December 13, 1999

Mr. Oliver D. Kingsley
President, Nuclear Generation Group
Commonwealth Edison Company
ATTN: Regulatory Services
Executive Towers West III
1700 Opus Place, Suite 500
Downers Grove, IL 60515

SUBJECT: NRC INSPECTION REPORT 50-254/99021(DRS); 50-265/99021(DRS)

Dear Mr. Kingsley:

On November 19, 1999, the NRC completed the pilot baseline biennial safety system design and performance capability inspection at your Quad Cities Nuclear Station. The results of this inspection was discussed with Mr. J. Dimmette and other members of your staff at the end of the inspection. The enclosed report presents the risk-significant results of this inspection.

The inspection was an examination of activities conducted under your license as they related to ensuring that the automatic depressurization system and the 125 volt direct-current system were capable of performing their required post-accident functions, and to verify compliance with the Commission=s rules and regulations and with the conditions of your license. Within these areas, the inspection consisted of a selected examination of procedures and representative records, observations of activities, and interviews with personnel.

During the inspection one non-cited violation was identified regarding the failure to correctly translate the design basis of the four Unit 1 electromatic relief valves into design specifications. Specifically, the inspection team found that the licensee had not ensured that the valves were qualified to operate under the minimum degraded voltage conditions that they would experience when being powered by the battery source. This issue was reviewed under the significance determination process and determined to be of low risk significance.

If you contest the violation or the severity level of any non-cited violation, you should provide a response within 30 days of the date of this inspection report with the basis for your denial, to the Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington DC 20555-0001, with copies to the Regional Administrator, Region III, the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, DC 20555-0001, and the NRC Resident Inspector at the Quad Cities facility.

In accordance with 10 CFR 2.790 of the NRC=s ARules of Practice,@ a copy of this letter and its enclosure will be placed in the NRC Public Document Room (PDR).

We will gladly discuss any questions you have concerning this inspection.

Sincerely,

/s/ J. M. Jacobson

John M. Jacobson, Chief
Mechanical Engineering Branch

Docket Nos. 50-254; 50-265
License Nos. DPR-29; DPR-30

Enclosure: Inspection Report 50-254/99021(DRS); 50-265/99021(DRS)

cc w/encl: D. Helwig, Senior Vice President, Nuclear Services
C. Crane, Senior Vice President, Nuclear Operations
H. Stanley, Vice President, Nuclear Operations
R. Krich, Vice President, Regulatory Services
DCD - Licensing
J. Dimmette, Jr., Site Vice President
G. Barnes, Quad Cities Station Manager
C. Peterson, Regulatory Affairs Manager
M. Aguilar, Assistant Attorney General
State Liaison Officer, State of Illinois
State Liaison Officer, State of Iowa
Chairman, Illinois Commerce Commission
W. Leech, Manager of Nuclear
MidAmerican Energy Company

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Chairman, Illinois Commerce Commission
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U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket Nos:	50-254; 50-265
License Nos:	DPR-29; DPR-30
Report No:	50-254/99021(DRS); 50-265/99021(DRS)
Licensee:	Commonwealth Edison Company
Facility:	Quad Cities Nuclear Power Station Units 1 and 2
Location:	22710 206th Avenue North Cordova, IL 61242
Dates:	November 1 - November 19, 1999
Inspectors:	Patricia Lougheed, Team Lead Ron Langstaff, Reactor Inspector Roger Mendez, Reactor Inspector Tom Tella, Reactor Inspector Omar Mazzoni, Contractor
Approved by:	John M. Jacobson, Chief, Mechanical Engineering Branch Division of Reactor Safety

SUMMARY OF FINDINGS

Quad Cities Nuclear Power Station, Units 1 & 2
NRC Inspection Report 50-254/99021(DRS); 50-265/99021(DRS)

This report covers the pilot baseline inspection for the biennial safety system design and performance criteria. This inspection reviewed the automatic depressurization system and the 125 volt direct-current system.

Inspection findings were assessed according to potential risk significance, and were assigned colors of GREEN, WHITE, YELLOW, or RED. GREEN findings are indicative of issues that, while not necessarily desirable, represent little risk to safety. WHITE findings would indicate issues with some increased risk to safety, and which may require additional NRC inspections. YELLOW findings would be indicative of more serious issues with higher potential risk to safe performance and would require the NRC to take additional actions. RED findings represent an unacceptable loss of margin to safety and would result in the NRC taking significant actions that could include ordering the plant shut down. No individual finding by itself would be indicative of either acceptable or unacceptable performance. The findings, considered in total with other inspection findings and performance indicators, will be used to determine overall plant performance.

Cornerstone: Mitigating Systems

Green: Four of the five Unit 1 automatic depressurization system relief valves were not initially qualified for the degraded voltage conditions that would be seen under accident conditions. The licensee had not originally accounted for the voltage drop that would occur between the station batteries and the valve solenoids when specifying the minimum voltage for which the valves needed to be qualified. The licensee subsequently identified a test report, done for another nuclear station, which qualified the valves to a lower voltage. The inspectors, in conjunction with the Office of Nuclear Reactor Regulations, reviewed the test and accepted it under condition that a ten volt penalty be applied to account for some test deficiencies. The licensee also performed calculations to show that the actual available voltage to the valves, under degraded conditions, was above the accepted minimum voltage. A non-cited violation was identified.

