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SUBJECT: Responds to NRC 901109 ltr re violations noted in Insp Repts
50-315/90-201 & 50-316/90-201.

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AEP:NRC:1125H

Donald C. Cook Nuclear Plant Units 1 and 2
Docket Nos. 50-315 and 50-316
License Nos. DPR-58 and DPR-74
INSPECTION REPORTS 50-315/90201 (DRS) AND 50-316/90201 (DRS)

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, D.C. 20555

Attn: A. B. Davis

December 7, 1990

Dear Mr. Davis:

This letter is in response to Mr. T. O. Martin's letter of November 9, 1990, which forwarded the results of the NRC staff evaluation of items identified in the subject NRC inspection reports regarding a safety system functional inspection (SSFI) of the essential service water (ESW) system at the Donald C. Cook Nuclear Plant. Mr. Martin's letter identifies four Severity Level IV violations involving various criteria of 10 CFR 50 Appendix B. The letter also requested a response on three items identified as unresolved items in the subject inspection reports.

The actions taken to correct the specific violations are addressed in Attachment 1 to this letter. Our response for the unresolved items is contained in Attachment 2.

This document has been prepared following Corporate procedures which incorporate a reasonable set of controls to ensure its accuracy and completeness prior to signature by the undersigned.

Sincerely,

M. P. Alexich
Vice President

ldp

IE01

cc: D. H. Williams, Jr.
A. A. Blind
J. R. Padgett
G. Charnoff
A. B. Davis - Region III
NRC Resident Inspector - Bridgman
NFEM Section Chief

ATTACHMENT 1 TO AEP:NRC:1125H

RESPONSE TO NRC NOTICE OF VIOLATION

NRC Violation 1

"10 CFR Part 50, Appendix B, Criterion III, states, in part, that design changes, including field changes, shall be subject to design control measures commensurate with those applied to the original design.

Contrary to the above, in June 1989, the licensee replaced a valve in the component cooling water heat exchanger inlet line manufactured by Centerline Company with a valve manufactured by the Henry Pratt Company under maintenance job order 728163 without engineering review or evaluation. Further, at least one of the pipe flange bolt holes was enlarged to achieve alignment with the new valve without engineering evaluation."

Response to Violation

Installation of the Henry Pratt Company valve in place of the Centerline Company valve was identified as necessary in 1987. This pre-dates our minor modification process, which is the mechanism that would be used for similar modifications today. By the time work commenced in 1989, the minor modification procedure requiring complete documentation was in place. However, since the process of valve replacement had begun before the minor modification mechanism was in place, the personnel involved did not recognize the need to implement the change as a minor modification, and work was performed based on verbal approvals which had been the previous practice.

Corrective Actions Taken and Results Achieved

Subsequent to identification of the deficiency, reviews and evaluations were undertaken. The change in brand of valves was subsequently determined to be consistent with current minor modification practices and would have no adverse impact. A similar conclusion was reached regarding the enlargement of pipe flange bolt holes. Given a worst-case scenario, all holes enlarged, it was concluded that there would be no adverse effect on the flange or the system.

Corrective Actions to be Taken to Avoid Further Violations

All similar changes to the plant are currently handled via the design change process and assigned to a maintenance engineer whose task it is to ensure all necessary documentation is complete and all approvals have been received from other plant departments and corporate engineering. Having an engineer as a control point for processing of minor modifications provides an adequate level of confidence that necessary reviews are performed and procedures followed.

Date When Full Compliance Will Be Achieved

Required evaluations of the installation of the Henry Pratt valve and the effect of enlargement of pipe flange bolt holes were completed on November 14, 1990.

NRC Violation 2

"10 CFR 50, Appendix B, Criterion V, states, in part, that activities affecting quality shall be prescribed and accomplished by documented instructions, procedures, or drawings and that the instructions, procedures, or drawings shall include appropriate quantitative or qualitative acceptance criteria. Further, D. C. Cook Technical Specifications Section 6.8 requires that written procedures be established, implemented, and maintained, for activities such as those listed in Appendix A of Regulatory Guide 1.33, involving procedure adherence, temporary change method, procedure review and approval, and review procedures.

