnel being unaware of the requirements gives rise to the concern that more serious recurrences could happen. The failure to adequately implement corrective actions for this violation is considered an example of an apparent violation of 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action" (237/249-92005-05b(DRP)).

8. Failure to Report Inadvertent Engineered Safety Features Actuations

A. Background: The NRC also previously issued violations for failing to make required notifications in accordance with the requirements of 10CFR 50.72. These previous violations were:

1. On December 8, 1990, the licensee failed to report an unplanned engineered safety feature (ESF) actuation in accordance with the requirements of 10 CFR 50.72. The corrective action was to issue a memorandum to operations personnel defining an ESF actuation as any unplanned or unknown occurrence involving the actuation of an ESF train, which resulted in the completion of the desired repositioning of any piece of equipment.

2. On July 4, 1991, the licensee again failed to report an unplanned ESF actuation. The NRC issued a violation of 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective

Action," for the ineffective corrective actions from the previous violation. In response to this violation, the licensee committed to:

- Provide training to the shift engineers (SE) and shift control room engineers (SCRE).
- Clarify the guidance provided to operations personnel. The clarified guidance defined an ESF actuation as: "Unplanned actuation of ESF systems or components thereof (e.g., valve movement, pump starts) are expected to be reported regardless of what caused the actuation, even if the actuation was unnecessary or was not directly initiated by ESF actuation signals".
- Develop a flow chart to aid in determining reportability requirements and provide it by early first quarter, 1992.
- Place a copy of NUREG 1022, Licensee Event Reporting System, in the control room to aid in reportability determinations.

B. Description of Event: The corrective actions to these violations were insufficient to prevent further failures to make required notifications as noted below:

1. On March 14, 1992, the high pressure coolant injection (HPCI) suction valve (MOV 3-2301-6) unexpectedly opened during the Unit 3 integrated leak rate test when high drywell pressure provided an ESF actuation signal to the HPCI system. Shift operations management failed to recognize the opening of the MOV as an unplanned ESF actuation, and did not report the event within four hours as required. After NRC inquiries into the event, operations management determined a 10CFR50.72 report was appropriate. The report was made on March 18, 1992.

2. On April 19, 1992, the Low Pressure Coolant Injection (LPCI) minimum flow valve (3-1501-13A) unexpectedly closed twice when cycling of valves 3-1501-38A and 3-1501-38B occurred during performance of DOS 1500-1, "LPCI Valve Operability Test," on Unit 3. Shift operations management failed to recognize the closings of the minimum flow valve as an unplanned ESF actuation, and did not report the events per 10CFR50.72(b)(2)(ii). Later management review of the events determined a 10CFR50.72 report was appropriate. The report was made on April 20, 1992.

C. Analysis of Root Cause: These corrective actions were either not effective or not implemented to preclude the third and fourth failures to report an ESF actuation. Specifically:

- For the third failure, the SE did not consider the event reportable because the operation of the MOV was not spurious and the intended function was accomplished, that is, the valve went open.

- For the fourth failure, the SE did not consider the event to be reportable because only the minimum flow valve closed and the SE thought that the LPCI suction and discharge valves were the only ESF components in that system.

- The flow chart was not available to the operations personnel to aid in ascertaining reportability requirements. Personnel involved in the development of the flow chart did not consider it developed enough to be released to shift personnel. A new target date of June 1, 1992, was selected by the licensee to accomplish the corrective action. However, NRC was not notified nor was a variance from the violation response commitment requested.

- NUREG 1022 was available in the control room; however, it was not used in making the reportability determination.

- The responsible SE and SCRE during the MOV actuation did not recall the NUREG 1022 training. Additionally, none of the shift personnel interviewed had utilized NUREG 1022 to aid in a reportability determination. (The inspectors confirmed training was provided.)

D. Assessment of Safety Significance: As discussed above, neither of the individual recurrences was of major safety significance. However the fact that two nearly identical repeats of previous violations occurred within a five week period indicates that the corrective actions to the previous violations failed to prevent recurrence. This is another example of an apparent violation of 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action" (249/92005-05c(DRP)).

9. Failure to Consider Systematic Evaluation Program Commitments in Performing Safety Evaluations

A. Background: On February 8, 1991, a temporary alteration (T/A) on the Unit 2 HPCI system provided an interface between class 1E electrical equipment and non-safety measuring and test equipment (M&TE). This was contrary to a systematic evaluation program (SEP) commitment to incorporate the electrical isolation philosophy of IEEE 384 and Regulatory Guide 1.75 for plant modifications whenever practical (1985 CECo letter from B. Rybak

to R. Bilbert (NRR)). The corrective action to the violation was to revise DAP 10-02, "10CFR50.59 Safety Evaluation/ Screening" to incorporate a safety evaluation screening review work sheet.

B. Description of Event: Although the DAP was revised, it was not used when on March 19, 1992, M&TE to monitor voltage was installed on the auxiliary compartment of ESF 4160 VAC Bus 34-1 under temporary alteration (T/A) III-7-92. The T/A provided an indirect interface between Class 1E electrical equipment and non-safety M&TE. The 10 CFR 50.59 safety evaluation did not address the probability or the consequences of malfunctioning M&TE or the bases for why the Class 1E circuit would be protected following a malfunction of the M&TE.

