Commonwealth Edison
1400 Opus Place
Downers Grove, Illinois 60515

August 14, 1991

U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Attention: Document Control

Subject: Dresden Nuclear Power Station Units 2 and 3
Response to Notice of Violation Associated with
Inspection Report 50-237/91016; 50-249/91015
NRC Docket Numbers 50-237 and 50-249

References: E.G. Greenman letter to Cordell Reed dated
July 15, 1991 transmitting NRC Inspection
Report 50-237/91016; 50-249/91015

Dear Sirs:

Enclosed are Commonwealth Edison Company's (CECo) responses to the Notice of Violation (NOV) and to the request for additional information on a separate event. One NOV involved a failure to establish an adequate calibration program. That violation is acknowledged and corrective actions are described in Attachment A.

The second violation involved an apparent failure to make a 4-hour notification as required by 10 CFR 50.72(b)(2)(i). CECo is concerned that the violation introduces new uncertanties regarding the extent to which engineering judgment may be relied on for making reportability determinations. In our response we have outlined the circumstances associated with this event, especially the nature, extent and timing of engineering judgment available. The response (Attachment A) includes a review of the chronology relevant to the development of that judgment, as well as the uncertainty over the applicability of that engineering judgment to reportability that the violation appears to pose. As detailed in Attachment B, we recognize and are acting on the need to improve our reporting mechanisms. However, in light of the chronology and the concerns identified, CECo respectfully requests the NRC to reconsider the violation.

Attachment B, as requested, addresses the other reporting concerns which were identified in the Inspection Report (IR).

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Finally, Attachment C provides clarification of issues associated with the non-cited violation involving a HPCI temporary alteration.

If there are any questions or comments regarding this response, please contact Rita Radtke, Compliance Engineer, at 708/515-7284.

Very truly yours,

T.J. Kovach
Nuclear Licensing Manager

Attachments

cc: A. Bert Davis, Regional Administrator - Region III
B.L. Siegel, Project Manager, NRR
W. Rogers, Senior Resident Inspector

ATTACHMENT A

RESPONSE TO NOTICE OF VIOLATION
NRC INSPECTION REPORT
50-237/91016; 249/91015

VIOLATION 1

10 CFR 50, Appendix B, Criteria 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...

Contrary to the above, for the period of initial operating license issuance thru June 28, 1991, the licensee failed to establish an adequate test program in that periodic calibration testing requirements were not established for safety-related fail-safe pressure actuation switches on specified reactor building ventilation air operated isolation dampers for Units 2 and 3 and for the Unit 2 emergency diesel generator cooling water local flow instrumentation.

This is a severity Level IV violation (Supplement I).

THE REASON FOR THE VIOLATION

The valves discussed in the inspection report are air operated butterfly valves. If air is lost to both sides of the piston of the air operator, the valve will fail in its current position. To assure that the valves will fail to a "failsafe" position, switches are provided which sense a loss of instrument air. The valves are provided with accumulators which have sufficient capacity to position the valves to the failsafe position, once the switch senses a loss of instrument air.

Prior to the Dresden Unit 2 Fall 1990 refueling outage (D2R12), procedure DIS 1600-11, "Pressure Suppression Air Operated Valve Pressure Switch Setpoint Check," accomplished the function of assuring that the subject valves would go to their failsafe position before instrument air pressure had decayed to a point where there was no longer sufficient energy to reposition the valves. Revision 1 to DIS 1600-11, dated January 20, 1977, accomplished this by monitoring the pressure at which the switch would trip thus causing the associated valve to failsafe. The normal switch setpoint was 75 psi. In certain cases where system interaction caused valve operational problems, a lower setpoint was selected.

Revision 2 of DIS 1600-11, dated May 21, 1980, changed the method of assuring that the valve would failsafe on loss of instrument air. The switch, operator, and accumulator were isolated from instrument air. The instrument air supply line was bled-off until the valve repositioned to its failsafe position, thus assuring proper valve operation. The pressure at which the switch would trip was no longer calibrated or recorded. While this method provided adequate assurance that the valve would currently failsafe on loss of instrument air, assurance was not provided that future drift in switch trip setpoint could be accommodated. The reason for changing the switch testing methodology could not be determined; however, the procedure revision review process failed to identify this deficiency.