Contrary to the above, the following violations of this requirement were identified:

NRC Violation 2.a

Licensee procedure PMI.2010, "Plant Manager and Department Head Instructions, Procedures, and Associated Indexes", required that a procedure that had not been reviewed by its biennial review date should be so annotated and not be used prior to this review. Procedure 1THP6030.STP.068, "Essential Service Water Liquid Process Monitor (R-20) Surveillance Test", was performed on June 21, 1990, with its biennial review overdue by more than five months (last reviewed in January 1988). The procedure was not annotated to indicate the overdue status."

<u>Response to Violation</u>

Subsequent to the inspection, a search of computerized records indicated that the procedure had received its biennial review on March 18, 1989. Thus, the procedure did receive the biennial review as required prior to its use on June 21, 1990.

<u>Corrective Action and Results Achieved</u>

Not applicable.

<u>Corrective Action to be Taken to Avoid Further Violation</u>

Not applicable.

<u>Date When Full Compliance Will Be Achieved</u>

Not applicable.

NRC Violation 2.b

"Licensee procedure PMI.2010 required that any procedure designated with a double asterisk be present and used at the job site. On June 13, 1990, procedure 12MHP5021.032.025, "Emergency Diesel Engine Timing and Balancing," which was designated with a double asterisk and provided instructions for adjusting exhaust temperatures and for obtaining measurements and adjustments in combustion and compression pressures, was not available at the job site nor used when taking cylinder pressure measurements on the 1AB Emergency Diesel Generator."

Response to Violation

PMI-2010, Step 3.1.1, states, in part

> Personnel in the field need only those portions of the procedure which apply to their specific job, e.g., valve lineups, computation table, performance data sheets, etc. in their possession.

On June 13, workers used the subject procedure to perform timing and balancing of the 1AB diesel. The workers signed off on steps which were completed and indicated "N/A" on parts which did not apply. The procedure steps which were necessary were completed, and the procedure, which had been at the job site in its entirety, was removed from the area. Subsequently, engineers requested measurement of cylinder temperatures. Sections of the procedure applicable to this task were at the job site during this testing. This was consistent with the PMI-2010, Step 3.1.1.

(The Notice of Violation refers to measurement of cylinder pressure as opposed to temperature. The subject procedure would not have been at the job site for taking measurements of cylinder pressure as there is no procedural direction contained in the document regarding that activity.)

Corrective Actions Taken and Results Achieved

Not applicable.

Corrective Actions to be Taken to Avoid Further Violation

Not applicable.

Date When Full Compliance Will Be Achieved

Not applicable.

NRC Violation 2.c

"Section 5.0 of licensee procedure 12PMP2030.VICS.001, "Control of Vendor Documents," required that all vendor information, including bulletins, letters, and vendor manuals or revisions, be processed and controlled to ensure their proper availability and use under the licensee's document control system. Procedure 12MHP50211.032.026, "Emergency Diesel Engine Inlet and Exhaust Hydraulic Valve Lifters Inspection and Testing," provided an acceptance criteria that was contrary to the vendor's instruction manual. The licensee produced a copy of a vendor letter dated March 19, 1984, which authorized the criteria the licensee was using. However, this letter was not included in the controlled vendor information/manual file."

Response to Violation

The VICS for the Cook Nuclear Plant was not formally established until March 21, 1985. Thus, VICS did not exist at the time the March 19, 1984, letter was received.

Although the vendor letter was not processed through VICS or any other formal system, the information contained therein was incorporated into the appropriate plant procedure (12MHP5021.032.001E) eight days after the letter was received (approval date March 27, 1984). Note that the procedure number has since been changed to 12MHP5021.032.026.

Even though the VICS program was not in place at the time the referenced letter was received, the vendor information was incorporated into plant procedures in a timely fashion.