C. Analysis of Root Cause: Neither the safety evaluation preparer or reviewer used the checklist or were aware of the commitment to IEEE 384. The reasons for this are very similar to those in the examples of failure to follow the DAPs.

D. Assessment of Safety Significance: As with the other examples, the individual failure is not significant; however the failure of the corrective action program to prevent recurrence is significant. This is another example of an apparent violation of 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action" (249/92005-05d(DRP)).

10. Review of Inspection Report 50-237/91036(DRP)

Upon review of inspection report 50-237/91036(DRP) as part of ensuring that all aspects of the unresolved item had been closed, the following areas were found to either be misleading or to contain violations of minor safety significance:

A. Section 4.b: Failure to Document Personnel Qualification

The inspection report stated that "the licensee had not maintained records documenting station and corporate personnel qualification to NOD-MA.1 requirements." During further review, it was found that station records complied with the Commonwealth Edison Nuclear Quality Assurance Manual requirements. Procedure NQA-1 of this manual requires that records of the implementation of indoctrination and training are to be maintained in the form of attendance sheets, training logs, or personnel records. Review of site personnel training records for site personnel showed that the required courses had been taken and documentation was available. This section of Inspection Report 50-237/91036 should be amended to read "the licensee had not maintained records documenting corporate personnel qualification to NOD-MA.1 requirements."

Corporate records were not available, as identified by the licensee in Quality Assurance/ Nuclear Safety Audit CE-91-04. The licensee determined that required training had been performed but

documentation was unavailable. The licensee committed to reestablish corporate training records to show compliance with the NOD requirements. The inspectors verified that this commitment was met. The failure to retain training records is a violation of 10 CFR Part 50, Appendix B, Criterion II, "Quality Assurance Program." However, the violation was licensee identified, is of minor safety significance, and corrective actions were initiated. Therefore this violation meets the criteria for a noncited violation as discussed in 10 CFR Part 2, Appendix C, Section V.G.1 (1991).

B. Section 4.c: Use of an non-safety-related consulting service

The inspection report identified that safety-related work (consulting services as to the proper zero point to be used to determine the as-left torque switch setting) was performed under a non-safety-related contract. This should have been characterized as a violation of the requirements of 10 CFR Part 50, Appendix B, Criterion IV "Procurement Document Control" which requires, in part, that measures be established that applicable regulatory requirements necessary to assure adequate quality are suitable included or referenced in the documents for procurement of material, equipment, and services. However, no violation will be cited because the designation of the contract as non-safety-related had no bearing on the torque switch settings, the classification appeared to be an isolated example, and corrective actions were initiated. Therefore the criteria for a noncited violation under 10 CFR Part 2, Appendix C, Section V.A have been fulfilled, and no violation will be issued.

C. Section 4.d: Test Procedure Quality

The inspection report stated that "Neither the work package nor the testing procedure delineated any quantitative or qualitative acceptance criteria related to the VOTES diagnostic, the as-found or as-left torque switch settings or the valve thrust windows."

Further review found this statement to be partially incorrect. The test package did contain the target thrust values to be used, along with a letter from corporate engineering specifying that setting the thrust values outside the target values required corporate approval. Although they were not specifically identified as acceptance criteria, the personnel performing the testing regarded them as such. Additionally, qualitative test acceptance criteria (that the valve stroke open and closed) were included. This section of Inspection Report 50-237/91036 should be amended to read "Although the work package and the testing procedure did not contain VOTES diagnostic quantitative or qualitative acceptance criteria, the corporate engineering approved target thrust windows were included with the package and were considered to be acceptance criteria by the testing personnel."

Section 4.d goes on to state that "The MOV coordinator's concerns about selecting the zero point on the valve and use of industry expert services were not documented."

The MOV coordinator should have documented his concerns on form DAP 9-11A, Procedural Comment Supplement, as required by DAP 9-11 "Procedure Usage and Adherence," when the failures occurred. He failed to do so. This is an example of failure to follow Dresden Administrative Procedures which occurred in December 1990. The failure of the MOV coordinator to follow the DAP requirements and document the concerns with the recirculation discharge valve would not have prevented the valve failure. The violations identifying problems with failing to follow DAPs were issued in November and December 1990. Therefore, corrective actions to these violations would not have prevented this violation. Based on this, the criteria of 10 CFR Part 2, Appendix C, Section V.A were satisfied, and no violation against 10 CFR Part 50, Appendix B, Criterion V will be cited.

11. Violations for Which a Notice of Violation Will Not Be Issued

The NRC uses the Notice of Violation as a standard method for formalizing the existence of a violation of a legally binding requirement. However, because the NRC wants to encourage and support licensee's initiatives for self-identification and correction of problems, the NRC will not generally issue a Notice of Violation for a violation that meets the requirements set forth in 10 CFR Part 2, Appendix C, Sections V.A (1991). Violations of regulatory requirements identified during the inspection for which a Notice of Violation will not be issued are discussed in Section 10, above.

11. Exit Interview

The inspectors met with licensee representatives (denoted in section 1) throughout the inspection period. Exit meetings were held on April 16 and May 1, 1992, to summarize the scope and apparent findings of the inspection activities. The inspectors also discussed the likely informational content of the inspection report with regards to documents or processes reviewed by the inspectors during the inspection. The licensee did not identify any such documents or processes as proprietary.