Report Details

1. REACTOR SAFETY (Cornerstone: Mitigating Systems)

1R21 Safety System Design and Performance Capability

a. Inspection Scope

Two systems were selected for this inspection: the automatic depressurization system (ADS) and the 125 volt dc system. These systems were selected based upon:

- **$** performing a mitigating system function;
- **$** having high safety significant maintenance rule functions;
- **$** having high risk achievement worths in the probabilistic risk assessment;
- **$** not having received recent NRC review (for ADS); and
- **$** supporting multiple systems (125 volt dc).

The systems were considered complementary in that 125 volt dc power is necessary for the ADS to function.

For the ADS, the following inspection attributes were reviewed in detail:
Energy Source - electricity
Control System - initiation and control
Environmental Requirements - temperature
Equipment Protection - seismic, vacuum relief
Operator Actions - control
Installed Configuration - walkdowns of accessible areas and review of piping drawings
Operation
Design Corrective Actions - selected licensee problem identification forms
Tested Parameters

For the 125 Vdc system, the following inspection attributes were reviewed in detail:
Energy source - batteries and battery chargers
Control System - control power to breakers and end devices
Environmental Requirements - temperature and humidity
Equipment Protection - seismic, tornado and temperature control of battery rooms
Installed configuration - walkdowns of the battery rooms, battery chargers, dc breakers, fuses, dc buses, ground detection and the inputs to the inverters
Maintenance - predictive and preventative maintenance including how the licensee controlled age degradation and service related wear
Design - dc load requirements (adequate voltage and current available at the end device) through review of dc load and battery calculations
Design Corrective Actions - selected licensee problem identification forms
Tested parameters - to ensure the surveillance procedures adequately test the licensing and design bases and whether the tested parameters are supported by calculations
Component Configuration
Procurement - vendor requirements
Industry Operating Experience

b. Observations and Findings

The inspectors identified that four of the five Unit 1 ADS valves would not operate during accident conditions because the minimum voltage for which the valves were qualified did not account for voltage drop. The valves, which are located within the containment drywell, were qualified for a minimum pickup voltage of 105 volts dc during accident conditions; however, the station batteries, which supplied power to the valves, could have a minimum voltage of 105 volts dc. Accounting for voltage drop through the cabling and contacts, the voltage at the valve solenoids would have been significantly below their minimum qualified voltage.

The design engineering staff reviewed the concern and concurred that the voltage drop had not been properly accounted for when the minimum valve qualification voltage was specified. On November 16, 1999, the licensee issued problem identification form Q1999-3992. To resolve the issue, the licensee obtained a copy of a test report done for the Oyster Creek Nuclear Station, which concluded that an identical valve would operate down to a minimum voltage of 80.1 volts dc. Based on this test report, the licensee revised the valve environmental qualification binder 41Q to include the test results and indicate that the Quad Cities electromatic relief valves were qualified to a minimum voltage of 80.1 volts dc.

The inspectors reviewed the test report which provided the basis for the lower voltages and found three major deficiencies which seriously compromised the acceptability of the 80.1 volt dc minimum terminal voltage. The first deficiency was that the test as described by the report failed to meet the power supply voltage specification of "0.1 volts dc at the solenoid, as specified in the test report. Instead, the test report showed an alternating voltage with various degrees of divergence, dependent upon the test conditions. The inspectors noted that the 80 volts dc corresponded to the average voltage of the wave, rather than the peak voltage seen. The inspectors also noted that the peak voltages occurred at the same time the peak currents were experienced, indicating that the voltage regulation of the rectifier was insufficient for the application. Another issue with the acceptability of the test report was that the valve was fully refurbished prior to being tested, with no as-found testing. The third issue was that only one valve was tested, which did not provide a statistically meaningful result. Particularly, the inspectors were concerned with the use of the minimum voltage at which the tested valve would function, without adding any margin for uncertainties or statistical variations among valves. For these reasons, the inspectors, with concurrence from the Office of Nuclear Reactor Regulation, concluded that it was more reasonable to assume a minimum acceptable voltage of 90 volts dc. The licensee performed a preliminary calculation and determined that the minimum voltage that would be available at the valve solenoid terminals would be 92.4 volts dc. Therefore, the inspectors concluded that the valves were operable.

10 CFR Part 50 Appendix B, Criterion III "Design Control" requires, in part, that the design basis be correctly translated into specifications, drawings, procedures or instructions. This criteria was not met in that environmental qualification binder 41Q, "Environmental Qualification of Dresser Electromatic Relief Valve Solenoid Actuator Model 1525VX,"

Revision 5, specified a minimum acceptable solenoid voltage that was equal to the battery discharge voltage and did not account for voltage line losses, a design basis consideration. This is a violation of Criterion III. The inspectors reviewed the criteria in the significance determination process and determined that this item was of low risk significance because the issue only arose when the ADS valves were subjected to high containment temperatures which limited the significance determination process to the small and medium break loss of coolant accident worksheets. Since only four of the five ADS valves were affected and one ADS valve would still have been operational, all success criteria were met.

The inspectors noted that under design basis conditions, the loss of four ADS valves would have been of significance because a loss of offsite power would have been assumed concurrent with a medium break less of coolant accident. The design basis event would also have required operation of three ADS valves. Based on the initial information available, the design basis function would not have been met. Because the licensee was able to provide sufficient information by the end of the inspection to show that the valves would operate under the degraded voltage that would be experienced, this issue will be considered a non-cited violation, in accordance with the "Interim Enforcement Policy for Use During the NRC Power Reactor Oversight Process Pilot Plant Study."