Corrective Action Taken and the Results Achieved

Even though no violation of the VICS procedure occurred, a review of plant files associated with the emergency diesel generators (EDGs) was performed by the plant system engineer to look for other vendor letters. A similar review of corporate files associated with the EDGs was made by the AEPSC cognizant engineer. These reviews revealed several vendor letters which had been transmitted to Indiana Michigan Power/AEPSC prior to establishment of VICS. Review of the letters and plant procedures indicates that the procedures reflect the information in the vendor letters.

Corrective Actions Taken to Avoid Further Violation

Not applicable.

Date When Full Compliance Will Be Achieved

Not applicable.

NRC Violation 2.d

"Licensee procedure PMI.7030, "Condition Reports and Plant Reporting", required that a condition report be issued for nonconforming conditions. No condition report was issued for the following surveillance deficiencies, nor was any basis given for not issuing the report:

2.d.1 The specified counts per minute (CPM) range was exceeded for surveillances 1THP4030STP.068 (step 7.12.3.2) on October 10, 1989, July 25, 1989, and January 15, 1989; 1THP4030STP.075 (step 7.12.3.2) on May 8, 1989; and 2THP4030STP.175 (step 7.12.3.2) on July 7, 1989.

2.d.2 The required CPM values were not recorded for surveillances 1THP4030STP.068 (Steps 7.12.3.2 and 7.11) on February 27, 1989; 2THP4030STP.168 (step 7.12.2.2) on June 23, 1989; and 1THP4030STP.075 (step 7.1.2.3.2) on April 2, 1990."

Response to Violation

These items cite two concerns raised during reviews of four procedures utilized, in part, to verify proper operation of radiation monitor check sources. The check source is a fixed radiation source that is electromechanically positioned into and out of a window to the detector to check the operation of the detector and the electronic devices of the monitor. The test is typically a go/no-go test as initial background radiation and half-life strength of the source will change with plant conditions and time. The procedures did not fully explain this, however, and requested a one decade increase in indication (unless the background reading was already greater than 50% of scale). Review of the past data indicates that the check source had been functioning properly, although exceeding the procedural cpm range requirement on occasion due to a reduced background. The technicians and their supervisors were knowledgeable and understood the requirement of the step and recognized the term to indicate a substantial increase was needed. They did not see it as a limit to the expected increase. Although the procedures required entry of the cpm values, in some cases the cpm was entered on the blank in the data sheet while at other times the technician's initials were entered to indicate the required response occurred.

Corrective Actions Taken and Results Achieved

Condition reports were written addressing these issues upon receipt of the SSFI Inspection Report. Investigation of the conditions determined that the procedures should be changed to require conditions actually expected to result from a successful check

source test, and to more clearly indicate where cpm values must be recorded. Changes were made to all four procedures on June 27, 1990.

Corrective Action Taken to Avoid Further Violation

As discussed above, the subject procedures were revised on June 27, 1990, to require conditions actually expected to result from a successful check source test, and to more clearly indicate where cpm values must be recorded. Additionally, the Maintenance Department developed and has been implementing training for its personnel in procedural compliance. Training sessions for maintenance personnel on procedural compliance will be completed June 30, 1991.

Date When Full Compliance Will Be Achieved

Procedures were changed on June 27, 1990.

NRC Violation 2.d.3

"The low level alarm setpoint (step 7.10) for surveillance 1THP4030STP.068 was exceeded by a factor of ten on March 30, 1990."

Response to Violation

This item cites an incident where the recorded low radiation alarm setpoint (indicating monitor failure) exceeded the requirement by a factor of ten. Our review of the data sheet entry, as well as previous surveillance data, indicates that the entry of "1×10^{-2}" was in error. Subsequent testing revealed, as was the case prior to the event date, that the reading was actually "1×10^{-1}". No history of setpoint drift was identified. Although the surveillance had been reviewed by a supervisor, the transcription error was accidentally overlooked and therefore a condition report was not initiated.

