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Quad Cities Nuclear Power Station
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Telephone 309/654-2241

NJK-82-299

July 8, 1982

Mr. James G. Keppler, Regional Administrator
Directorate of Inspection and Enforcement
Region III
U. S. Nuclear Regulatory Commission
799 Roosevelt Road
Glen Ellyn, IL 60137

Subject: Quad-Cities Nuclear Power Station Units One and Two
Loss of Auxiliary Electrical Power Event of June 22, 1982,
Special Report
NRC Docket Numbers 50-254 and 50-265

Dear Mr. Keppler:

Enclosed please find the subject Special Report for your information and use. This report was written and is being submitted to you as per our commitment to Mr. Roger Walker of your office during a telephone conversation on June 23, 1982.

One original and ten (10) copies of the Special Report are provided.

Very truly yours,

COMMONWEALTH EDISON COMPANY
QUAD-CITIES NUCLEAR POWER STATION

N. J. Kalivianakis
Station Superintendent

NJK:LFG/bb

Enclosure

cc (w/enc) N. Chrissotimos
T. Rausch
J. Abel
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INPO Records Center
Station Distribution

IE01

COMMONWEALTH EDISON COMPANY

QUAD-CITIES NUCLEAR POWER STATION

UNITS 1 and 2

SPECIAL REPORT

LOSS OF AUXILIARY ELECTRICAL POWER

EVENT OF JUNE 22, 1982

NRC DOCKET NOS. 50-254 and 50-265

Submitted July 9, 1982

Special Report

Loss of Auxiliary Electrical Power
Event of June 22, 1982

A. Introduction and Background

Several months ago, an oil leak developed on the low voltage side bushing of Quad-Cities Station's Unit Two Reserve Auxiliary Power Transformer 22 (T22). It was suspected that there was a gasket leak, or a broken bushing that was the cause of the oil leakage. The decision was made for the CECo Sub-Station Construction Department to repair the oil leak during unit operation. A ten hour repair period was specified, based on prior experience on other transformer work. The nature of the oil leak was such that it warranted attention and the repair should not be put off to an indefinite time period. The oil leak could develop into a major problem (potential fire hazard) if the leakage became excessive.

T22 is situated in the Station auxiliary electrical power distribution system to take 345 KV power from the Station switchyard and step-down the voltage to 4 KV for in-house use on Unit Two. These in-house loads are normally split between the Unit Auxiliary Power Transformer 21 (T21) and T22. T21 takes 18 KV power from the Unit Two main Generator and steps the voltage down to 4 KV. A similar arrangement is provided for Quad-Cities Unit One. A single-line diagram of the essential constituents of the in-house electrical distribution system for both Units One and Two is provided as an attachment to this report.

The repair of T22 would necessitate that the transformer be de-energized. This would involve disconnecting its high voltage side from the switchyard, and switching its normal 4 KV loads (Buses 22 and 23) to T21. Procedure QOP 6100-3, Removing Reserve Auxiliary Transformer 22 From Service with Unit Two Operating, provides the necessary step-by-step actions to electrically isolate the transformer, as well as specifying the performance of Standby Diesel Generator (DG) operability surveillance on the Unit Two and the shared (1/2) Diesel Generator.

A major factor in scheduling the outage for T22 was the weather. It is necessary that the work be done when it is not raining. The unpredictable and poor weather conditions in May and early June caused repeated delays in scheduling a convenient time period to perform the repairs. The date of June 22, 1982, was selected and resolved as being the date to perform the work. At that time, the Unit One Diesel Generator had been previously rendered inoperable and was under a 7-day LCO permitted by the Technical Specifications. The cause of the inoperability was due to repairs being made on the motor for the associated cooling water pump for the Diesel. The decision to remove T22 from service with the Unit One Diesel Generator inoperable was made based on the availability and operability of the Unit Two and 1/2 Diesel Generators, and the fact that the unavailability of T22 does not directly have impact on the Unit One emergency power distribution system. Power would be available also from the normal source (main Generator) for each unit. Also, the relatively short time duration planned for the T22 outage influenced the decision.