4. OTHER ACTIVITIES

4OA5 Management Meetings

.1 Exit Meeting Summary

The inspector presented the inspection results to members of licensee management in an exit meeting on November 19, 1999. The licensee acknowledged the information and findings presented. The inspectors identified the proprietary information reviewed during the inspection and questioned the licensee as to whether proprietary information had been retained. The inspectors also discussed the potential for proprietary information to be included in the inspection report. The licensee confirmed that no proprietary information was retained at the completion of the inspection. The licensee concurred that the proposed inspection report content would not compromise any proprietary information.

PARTIAL LIST OF PERSONS CONTACTED

Com Ed

G. Barnes, Station Manager
J. Dimmette, Site Vice President
M. McDonald, Operations Manager
C. Peterson, Regulatory Assurance
D. Wozniak, Engineering Manager

Mid-America Energy

D. Tubbs, Site Representative

NRC

J. Jacobson, Chief, Mechanical Engineering Branch, Division of Reactor Safety
C. Miller, Senior Resident Inspector
K. Walton, Resident Inspector
L. Collins, Resident Inspector
S. Malur, Senor Reactor Engineer, Inspection Program Branch, NRR

IDNS

R. Ganser, Resident Inspector

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

50-254/99021-01	NCV	Failure to Ensure ADS Valves Qualified for Minimum Possible Voltage

Closed

50-254/99021-01	NCV	Failure to Ensure ADS Valves Qualified for Minimum Possible Voltage

LIST OF ACRONYMS USED

ADS	Automatic Depressurization System
NCV	Non-Cited Violation
NRC	Nuclear Regulatory Commission
NRR	Office of Nuclear Reactor Regulation
PDR	Public Document Room

INSPECTION PROCEDURE USED

IP 71111-21 (draft)	Safety System Design and Performance Capability

LIST OF DOCUMENTS REVIEWED

The following is a list of licensee documents reviewed during the inspection, including documents prepared by others for the licensee. Inclusion on this list does not imply that NRC inspectors reviewed the documents in their entirety, but, rather that selected sections or portions of the documents were evaluated as part of the overall inspection effort.

Calculations

4834-40-19-29 Voltage Drop in Appendix R Repair Cables Feeding Electromatic Relief Valves, Revision 2
5570-31-19-01 125 Volt dc Fault Current, Revision 2
5570-31-19-02 125 Volt dc System Breaker and Fuse Coordination, Revision 2
6838-00-19-01 Verification of 125 Volt dc Battery Capacity, Revision 0
7175-30-19-02 Determination of the Minimum Voltage at the Battery Equipment Room, Revision 0
7294-00-19-01 Battery Charger Recharge Time of Batteries, Revision 2
7318-32-19-1 125 Volt dc Load Profiles for The Electric Loads Management System, Revisions 33 & 36
7330-30-19-1 125 Volt dc and 250 Volt dc Batteries - Actual Overall Capacity, Revision 0
7330-TB-05 Battery Replacement 125 Volt dc and 250 Volt dc, Revision 2
7331-01-19-02 125 Volt dc Battery Bus Voltage Drop Testing (Temporary and Permanent Batteries) for Modification M4-2-88-43, Revision 0
8256-14-19-01 dc Ground Task Force Report, Revision 0
8.9.1-3 Tornado Missile Probability Analysis for Various Tornado Intensities, Revision 0
8913-13-19-01 dc Distribution and Battery Rooms Electrically Generated Heat Loads, Revision 0
9149-20-19-1 125 Volt dc Bus Voltage Calculations for Quad Cities Station, Revision 5
9399-02-19-00 Calculation for Determining Acceptability of Using Existing circuitry for the Replacement of Relief Valves, Revision 0
COM-42-014 Safety Relief Valve Blowdown Control Logic, Revision 1
COM-42-130 Mark I Program Safety Relief Valve Inhibit Logic Modifications for Quad Cities Units 1 and 2 and Dresden Units 2 and 3, Revision 0
CQD-015004 Environmental Qualification (EQ-430) of Target Rock/AVCO Solenoid Valves, Revision 7
CQD-020657 Seismic Qualification of 125 Volt dc Battery Racks with GNB NCX-1500 Cells, Revision 1
CQD-515011 Design and Seismic Qualification of 125 Volt dc Battery Rack, Revision 0
CQD-515064 Seismic Qualification of 125 Volt dc Battery Racks with GNB NCX-1500 Cells, Revision 1
CQD-515066 Effect of Tornado on Battery Gould NCX-1500, Revision 0
ENC-QE-76.5 Overcurrent Protective Device Coordination, Revision 0
NED-E-EIC-0058 125 Volt Battery Chargers 1, 1A, 2, and 2A Settings, Revision 0
NED-I-EIC-0043 Reactor High Pressure Automatic Blowdown Setpoint Error Analysis, Revision 1
NEdc-24379 Evaluation of Mark I Safety/ Relief Valve Load Cases C3.2 and C3.3 (Proprietary), December 1981

NES-EIC-20.04	Analysis of Instrument Channel Setpoint Error and Instrument Loop Accuracy, Revision 1
QC-IET-E-001 Battery	Room (Units 1 & 2) Minimum Airflow Requirements, Revision 0
QC-74Q-E-001	Determination of Minimum dc Control Voltages at Station Black Out Tie-In Breakers at the 4 kV Buses, Revision 0
Qdc-0030-S-0359	Seismic Qualification of the Battery Rooms in the Turbine Building, Revision 0
Qdc-0287-M-0701	Pressure Rating for Target Rock Automatic Depressurization System Valve Accumulators, Revision 0
Qdc-8300-E-0482	Evaluation of 125 Volt dc System Coordination for Appendix R, Revision 2
Qdc-8300-E-0577	125 Volt dc Control Circuit Voltage drop evaluation for new 4kV breakers Close and Trip coils at 13-1 and 14-1 4kV Buses, Revision 1
Qdc-8300-S-0441	Required Seismic Clearance for 125 Volt dc Bus 1A/1A-1, Revision 0
Qdc-8300-S-0673	Review of Aged Battery Seismic Qualification Report, Revision 0
TID-E/I&C-28	Sizing Control Cables, Revision 0
VT-08	Determination of the Ventilation Requirements for Battery Rooms Unit 1 and Unit 2, Revision 3