Corrective Action Taken and Results Achieved

A condition report was issued which addressed this concern upon receipt of the SSFI Inspection Report. As discussed above, the investigation determined that the recorded value was in error.

Corrective Action to be Taken to Avoid Further Violation

The entry error was reviewed with the personnel involved with the error. Additionally, the Maintenance Department developed and has been implementing training for its personnel in procedural compliance to minimize the potential for recurrence.

Date When Full Compliance Will Be Achieved

Resolution of the data entry error was completed on October 22, 1990. Training sessions for maintenance personnel on procedural compliance will be completed June 30, 1991.

NRC Violation 2.d.4

"The flow through valve 1-ESW-113 during surveillance 1OHP4030STP.022E (steps 8.9.2 and 8.9.3) did not exceed the required 610 gallons per minute."

Response to Violation

This item identifies that the recorded essential service water flow rate was less than required during a surveillance on December 18, 1989. The data sheet for that procedure revealed that the flow rate was identified as 600 gpm. A rate of 610 gpm is the minimum required to demonstrate operability per the procedure. Data obtained subsequent to the December 18, 1989, surveillance was reviewed. In all cases the surveillance identified flow rates in excess of 740 gpm with the majority found at 800 gpm. We have determined that the entry of 600 gpm on December 18, 1989, was an error and should have been, as substantiated in subsequent tests, 800 gpm. The reviews and investigation concluded that the operability of the east ESW pump had been in compliance with the required flow rate. Although the surveillance had been reviewed by a supervisor, the transcription error was accidentally overlooked and therefore a condition report was not initiated.

Corrective Action Taken and Results Achieved

A condition report was issued which addressed this concern upon receipt of the SSFI Inspection Report. As discussed above, the investigation determined that the recorded value was in error.

Corrective Action to be Taken to Avoid Further Violation

A self-checking program is being developed by the Operations Department with implementation scheduled by November 15, 1991. This program was committed to the NRC in Licensee Event Report 315/90-004 Rev. 2. The self-checking program will involve conducting an awareness campaign on the attributes of self-checking/attention to detail, the benefits which can be derived, and management's expectations. Existing standards will be reiterated and new standards developed if needed.

Date When Full Compliance Will Be Achieved

An investigation which determined that there was not an operability problem with the east ESW pump was completed November 15, 1990. The self-checking program being developed by the Operations Department will be implemented by November 15, 1991.

NRC Violation 2.d.5

"The required independent verification was not performed on June 21, 1990, for surveillance 1THP6030.IMP.012. Further, four change sheets were needed before the surveillance procedure could be accomplished."

Response to Violation

A condition report addressing these concerns was issued upon receipt of the SSFI Inspection Report. The investigation of the finding regarding independent verification was completed on August 23, 1990. Plant procedures require that the verifier not be in the area during the performance of the work activity. The conclusion was that the verifying individual had not observed switch manipulation when done by another worker. He had turned away when the switch was moved and had turned back to observe its position. Although time and space separation could have been improved in this instance, the intent of the independent verification requirement was met in that the verifier had not witnessed the performance of the work.

The investigation of the finding regarding the need for change sheets was completed on October 22, 1990. The procedure is used as an aid to troubleshooting problems found during surveillance or normal repair activities of Westinghouse Electric Corporation radiation monitors. Our investigation concluded that the troubleshooting could have been completed without the change sheets with desired results. The change sheets inserted requirements for switch positions to be used in troubleshooting and calibration. Had those positions not been indicated procedurally, technicians would still have had to place switches in certain modes to complete their work. Troubleshooting and calibrations performed without use of the four change sheets would have accomplished the desired level of accuracy. The change sheets were added as procedure enhancements.

Corrective Action Taken and Results Achieved

Not applicable.

Corrective Action Taken to Avoid Further Violation

Not applicable.

Date When Full Compliance Will Be Achieved

Not applicable.

NRC Violation 2.e

"Licensee procedure NEP 6.4, "Calculations", required that an independent review of calculations be completed and documented on a verification checklist. It also required that the cognizant section manager approve the verification checklist.