In response to Generic letter 88-14, "Instrument Air Supply System Problems Affecting Safety-Related Equipment," a surveillance, DTS 1600-29, "Containment Vent and Purge and Reactor Building Ventilation Isolation Valve Fail Safe Mode and Air Operator Accumulator System Integrity Test" was developed. This surveillance accomplished all of the actions of DIS 1600-11, so DIS 1600-11 was cancelled. D2R12 was the first outage in which DIS 1600-11 was not performed. Like DIS 1600-11, DTS 1600-29 did not calibrate the loss of instrument air pressure sensing switches.

The modification checklist did not address calibration requirements, thus leading to inadequate review of modification M12-2-87-054 for diesel generator cooling water flow indication in that establishing a periodic calibration requirement for the flow instrument was not considered.

THE CORRECTIVE STEPS THAT WILL BE TAKEN TO AVOID FURTHER VIOLATIONS

1. A review has been performed to determine if other valves exist which have a loss of instrument air pressure sensing switch that is not being calibrated. In addition to the valves listed in the inspection report, the following valves have also been determined to have pressure switches which have not been calibrated: 2(3)-1601-20A(B) and 2/3-7510A(B).

2. A Special Procedure will be written based on Revision 1 of DIS 1600-11 to provide a preliminary means of calibration for the twenty-eight valves. This calibration will be completed by August 31, 1991.

3. The bases of the 75 psi (or lower) switch calibration setpoint and the calibration frequency of every refueling outage cannot be determined. The valve and switch vendors are being consulted to provide appropriate setpoint valves and calibration frequencies. Once obtained, a new procedure will be written to provide an adequate means to assure proper switch and valve performance in the event of degraded instrument air pressure. The new procedure will be in place and all switches will be calibrated as necessary based on the new procedure by December 31, 1991.

4. A periodic calibration program was established for the Unit 2, Unit 3 and Unit 2/3 Emergency Diesel Cooling Water flowmeter on May 31, 1991.

5. DAP 5-1, "Plant Modification Program," will be revised by December 31, 1991 to explicitly require the Modification Engineer to consider periodic calibration and functional testing requirements for newly installed equipment.

6. Dresden Station is currently developing an Instrument Data Sheet Program which is expected to be completed by June, 1992. All plant instruments will be identified and each safety related instrument will be reviewed to determine whether it is currently included in a periodic calibration program. Those safety-related instruments found not to be included in a periodic calibration program would subsequently be included.

DATE WHEN FULL COMPLIANCE WILL BE ACHIEVED

Full compliance will be achieved for the pressure switch calibration requirements by August 31, 1991 when the twenty-eight valves will be calibrated based upon values historically shown to provide adequate system performance.

Full compliance was achieved for the flowmeter calibration requirements on May 31, 1991 when a calibration program was established.

Violation 2

10 CFR 50.72(b)(2)(i) requires notification of the NRC within four hours of the occurrence of any event that, had it been found while the reactor was in operation, would have resulted in the nuclear power plant being in an unanalyzed condition that significantly compromises plant safety.

Contrary to the above, on January 16, 1991, the NRC was not notified within four hours that Unit 2, had it been operating, would have been in an unanalyzed condition that significantly compromised plant safety when a review by the licensee identified reactor head studs that did not meet the material toughness requirements of the Final Safety Analysis Report.

Background

During the eleventh refueling outage of Dresden 2 (D2R11), a routine ultrasonic examination of reactor head closure studs (studs) revealed that one of them had a crack indication in its lower threaded portion. No such indications had ever been seen before at either Dresden 2 or 3 or at Quad Cities 1 and 2. CECo responded conservatively by examining all 92 of the studs. An indication was found in one other stud; both studs with indications were replaced. The indications were reported as required in the Inservice Inspection (ISI) Report.