Correspondence - to NRC

June 30, 1978	Evaluation of the Consequences of a Multiple Subsequent Relief Valve Discharge on the Mark I Containments (ComEd)
January 24, 1980	Temporary Technical Specification Changes Concerning Automatic Depressurization System Operability (ComEd)
January 25, 1980	Additional Response to IE Bulletin 80-01 (ComEd)
July 2, 1980	Revised Schedule for Mark I Containment and Feedwater Sparger Replacement Projects (ComEd)
December 16, 1981	BWR Mark I Containment Order Date Extensions (ComEd)
April 29, 1982	Proposed Amendment to Technical Specifications Concerning Surveillance of Automatic Pressure Relief Valves when the High Pressure Coolant Injection Subsystem Is Inoperable (ComEd)
October 18, 1982	Proposed Technical Specification Change Concerning Safety/Related Valve Setpoints (ComEd)
December 22, 1982	Clarification of BWR Mark I Containment Minor Modifications
May 10, 1983	Response to NUREG-0737 Item II.K.3.28, Verify Qualification of Accumulators on Automatic Depressurization System Valves (ComEd)
June 27, 1983	BWR Mark I Containment Program Plant Unique Analysis Reports (ComEd)
July 15, 1983	Proposed Technical Specification Change Concerning Safety/Relief Valve Setpoints (ComEd)
July 18, 1983	Status of NUREG 0737 Item II.K.3.18, Modification of Automatic Depressurization System Logic (ComEd)
July 28, 1982	Quad Cities Station Nuclear Power Station, Units 1 and 2, Engineered Safety Features Reset Controls (Memorandum from T. M. Novak, NRC, to E. L. Jordan, NRC)

February 28, 1986	NUREG-0737 Item II.K.3.28, Qualification of Automatic Depressurization System Accumulators (ComEd)
April 9, 1986	NUREG-00737 Item II.K.3.28, Qualification of Automatic Depressurization System Accumulators (ComEd)
March 21, 1989	Application for Amendment to Facility Operating Licenses DPR-29 and DPR-30, Appendix A, Technical Specifications (ComEd)
January 10, 1996	Response to NRC Generic Letter 96-01, Testing of Safety-Related Logic Circuits (ComEd)
May 20, 1999	Revision to Schedule for Completing Setpoint/Uncertainty Calculations for Specific Functions (ComEd)

Correspondence - to Licensee

February 2, 1979	Staff Evaluation of Interim Multiple-Consecutive Safety-Relief Valve Actuations (NRC to all BWR licensees)
February 1, 1980	Issuance of Amendments 46 and 40 (NRC)
January 7, 1982	Resolution of TMI Action Item II.E.4.2.5, Containment Pressure Setpoint - Quad Cities Units 1 and 2 (NRC)
September 22, 1982	Response to Suggested Changes in SRV Inhibit Logic (NuTech),
December 15, 1982	Issuance of Amendment 83 (NRC)
December 30, 1982	Safety Evaluation Report for Appendix R to 10 CFR Part 50, Items III.G and III.L (NRC)
March 16, 1983	NUREG-0737 Item II.K.3.20, Verify Qualification of Accumulators on Automatic Depressurization System Valves (NRC)
May 11, 1983	Request for Additional Information - NUREG-0737, Item II.K.3.28, Qualification of Automatic Depressurization System Accumulators (NRC),
May 18, 1983	Resolution of NUREG-0737 Item II.K.3.45, Depressurization With Other Than Automatic Depressurization System (NRC)
June 3, 1983	NUREG-0737, Item II.K.3.18, Automatic Depressurization System Logic Modifications (NRC)
January 17, 1985	Request for Additional Information - MPA F-55 (TMI II.K.3.28), Qualification of Automatic Depressurization System Accumulators (NRC),
January 24, 1985	Response to NUREG-0737, Item II.K.3.18, Automatic Depressurization System Modification (NRC)
June 16, 1986	Resolution of NUREG-0737 Item II.K.3.28 Verify Qualification of Accumulators on Automatic Depressurization System Valves (NRC)
May 31, 1996	Disposition of ATWS Event at Quad Cities (Siemens Proprietary)
July 14, 1997	Completion of Licensing Action for GL-96-01, Testing of Safety-Related Logic Circuits (NRC)

Design Change Packages

8200057	Safety Relief Valves Logic Modification
9700286	Temporary Installation of Air Ducts on Temporary Air Conditioning Unit to the Unit 1 Battery Room, (Temporary Alteration), July 28, 1997
9800014	Install Portable Heaters in Unit 2 Battery Room, January 14, 1998