Contrary to the above, the verification checklists for the following calculations were improperly completed in that relevant review items were noted as "not applicable" or were not addressed:

- o HXP900628AF dated July 6, 1990
- o HXP900627AF dated July 6, 1990
- o HXP900629AF dated July 9, 1990."

In addition, the verification checklist for calculation HXP900613.AF was not approved by the cognizant section manager.

<u>Response to Violation</u>

In accordance with instructions received from Ms. P. Rescheske of Region III on November 20, 1990, our response to this violation addresses the general concerns with design verification identified during the SSFI and not the specific items listed in the Notice of Violation. The general concerns included:

1) instances of failure to adhere to procedures which required completion of a verification checklist for design control activities,

2) instances of failure to adequately document the bases for responses to verification checklist questions, and

3) cases of lack of attention to detail.

These concerns were mainly associated with implementation of procedures developed within the Nuclear Engineering Department in 1989, after the department's formation which merged electrical and mechanical engineering disciplines in 1988.

It should be stressed, however, that the SSFI team did not link any of these design verification issues with deficient design outputs. Although there were a number of technical issues raised in the SSFI Inspection Report, these issues are more appropriately labeled technical disagreements between inspector and licensee than instances of "poor designs" which resulted from inadequate design verifications. Specific technical issues linked with design verification were discussed during a meeting with the NRC held on October 4, 1990, and were summarized in our letter AEP:NRC:1138 dated October 23, 1990.

Corrective Actions Taken and Results Achieved

In order to assess the overall effectiveness of implementation of new design control procedures, a historical review has been initiated of design outputs produced to new procedures within the Nuclear Engineering Department and the Nuclear Design Group of the Design Department. (Although there were no specific SSFI design verification findings directed at the Nuclear Design Group's activities, the organization's activities were included in the historical review scope since the group was also formed relatively recently and is operating under a relatively new set of implementing procedures, similar to the Nuclear Engineering Department.) This historical review in both organizations began in the fourth quarter of 1990. Due to the number of design outputs to be included in this review, which will be conducted in parallel with other ongoing activities, this effort is scheduled for completion by the end of 1991. Significant safety-related deficiencies discovered during this review effort will be resolved by correcting the associated output document.

Corrective Actions Taken to Avoid Further Violations

Shortly following the SSFI, on August 20 and 22, 1990, training sessions were administered to familiarize engineering and design personnel with the design verification issues noted during the SSFI. Specific deficiencies noted during the inspection were used as training examples. This training was completed prior to receipt of the SSFI report.

On September 21, 1990, the AEPSC Senior Executive Vice President - Engineering and Construction, reissued an existing policy statement on the requirement for procedural adherence. This policy statement was disseminated to AEPSC and Cook Nuclear Plant personnel involved in implementing the Quality Assurance Program for the plant with emphasis on the fact that an employee's performance would be rated based on how effectively and completely the employee complies with procedures governing his/her work. This policy statement was also added to our General Procedures and Plant Manager's Instruction Manuals to ensure availability to the involved individuals.

Additionally, subject training is being developed in the area of design control to ensure that personnel understand procedural requirements for assigned tasks. Training within the Nuclear Design Group is underway. Development of lesson plans is underway within the Nuclear Engineering Department, and training will commence in early 1991. This initial cycle of training will be completed in both departments in 1991.

Finally, Quality Review Teams (QRTs) have been formed in both the Nuclear Engineering Department and the Nuclear Design Group to review a sample of future design output documents. This review is intended to ensure that outputs are technically adequate and procedurally correct. QRT activity commenced in the fourth quarter of 1990, and will continue until it has been determined that these supplementary reviews are no longer required.

Date When Full Compliance Will Be Achieved

As described above, corrective actions will be completed and full compliance achieved by December 31, 1991.