CECo then initiated a program of physical, chemical and metallurgical evaluations to determine whether the indications actually reflected cracks and, if so, to determine the source of cracking. Those evaluations were carried out by CECo's System Material Analysis Department (SMAD) with support from the Argonne National Laboratories (ANL). The results of the evaluations showed that the tested stud was cracked. The cracking appeared to be due to stress-corrosion cracking (SCC) initiated by a repeated exposure of the studs to water while under tension for the approximately two week period prior to vessel heat-up after each refueling outage.

Violation 2 (continued)

Included in the evaluation of the stud was a redetermination of its materials properties as compared with those initially reported on the Certified Material Test Report (CMTR). Two unexpected differences were observed: tensile strength was higher than expected by 10-20 ksi and material toughness was lower than expected by at least 10 foot-pounds at 10 degrees F. The causes of these unexpected differences are currently unknown but are under continuing investigation.

Dresden Station received the report of the SMAD evaluation on January 11, 1991, while Unit 2 was shut down for its twelfth refueling cycle. That same day, CECo personnel called cognizant NRC personnel both at Headquarters and at the Region to describe the test results. An extensive discussion ensued with many technical questions asked by the NRC. Two additional teleconferences were conducted on January 15, 1991 and on January 22, 1991. They are discussed below.

Almost immediately after the first teleconference, CECo requested that General Electric (GE) evaluate the safety significance of these observed differences in material properties. GE's preliminary conclusions were discussed during the second teleconference on January 15, 1991. Those GE preliminary conclusions and a copy of the SMAD report were telecopied to the NRC prior to the teleconference to facilitate the discussions.

During that teleconference, GE stated that, in its engineering judgment, the reactor pressure vessel could still meet the requirements which had been adopted from the American Society of Mechanical Engineer's (ASME) Code under certain conditions, all of which included (1) much greater degradation than the observed cracking of only two studs, and (2) the observed level of material toughness. Those conclusions were discussed at length.

No formal 4-hour notification was made by CECo. Based on the information then available, including GE's conclusions based on its engineering judgment, CECo determined that the non-conforming material condition did not indicate (1) either a serious degradation of the nuclear power plant, including its principal safety barriers, or (2) an unanalyzed condition that significantly compromised plant safety. Accordingly, with the reactor shut down, CECo concluded that there was no reason for making a 4-hour report.

CECo did, however, continue to review the condition for reportability. By January 16, 1991, CECo determined that the Final Safety Analysis Report (FSAR), in Appendix D, paragraph 10.10, specified material toughness for the studs. Because that toughness specification had not been met, the reactor had been outside the design basis for the plant. Therefore, a Licensee Event Report (LER) was initiated and filed on February 13, 1991. However, the determination that the plant was outside its design basis did not affect the prior preliminary conclusion that the change in material strength did not have a significant impact on plant safety.

On January 22, 1991, a third teleconference between CECo and the NRC presented the results of the stud inspections performed at Dresden Station to ensure safe unit operation. It was indicated that a LER had been initiated to document this event.

STATEMENT OF CECO CONCERN

CECo is concerned that this violation has serious implications for its ability to operate its plants consistent with the need to utilize engineering judgment in reportability determinations. Engineering judgment remains common to the determination of reportability and operability. If CECo engineering judgment is inappropriate when determining the proper reporting mechanism then the application of engineering judgment to demonstrate a reasonable expectation of operability is compromised.

We are concerned that the uncertainty the violation poses in the role of engineering judgment in reportability and operability decision, may result in the unnecessary cycling and shutdown of operable reactors.

The basis for our concerns relative to unnecessary reactor cycling and shutdowns are highlighted through a review of this violation. The NOV stated that the reactor, had it been operating,

> would have been in an unanalyzed condition
> that significantly compromised plant safety
> when a review by the licensee identified reactor
> head studs that did not meet the material toughness
> requirements of the Final Safety Analysis Report.

However, it was never our engineering judgment that plant safety was significantly compromised by the unanalyzed condition. The available expert engineering judgment provided CECo a reasonable expectation that there were no significant safety implications due to the observed changes in material properties. The violation does not recognize the extensive teleconferences which began immediately after the condition was identified and included discussions of GE's engineering judgment before the day that the violation concluded a 4-hour call would have been required.