9800192	Replace 125 Volt dc Main Feed Breaker at RB-2 for the 4 kV Switchgear Bus 23-1, May 30, 1998

Drawings

111502	External Wiring Diagram for Consoli Type Electromatic Relief Valve with 1530VX or 1530VC Controller and 1537 Control Station, Dresser, 5/4/72
4E-1050	Cable Routing Turbine & Reactor Building Ground Floor, Revision BB
4E-1051	Cable Routing Turbine & Reactor Building Mezzanine Floor, Revision AP
4E-1059	Cable Routing Torus Area, Control Room, and Filter Building, Revision S
4E-1067F	Connection Layout 125 Volt dc and 250 Volt dc Battery Cells, Revision G
4E-1318	125 Volt dc Turbine Building Reserve Bus Distribution Panel and 125 Volt dc Reactor Building Distribution Panel 1, Revision F
4E-1318A	Turbine Building 125 Volt dc Main Bus Distribution Panel, Revision R
4E-1318B	Overall Key Diagram- 125 Volt dc Distribution Centers, Revision E
4E-1389B	125 Volt dc Battery Charger #1 Quad-Cities Station Unit 1, Revision L
4E-1389F	Wiring and Schematic Diagram 125 Volt dc Trickle Charger for Alternate Battery 1, Revision A
4E-1428	Schematic Diagram Annunciator & Computer Core Spray System I & II, Revision AE
4E-1429	Relaying Metering & Schematic Diagram Core Spray Pumps 1A & 1B, Revision U
4E-1430-1 & 2	Schematic Diagram Core Spray Systems I & II, Revisions AY & AT
4E-1461-1 & 2	Schematic Diagram Auto Blowdown Part 1, Revisions AP & AT
4E-1461A	Schematic Control Diagram Auto Blowdown Part 1, Revision C
4E-1462-1,2 & 3	Schematic Diagram Auto Blowdown Part II, Revisions AK, AJ & AF
4E-1462A	Schematic Control Diagram Auto Blowdown Part 2, Revision C
4E-1685A	Wiring Diagram Turbine Building 125 Volt dc Reserve Bus 1B-1 Distribution Panel, Revision AP
4E-1685D	Turbine Diagram Turbine Building 125 Volt dc Main Buses 1A-1 and 1A-2 Distribution Panel, Revision M
4E-1798F	Wiring Diagram 125V dc Battery #1A Quad-Cities Station Unit 1, Revision J
4E-2318	Key Diagram 125 Volt dc Distribution Panel, Revision M
4E-2428-1	Schematic Control Diagram Annunciator & Computer Core Spray System I & II, Revision D
4E-2429-1	Schematic Diagram and Relaying & Metering Core Spray Pumps 2A & 2B, Revision N
4E-2430-1, 2 & 3	Schematic Diagram Core Spray System I & II, Revisions AJ, AG & Z
4E-2461-1 & 2	Schematic Diagram Auto Blowdown Part I, Revision AH & AG
4E-2461A	Schematic Control Diagram Auto Blowdown Part 1, Revision D
4E-2462-1 & 2	Schematic Diagram Auto Blowdown Part II, Revisions AA & AB
4E-2462A	Schematic Control Diagram Auto Blowdown Part 2, Revision D
4E-2798E	Wiring Diagram 125 Volt dc Battery Charger 2A, Revision M
93V-001	Power Operated Relief Valve (Class 900),Target Rock Corporation, 3 sheets, 3-9-94

M-13-1 Diagram of Main Steam Piping, Unit 1, Revision AM
M-24-12 & 13 Diagram of Instrument Air Piping Reactor Building, Revisions C and E
M-35- 1 Diagram of Nuclear Boiler & Reactor Recirculating Piping, Revision AR
M-36 Diagram of Core Spray Piping, Uni t 1, Revision AV
M-39-1,2, & 3 Diagram of Residual Heat Removal Piping, Unit 1, Revisions BD, AY, & C
M-60-1 Diagram of Main Steam Piping, Unit 2, Revision AK
M-76-1 Diagram of Pressure Suppression Piping, Revision BA
M-78 Diagram of Core Spray Piping, Unit 2, Revision BA
M-81-1, 2 & 3 Diagram of Residual Heat Removal Piping, Unit 2, Revisions BJ, BG, & C
M-305 Target Rock Valve Instrument Air Line Modification, Revision B
Unnumbered Electromatic Relief Valve, Cross Sectional Views, 9 sheets

Environmental Qualification Binders

Binder EQ-17Q Target Rock Corporation Power Operated Relief Valves, Model #93V-001, Revision 00
Binder EQ-41Q Environmental Qualification of Dresser Electromatic Relief Valve Solenoid Actuator Model 1525VX, Revisions 5 & 5A

Information Notices

80-01 Operability of Automatic Depressurization System Valve Pneumatic Supply, May 3, 1983
80-25 Operating Problems with Target Rock Safety/ Relief Valves at Boiling Water Reactors, December 19, 1980
80-40 Excessive Nitrogen Pressure, November 7, 1980
81-38 Potentially Significant Equipment Failures from Contamination of Air-Operated Systems, December 17, 1981
82-41 Failure of Safety/ Relief Valves to Open at a Boiling Water Reactor, October 22, 1982
83-39 Failure of Safety/ Relief Valves to Open at Boiling Water Reactor - Interim Report, June 17, 1983
83-82 Failure of Safety/ Relief Valves to Open at Boiling Water Reactor - Final Report, December 20, 1983
85-35 Failure of Air Check Valves to Seat, Supplement 1, May 17, 1988
86-12 Target Rock Two Stage Safety/ Relief Valve Setpoint Drift, February 25, 1986
86-50 Inadequate Testing to Detect Failures of Safety-Related Pneumatic Components or Systems, June 18, 1986
86-51 Excessive Pneumatic Leakage in the Automatic Depressurization System, June 18, 1986
88-30 Target Rock Two Stage Safety/ Relief Valve Setpoint Drift Update, May 25, 1988
89-26 Instrument Air Supply to Safety-related Equipment, March 7, 1989
89-90 Pressurizer Safety Valve Lift Setpoint Shift, December 28, 1989; Supplement 1, April 3, 1991; Supplement 2, September 5, 1991