NRC Violation 3

"10 CFR Part 50, Appendix B, Criterion VI states, in part, that measures shall be established to control the issuance of documents, such as instructions, procedures, and drawings, including changes thereto, which prescribe activities effecting quality. Further, licensee implementing procedure PMI.2030 states, in part, that controlled documents shall be filed in a timely manner . . . . . Superseded documents are to be destroyed.

Contrary to the above, six controlled aperture cards in the maintenance library were not the latest revision (PS2-94208-4; PS2-94209-15; PSI-94208-14; PS2-94209-9; PSI-94209-9; and PS2-94206-4), and for 1094202-14-17 had both the current and outdated revisions in the file."

Response to Violation

A condition report covering this concern was issued upon receipt of the SSFI Inspection Report. At the time of our investigation, all of the drawing problems cited in the Notice of Violation had been corrected. One aperture card problem cited in the inspection report (but not appearing in the Notice of Violation) was found. For this drawing, the correct revision was in the drawer but an earlier revision had not been pulled.

Corrective Action Taken and Results Achieved

The files were checked to ensure the drawing problems noted by the NRC were corrected. The one drawing error found during this check was corrected. It appears from our investigation that current file maintenance practices have largely corrected problems with the file. (A new clerk was assigned to the file system in August 1990). Our QA organization will conduct a surveillance of the file by January 31, 1991, to ensure the file system is working adequately.

Corrective Action Taken to Avoid Further Violation

As discussed above, we believe our current file maintenance practices have corrected the problems with the file. This will be confirmed by a surveillance which will be conducted by our QA organization by January 31, 1991.

Date When Full Compliance Was Achieved

Full compliance was achieved on October 24, 1990, when the last aperture card known to be incorrect was removed from the files and destroyed.

NRC Violation 4

"10 CFR Part 50, Appendix B, Criterion XI states, in part, that a test program shall be established to assure that all testing required to demonstrate that structures, systems, and components will perform satisfactorily in service is identified and performed in accordance with written test procedures which incorporate the requirements and acceptance limits contained in applicable design documents.

Contrary to the above, station batteries 2AB, 2CD, and 1AB emergency load discharge (service) tests were being performed to a load profile 35 to 65 percent below the actual load profile calculated by the engineering staff in 1984-85. As a result, these tests did not confirm that the batteries could perform their emergency function."

Response to Violation

Sizing calculations were performed and are documented in RFCs 1-2764, 2-2784, and 2-2791. The calculations were performed in accordance with the manufacturer's battery sizing worksheets. The worksheets are similar to those of IEEE 485 in that they both require temperature correction, design margin, and aging factors. Using these sizing worksheets, a ampere hour battery of sufficient size to meet the duty cycle was chosen. These calculations indicate that the batteries can supply the emergency loads. Capacity and performance tests were performed on installation and are routinely performed on station batteries every 60 months. This demonstrates the battery capacity and detects any changes in the capacity.

Prior to May 1989, the Technical Specifications did not require a service test as defined in IEEE-450. Emergency loads were identified and durations specified during the 18-month surveillance test as specified in the Technical Specifications. Following discussions with INPO, we reviewed our battery testing methods and determined that a service test more in line with IEEE-450 could be performed using a simulated load established by a computerized battery load testing device. Because the Technical Specifications did not recognize the use of simulated loads, a Technical Specification change was necessary to use the simulated loads to satisfy Technical Specification requirements. The required Technical Specification change was received from the NRC in May 1989. This was during the Unit 1 outage, in time to permit only the testing of the 1CD battery with the new method. The 1AB battery had been tested earlier in the outage to the previous Technical Specification requirements.

As of this date, the safety-related batteries have been tested using the new equipment and they have demonstrated the capacity to supply the emergency loads.

In conclusion, testing was performed in accordance with Technical Specification requirements in place at the time of the testing. We agree that the IEEE-450 method of testing using a computerized load testing device is more rigorous and had already taken the appropriate engineering and licensing steps to incorporate the service testing into our surveillances.

Corrective Actions Taken and the Results Achieved

Not applicable.