Given the event chronology and the concerns identified, CECo respectfully requests the NRC to reconsider the violation.

ATTACHMENT B

DISCUSSION ON REPORTABILITY

Dresden acknowledges the NRC's concern that improvements are necessary in the Station's system of evaluating events for 10 CFR 50.72 reporting. To improve our performance in this area, we are focusing on two key issues:

> The first involves providing Operation's Shift personnel with timely information on the existence of conditions in the plant which have potential reportability concerns. As discussed in the inspection report, there was inadequate and untimely communication between the Instrument Maintenance Department and Operation's Shift personnel. To address this issue a memorandum was issued and a meeting was held with Instrument Maintenance personnel regarding timely and complete information to the Operation's Shift personnel. A meeting was held with Operation's Shift personnel regarding the timely evaluation of information for operability and reportability considerations. Additional information is being provided to Operation's Shift personnel to allow them to make timely evaluations of reportability and operability. In the case of the Emergency Core Cooling System level switches, an operator aid is being developed which should provide for more timely operability assessments.
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> Dresden was previously cited (237/90027, 249/90026) for not reporting the closure of selected valves associated with the Group II primary containment isolation system. It had been Dresden's prior understanding that such actuations were not considered to be Engineered Safety Feature (ESF) actuations. Additionally, System isolations as opposed to Containment isolations were not historically reported. The Station now understands the NRC's position. For example, on August 5, 1991 a 10 CFR 50.72 report was made concerning the momentary cycling of the scram valve on a single control rod drive. During an Instrument Maintenance surveillance, a single scram valve momentarily 'chattered' causing the control rod to insert a few notches. A 10 CFR 50.72 report was made because an end component of an ESF system cycled even though a reactor scram signal was not received.

The Station recognizes that improvement is necessary. On August 6, 1991, in an event which involved a recirculation pump trip, the Low Pressure Coolant Injection (LPCI) was made inoperable. Although a seven day Limiting Condition of Operation (LCO) was entered, the required 10 CFR 50.72(b)(2)(iii)(D) report was not immediately recognized. A meeting was held on August 7, 1991 with the Shift Engineers, the Shift Control Room Engineers, the Operating Engineers, the Assistant Superintendent of Operations, an Assistant Technical Staff Supervisor, and the Regulatory Assurance Supervisor. Recent events and current policies involving reportability were discussed.

The Corporate Nuclear Licensing Department is initiating the development of a computer program which would contain the text of the questions and answers from NUREG-1022 Supplement 1. The program will provide word search capability of the NUREG-1022 reporting guidance. The program will be accessible by each nuclear station and the Corporate Office. It is expected that the program will be available for use by the end of 1991.

ATTACHMENT C

CLARIFICATION OF ISSUES
ASSOCIATED WITH THE HPCI TEMPORARY ALTERATION VIOLATION

The inspection report discussed a non-cited violation involving a Temporary Alteration in which some test equipment was connected to the High Pressure Coolant Injection (HPCI) system. The violation involved an inadequate safety evaluation required by 10 CFR 50.59 in that issues of electrical separation between safety related and non-safety related equipment were not considered. The Station agrees that the safety evaluation was inadequate. In response, a revision to the safety evaluation procedure was made to require consideration of electrical separation.

One issue discussed in the violation requires clarification:

> The inspection report implies that the Temporary Alteration should have been installed in accordance with Regulatory Guide 1.75, "Physical Independence of Electrical Systems," and IEEE 384. A March 6, 1985 CECo letter is cited as the source of this requirement. The applicable paragraph from the referenced letter reads as follows:
>
> > As modifications are incorporated into the plant design, whenever practical, greater separation between IE sources and non-IE loads will be provided in accordance with the current philosophy as stated in Reg. Guide 1.75. The DC system modification noted above is one such example. When new systems are installed they are in accordance with current standards (EG. IEEE 384) where practical.
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> The Station does not believe that the extrapolation from "installing new systems in accordance with current standards where practical" to a requirement to install all Temporary Alterations in accordance with IEEE 384 is appropriate.