90-11	Maintenance Deficiency Associated with Solenoid-Operated Valves, February 28, 1990
91-74	Changes in Pressurizer Safety Valve Setpoints Before Installation, November 25, 1991
93-02	Malfunction of a Pressurizer Code Safety Valve, January 4, 1993
94-56	Inadequacies of Safety Valve Setpoint Determination, August 11, 1994
95-47	Unexpected Opening of a Safety/ Relief Valve and Complications Involving Suppression Pool Cooling Strainer Blockage, October 4, 1995; Supplement 1, November 30, 1995
96-02	Inoperability of Power-operated Relief Valves Masked by Downstream Indications During Testing, January 5, 1996
98-23	Crosby Valve Setpoint Drift Problems Caused by Corrosion of the Guide Ring, June 23, 1998

Generic Letters

88-14	Instrument Air Supply System Problems Affecting Safety-Related Equipment, August 8,1988

Modifications

M4-1-82-018	Modify Battery Room Heating, Ventilation and Air Conditioning System to Provide More Adequate Heating and Cooling in the Room
M4-1/2-84-011	125 Volt dc Cross-Tie between 4 kV Switchgear 13-1 and 23-1
M4-2-93-012	Replace Dresser Electromatic Relief Valves with Target Rock Power Operated Relief Valves

Problem Identification Forms Generated From the Inspection

Q199903742	Contacts 5-6 for Switch 287-304 Shown Incorrectly on 4E02461 Sheet 1
Q199903738	Unknown Location of Design Change Package 9300355
Q199903943	Automatic Depressurization System Solenoid Fuse Protection
Q199903962	Relief Valve Blow down Range Design Basis Not Referenced in Set Point Calculation
Q1999-03992	No Voltage Drop Calculation Exists for the Unit One Electromatic Relief Valves (EPNs 1-0203-3B through E)
Q199903999	Superseded Calculations Shown as Active
Q199904000	Surveillance Procedure QCIS 0200-20 Has Incorrect Head Correction Identified

Problem Identification Forms Reviewed During the Inspection

Q199501844	Unit 2 Auto Blowdown Fuse 23 (287-711a) Was Found to Be a 3 Amp Fuse Instead of a 10 Amp Fuse per Design
Q199501929	Acceptance Criteria for the Inhibit Logic for Valves 203-3B and 3C Found to be 9 to 10 Seconds

Q199502145	Unit 2 Power Operated Relief Valve Auto-inhibit Relays Should Be Set to Approximately 14 Seconds in Order to Electrically Prohibit Valve Operation AFTER Closure for at Least 10 Seconds
Q199701744	Intercell Resistance Between Cell 4 and 5 is 6.2 Micro-ohms Above Criteria
Q199703203	QARP 1100-01 Closes 125 Volt dc 13-1 to 23-1 Cross-Tie Contrary to Safe Shutdown Document Analysis Requirement
Q199704735	Level 3 Ground on 125 Volt dc System
Q199800166	Unit Two 125 Volt Battery Inoperable Due to Average Cell Temperatures less than 65 Degrees, January 14, 1998
Q199800384	Trend Problem Identification Form on Inaccurate Battery Calculations
Q199800656	Station Procedures and Updated Final Safety Analysis Report not Revised after Automatic Depressurization System Inhibit Time Delay Setpoint Change
Q199800745	Station Battery Profiles for Safety-related 125 and 250 Volt dc Batteries Not Up to Date
Q199800816	Batteries Over 10 Years Seismic Life
Q199801744	dc Distribution Breaker Failed to Open
Q199802604	Possible Incorrect Breaker Installed in 125 Volt dc Bus 1B-1 Cubicle C02
Q199802915	125 Volt Battery Charger 2A Failed its 4 Hour Load Test
Q199803067	Unit One 125 Volt dc Level 3 Ground
Q199805039	Inadequate Testing of Automatic Depressurization System Logic per QCOS 0203-06, Revision 0
Q199805050	Breaker Failed Overcurrent Trip Check
Q199805677	Unit 2 125 Volt dc Charger Failed Post Maintenance Testing
Q199900543	Unit 1 Level 3 125 Volt dc Ground
Q199900576	Unit 2 Safety-related Battery Exceeded Intercell Resistance Limits
Q199900671	Thermal Runaway of Unit One IPEEE Battery
Q199900737	IPEEE Battery Cell Failure
Q199900807	Errors in Loss of Offsite Power/ Loss of Coolant Accident Calculation 9149-20-19-1
Q199900901	Plant Operations Review Committee Approved Temporary Modifications with Inadequate Testing Criteria
Q199901227	Unit One 125 Volt dc Battery Ground
Q199901290	Unit One Level 3 125 Volt dc Ground
Q199901542	Problems Identified with Intercell Connections on the Station Batteries
Q199901674	Unit 2 125 Volt dc Battery Maintenance Rule Functional Failure due to Low Electrolyte Temperature
Q199901728	Unable to Perform Failure Analysis on Defective Breaker
Q199902008	Received Hard 125 Volt dc Ground on Unit 1 Safety-related Battery Bus
Q199903534	Nuclear Operator Identifies Discrepancies with Dc Grounds Procedure
Q199903606	Administrative Issues with Calculations 7318-32-19-1, Revision 36 and 9149-20-19-1, Revision 5
Q199903633	Operability Evaluation Q1998-02604 Corrective Action Adequacy