Corrective Actions Taken to Avoid Further Violations

Not applicable.

Date When Full Compliance Will Be Achieved

Not applicable.

ATTACHMENT 2 TO AEP:NRC:1125H

RESPONSE TO UNRESOLVED ITEMS

UNRESOLVED ITEM 90-20-03

FINDING TITLE: Inadequate Terminal Voltage at Class 1E Inverter Terminals

DESCRIPTION OF CONDITION:

"The team reviewed specifications for Class 1E 250 volts dc to 120 volts ac instrument power inverters and noted that the inverters were qualified to operate at a minimum of 210 volts dc at their input terminals. Because the end-of-life (EOL) voltage at the station battery terminals was calculated to be 210 volts, inverter terminal voltage during battery EOL would always be less than 210 volts due to feeder voltage drop. Therefore, the inverter would not be operational. This condition could result in an inadequate power supply to plant instrumentation during a loss of ac power supply. During the inspection the licensee initiated actions to requalify these inverters for a minimum input voltage of 200 volts dc."

Response:

The 210 VDC value is the 8-hour end of charge, worst case battery terminal voltage. The CRID inverters are required for three hours per the Final Safety Analysis Report and Technical Specifications. Based on five-year station capacity tests, the battery terminal voltage is approximately 228 volts dc at three hours.

Under the worst-case scenario for station blackout, the CRID inverters must supply power for a duration of four hours. Again, based on the five-year capacity tests, the station batteries have a terminal voltage of some 226 volts dc.

The voltage drop to the CRID inverters has been checked for the 1CD station battery most severe case. These voltage calculations show a maximum cable feeder drop of 4-5 volts dc at the three-four hour points based upon conservative assumptions.

Thus the minimum available voltage at the CRID inverters for the most severe case is over 220 volts dc at the end of both three and four hours. This level is sufficiently above the minimum equipment rating of 210 volts dc that no further qualification of the CRID inverters is considered necessary.

UNRESOLVED ITEM 90-201-04

FINDING TITLE: Inadequate Terminal Voltage at Steam-Driven AFW Pump Feedwater Inlet Valve

DESCRIPTION OF CONDITION:

"The voltage drop calculation for the dc power feed cable to the steam-driven AFW pump feedwater inlet valve motor indicated that the worst-case terminal voltage at the motor was 178 Vdc. This condition could occur during a loss of ac power, with the station batteries at their end-of-life condition. The team could not determine if the 178 Vdc was sufficient for the valve to perform its required design function which is to control AFW flow. The team verified that vendor specification sheets of the valve only provided a single value of terminal voltage equal to 250 Vdc. Under these conditions the operability of the subject valve could not be verified."

Response:

The NRC inspection team reviewed the circuit parameters for the AFW pump inlet valve motor. The inspectors determined that the motor voltage terminal was 178 volts dc for the worst-case condition and questioned if this voltage was adequate. The acceptability of the circuit had originally been established by following a procedure described in the manufacturer's (Limitorque) maintenance update letter. This procedure required that the inrush circuit be calculated and that it be a multiple of the full load current. For this type of circuit the motor's terminal voltage and the inrush current are proportional and when the inrush current was found acceptable it was expected that the terminal voltage would also be acceptable. However, this procedure did not discuss a minimum acceptable voltage, and thus did not satisfy the NRC inspection team that the motor voltage was acceptable.

Following the inspection, it was decided to evaluate the motor voltage of 178 volts dc by a different method. The motor torque output was calculated at this reduced voltage against the required torque to operate the valve. For the types of motors involved, the output torque is proportional to the motor terminal voltage and was derated by the ratio of rated voltage to actual voltage. Other factors included were the valve operator gear efficiency and the gear ratio.

Calculation for this circuit showed that the motor produced an opening torque of 29.2 ft-lbs at 178 volts dc as compared to the 7 ft-lbs required to operate the valve. The motor voltage at 178 volts dc is therefore acceptable for this application.