Procedures

1100	Wyle Laboratories Test Procedure for Target Rock Three Stage Pilot-Operated Relief Valves Model 7467F (OM-1987), March 7, 1997
NEP-12-02	Preparation, Review and Approval of Calculations, Revision 9
NEP-0401	Plant Modifications, Revision 6
QCAN 901(2)-3 B-13	Automatic Blowdown Timer Start, Revision 3
QCAN 901(2)-3 C-15	Auto Blowdown Interlock Core Spray/Residual Heat Removal (Pump Running Permissive), Revision 2
QCAN 901(2)-3 D-15	Automatic Blowdown System in Unapproved Test Mode of Operation, Revision 0
QCAN 901(2)-3 E-13	Relief Valve 1(2)-203-3C/3D and/or 3D is Open, Revision 3
QCAN 901(2)-3 E-14	Acoustic Monitor Indicates Safety and/or Relief Valve Open, Revision 4
QCAN 901(2)-3 F-13	Automatic Blowdown System Drywell High Pressure (Depressurization System High Drywell Pressure Initiation), Revision 4
QCAN 901(2)-3 F-15	Automatic Blowdown System in Test Mode of Operation, Revision 0
QCAN 901(2)-3 G-13	Automatic Blowdown System 1 and 2 Main dc Power Failure, Revision 0
QCAN 901(2)-3 G-14	Automatic Blowdown System in Inhibit, Revision 1
QCAN 901(2)-3 G-15	Reactor Vessel Low Low Level, Revision 5
QCARP 0000-01	Implementing Procedure for Appendix R Safe Shutdown, Revision 8
QCAP 0400-03	Setpoint/ Scaling Change Request, Revision 7
QCAP 0460-01	Quad Cities Nuclear Station Plant Design Process, Revision 0
QCEM 0700-06	HFA Relay Magnetic Core Assembly Replacement and Relay Calibration, Revision 8
QCEMS 0250-13	Dresser Electromatic Solenoid Actuator Environmental Qualification Surveillance Unit 1(2), Revision 9
QCIS 0200-20	Reactor High Pressure Automatic Blowdown Calibration and Functional Test, Revision 8
QCIS 0200-30	Reactor High Pressure Automatic Blowdown Calibration and Functional Test, Revision 8
QCOA 0203-01	Failure of a Relief Valve to Close or Reseat Properly, Revision 6
QCOP 0010-01	Winterizing Checklist, Revision 11
QCOP 0010-02	Required Cold Weather Routines, Revision 6
QCOP 0203-01	Reactor Pressure Control Using Manual Relief Valve Actuation, Revision 8
QCOS 0203-03	Main Steam Relief Valves Operability Test, Revision 15
QCOS 0203-06	Automatic Blowdown Logic Test, Revision 5
QCOS 1600-28	Pressure Suppression System Check Valve Testing - Cold Shutdown, Revision 4
QCTS 0210-02	Battery Charger Testing for Safety Related 125 Volt dc and 250 Volt dc Batteries, Revision 4
QCTS 0210B04	Setup and Use of the BCT-2000 Battery/ Charger Test Computer, Revision 0
QCTS 0230-11	125 Volt dc Alternate Battery Service Test, Revision 0
QCTS 0310-02	Unit 1 Relief Valve Logic Functional Test, Revision 8
QCTS 0310-04	Unit 2 Relief Valve Logic Functional Test, Revision 7
QCTS 0820-05	Inboard Main Steam Isolation Valve and Target Rock Valve Pneumatic System Leak Test, Revision 3
QGA 100	Reactor Pressure Vessel Control, Revision 4
QGA 200	Primary Containment Control, Revision 6

QOA 6900-02	Total Loss of Unit One 125 Volt dc Supply, Revision 16
QOA 6900-04	Total Loss of Unit 2 125 Volt dc Supply, Revision 12
QOA 6900-07	Loss of AC Power to the 125 Volt dc Battery Chargers with Simultaneous Loss of Auxiliary Electrical Power, Revision 7
QOP 6900-02	125 Volt dc Electrical System, Revision 20

Purchase Orders

205-H0270	Dresden II & III, Quad Cities I & Cities II Relief Valves, March 31, 1967
232265	Purchase of Three Safety-related (Part 21) 125 Volt dc Battery Chargers, December 18, 1978
359880	Main Steam Safety Valve Testing
325946	Purchase of Two Safety-related (Part 21) 125 V, 58-cell, Lead Calcium Batteries, Type NCX-1500, January 31, 1989
807532	Purchase of Two Safety-related (Part 21) 125 V , 58-cell, Lead Calcium Batteries, Type NCX-1500, July 31, 1989
822066	Repair/Refurbishment of Main Steam Safety/Relief Valves - Model 7367F, October 1, 1998

Safety Analysis Reports

CRN 98-06	Safe Shutdown Report Fire Hazards Analysis Report, revision 97-02

Self Assessments

October 30, 1999	Focus Area Self-Assessment for the Safety System Design and Performance Capability

Standards

IEEE Std 450-1995	IEEE Recommended Practice for Maintenance, Testing, and Replacement of Vented Lead-Acid Batteries for Stationary Applications
IEEE Std 484-1987	IEEE Recommended Practice for Design Installation and Installation of Large Lead Storage Batteries for Generating Stations and Substations
IEEE Std 485-1983	IEEE Recommended Practice for Sizing Large Lead storage Batteries for Generating Stations and Substations
ISA-S67.04-Part II	Methodologies for the Determination of Setpoints for Nuclear Safety-Related Instrumentation, September 1994
ISA-S67.04.08 Setpoints for Sequenced Actions, March 1996	
NEMA AB-3-1991	Molded Case Circuit Breakers and Their Applications
S&L ESC-190	Electrical Engineering Reference for Applications of Low Voltage Fuses, 11/12/84 (Proprietary)
S&L ESC-291	Electrical Engineering Reference for dc Auxiliary Power Systems, 5/23/89

Surveillances

QCIS 0200-20	Reactor High Pressure Automatic Blowdown Calibration and Functional Test, Unit 1, performed August 28, 1998
QCIS 0200-20	Reactor High Pressure Automatic Blowdown Calibration and Functional Test, Unit 1, performed November 26, 1998
QCIS 0200-20	Reactor High Pressure Automatic Blowdown Calibration and Functional Test, Unit 1, performed January 28, 1999

QCIS 0200-20 Reactor High Pressure Automatic Blowdown Calibration and Functional Test, Unit 1, performed April 6, 1999
QCIS 0200-20 Reactor High Pressure Automatic Blowdown Calibration and Functional Test, Unit 1, performed June 21, 1999
QCIS 0200-20 Reactor High Pressure Automatic Blowdown Calibration and Functional Test, Unit 1, performed September 15, 1999
QCIS 0200-20 Reactor High Pressure Automatic Blowdown Calibration and Functional Test, Unit 2, performed August 5, 1998
QCIS 0200-20 Reactor High Pressure Automatic Blowdown Calibration and Functional Test, Unit 2, performed October 21, 1998
QCIS 0200-20 Reactor High Pressure Automatic Blowdown Calibration and Functional Test, Unit 2, performed January 14, 1999
QCIS 0200-20 Reactor High Pressure Automatic Blowdown Calibration and Functional Test, Unit 2, performed April 7, 1999
QCIS 0200-20 Reactor High Pressure Automatic Blowdown Calibration and Functional Test, Unit 2, performed June 28, 1999
QCIS 0200-20 Reactor High Pressure Automatic Blowdown Calibration and Functional Test, Unit 2, performed September 23, 1999

Technical Specifications

3.2.B	Emergency Core Cooling Systems (ECCS) Actuation
3.5.A	Emergency Core Cooling System - Operating
3.6.E	Safety Valves
3.6.F	Relief Valves
3.9.C	D.C. Sources-Operating
3.9.D	D.C. Sources-Shutdown
3.9.E	Distribution-Operating
3.9.F	Distribution-Shutdown

Test Reports

46882-0	Wyle Test Report Performance and Recertification Testing of Dresser Electromatic Relief Valve (Type 1525VX-3) for GPU Nuclear, Inc., February 11, 1999

Training Modules

TRNOPSLP\LIC-0203.R06 Automatic Depressurization System, October 21, 1998

Updated Final Safety Analysis Report Sections

5.2	Integrity of Reactor Coolant Pressure Boundary
7.3.1	Emergency Core Cooling Systems Instrumentation and Control
8.3.2	dc Power Systems

Vendor Manuals

29-167F	Westinghouse AB DE-ION Breakers, Types EHD, FD and HFD, Rated 125 Vdc , Attachment C, Page C10, Westinghouse Application Data, (typical of several)
24081	Gould Shawmut Class RK5, Trionic, type TR, rated 250 Volts, Attachment G, page G2, Revision A
Bussmann 1-9	Bussmann NON and NOS Fuses Rated on Ac Only, Attachment C, page C17
C0004	Gould National Batteries- Manual for Flooded Batteries, May 1, 1995
C0022	Consoli Electromatic Relief Valve Manual 333, January 1986
C0025	Target Rock Relief Valve, October 08, 1997
C0029	GE Horizontal Metal Clad 4 kV Switchgear, April 12, 1999
C0036	GE Metal Clad Switchgear- 4 kV Vertical, May 25, 1999
C0100	Power Conversion- 125 Volt Battery Charger, Model 3S-130-200, September 21, 1998
IM 9165200002	Three Phase Thyristor Battery Charger
Dresser	Instructions for Operating and Repairing Consolidated Electromatic Relief Valve 6" 1525-VX, April 6, 1968

Work Requests

Q02778	Remove 7 Cells from Unit 2, 125 Volt dc Alternate Battery for Later Use to Replace Missing Cells in Unit 1 125 Volt dc Alternate Battery
Q14417	Unit 1 125 Volt dc Battery Charger #1 Repair for Cause of Current Oscillations
Q74516	Replace Unit 1 Battery with a New NCX-21 Battery from GNB
Q74517	Replace Unit 2 Battery with a New NCX-21 Battery from GNB
960033952	Unit 1 Hard Ground (Level III) Investigate and Isolate
960034700	Rebuild Spare ERV S/N BK7070 Valve Removed From 1-0203-3B
980065960	3A Target Rock Safety Valve Replacement
980024077	1-0203-3D Dresser Electromatic Relief E.Q. Inspection (pages 1,2,6,32)
980024078	1-0203-3C Dresser Electromatic Relief E.Q. Inspection (pages 1,2,6,32)
980024079	1-0203-3B Dresser Electromatic Relief E.Q. Inspection (pages 1,2,6,32)
980024786	1-0203-3E Dresser Electromatic Relief E.Q. Inspection (pages 1,2,6,32)