It should be noted that current Nuclear Engineering Department procedures require verification that electrical equipment will develop sufficient electromotive force or torque during initial starting or actuating conditions with consideration given to the cable voltage drop due to inrush or locked rotor current.

UNRESOLVED ITEM 90-201-09

FINDING TITLE: Lack of Inclusion of Certain ESW Check Valves Into IST Program

DESCRIPTION OF CONDITION:

"Check valves in the ESW system (ESW-111 to -114, -141 to -144, -101E, -101W, -102E, and -102W) are required to perform a reverse flow closure function for certain scenarios where backflow may occur through the cross-connect valves between unit headers. However, testing of this function was not included in the licensee's IST program."

(The following is excerpted from the SSFI Inspection Report, and is included for clarity.)

> "These valves perform a safety function in the checked position for certain scenarios. The four check valves on each diesel generator cooling supply are each in series with a motor-operated valve. The four motor-operated valves open simultaneously on a diesel generator start signal and, in so doing, interconnect the two ESW headers of a single unit. Therefore, should an ESW pump fail or a line break occur upstream of the diesel generator branch stop valves, check valves ESW-111 to -114 and ESW-141 to -144 would be called upon to perform a checking function.
>
> Likewise, check valves ESW-101E, -101W, -102E, and -102W would perform a checking function should an ESW pump become idle. If a pump becomes idle, the standby pump on the same header will start. Because the motor-operated valve on the discharge of the idle pump (in series with the check valve) and the four header crosstie valves would be open, the check valve would be required to close so that the operating pump on that header does not feed the idle pump rather than the system loads. This scenario, in effect, leads to two lost pumps. Closure of the motor-operated discharge valves to terminate the backflow would require operator action.
>
> Section XI of the ASME Code requires that check valves of this type, which perform a safety function in the closed position, be tested in a manner which proves that the disk travels to the seat promptly on cessation or reversal of flow. This requirement for Category C check valves (valves that are self-actuated in response to a system requirement) had been reiterated in Generic Letter 89-04, "Guidance on Developing Acceptable Inservice Testing Programs." The licensee's IST program, as currently written, did not provide for testing this function for the 12 valves noted above."

Response:

The Donald C. Cook Nuclear Plant Inservice Testing (IST) Valve Test Program currently includes the above listed valves as normally closed valves which open during performance of their safety function (i.e., to pass flow on ESW actuation). These valves are exercised (full stroke) quarterly per Section XI, Paragraph IWV-3521.

Investigation of this SSFI finding indicated that under certain scenarios, these check valves would also be required to close in the reverse flow direction. We, therefore, agree with this SSFI finding that the check valves' safety function should be expanded to include closure in the reverse flow direction.

Although verification of the close function is not currently reflected in the IST Valve Test Program, check valves ESW-111 to -114 and ESW-141 to -144 are each disassembled and inspected on a refueling outage basis as required by our commitments to IE Bulletin 83-03, "Check Valve Failures in Raw Water Cooling Systems of Diesel Generators" (i.e., each valve is disassembled and inspected at approximately 18 month intervals).

Check valves in the INPO SOER 86-03 (Check Valve Failures or Degradation) program, which includes ESW-101E, -101W, -102E, and -102W, are included in a population of check valves for which a sample are disassembled and inspected at refueling outage frequency.

These disassembly and inspection activities provide assurance of the check valves' condition and ability to function.

A revision to the IST program will be submitted which addresses the safety-related backflow function for these valves. This revision of the IST Valve Test Program is tentatively scheduled to be submitted by April 9, 1991.

In addition, a review of the IST Valve Test Program will be performed for those check valves in the program not currently assigned a backflow checking safety function. This review will include other pump discharge check valves on parallel pumps, main feedwater header check valves, check valves in steam supply lines to the turbine driven AFW pumps, and CVCS volume control tank outlet check valves. Changes to the IST Valve Program deemed necessary as a result of this review will be included in the proposed revision to the program tentatively scheduled to be submitted by April 9, 1991.