B. Description of Event and Operator Actions

Unit Two was operating at 760 MWe (approximately 95 percent thermal power) when Bus 22 was inadvertently tripped at approximately 5:25 a.m. on June 22, 1982. The Equipment Operator was performing procedure QOP 6100-3, Removing Reserve Auxiliary Transformer 22 From Service with Unit Two Operating, when he mistakenly pulled out the 4 KV Bus 22 potential fuses instead of the potential transformer fuses from Transformer 22 (T22) to 4 KV Bus 22. Per the procedure, both the Unit Diesel Generator and the swing Diesel Generator (DG 1/2) were proven operable earlier that morning. Also, normal AC power had been transferred to the Unit Auxiliary Transformer (T21), and the Reserve Auxiliary Transformer (T22) had been disconnected from the switchyard.

When the Bus 22 tripped, the 2B Reactor Feed Pump also tripped causing a Reactor low water level scram. Approximately one minute later, the Unit Two Generator tripped, resulting in the loss of all normal AC power to Unit Two. During the time prior to the Unit Generator trip, the Unit Nuclear Station Operator (NSO) decreased the 2A Reactor Recirculation M-G Set speed and increased feedwater flow by opening a feedwater regulating valve. The rapid decrease in power and increased feedwater flow brought the Reactor water level back to about +48 inches.

When the Unit Generator tripped, both the Unit Two Standby Diesel Generator and the shared (1/2) Diesel Generator auto-started on under-voltage signals from their respective buses. Ten seconds later the Unit Two and 1/2 Diesel Generators were at speed and had automatically closed-in to 4 KV emergency Buses 24-1 and 23-1, respectively.

By 5:30 a.m., Buses 23 and 24 were back-fed from their respective emergency Buses, 23-1 and 24-1. At this time, the Unit NSO tried starting the HPCI System, but was unsuccessful due to the Reactor water level being greater than +48 inches, which is the high water level HPCI Turbine trip setpoint. Since the condenser was lost as a heat sink due to the tripping of the circulating water pumps, Reactor pressure increased to the Target-Rock relief valve, 203-3A, setpoint of 1105 psig. This valve auto-actuated four times by 5:38 a.m. Although the Acoustic Monitor for this valve did not annunciate, Reactor pressure did drop each time. It was decided at this point to open the 2-203-3C Electromatic Relief Valve (ERV) to minimize the challenges made to the Target-Rock valve. A few minutes later, the Operator had initiated Suppression Pool cooling on the B loop of the RHR System to minimize the Suppression Pool temperature.

It was then noticed that Reactor pressure was not decreasing, even though the red open light indication and the Acoustic Monitor were indicating that the 3C relief valve was open. The 203-3E ERV was then opened and a decrease in Reactor pressure was acknowledged.

At 5:47 a.m., the Operator started the 2A RHR Service Water pump for Suppression Pool cooling on the A loop. The Diesel Generator 1/2 had a load of approximately 500 KW prior to starting the pump. The Diesel Generator immediately tripped causing loss of power to 4 KV Buses 23-1 and 23 and 480V Bus 28. These buses ultimately feed the Instrument Bus, 2A Reactor Protection Bus, and the Essential Service Bus Static Inverter. However, it was determined later that the Essential Service Bus was never lost, since the inverter changed over to its 250 VDC power supply. Due to the vast number of alarms and lost indications which occurred when the

Instrument Bus and the 2A Reactor Protection Bus were lost, it was erroneously postulated that the Essential Service Bus was also tripped. With most of the Control Room indications not working, the Senior Operating Engineer quickly went to the local 2202-5 Instrument Rack in the Reactor Building and established communications with the Control Room. There, the Operating Engineer was able to give correct and accurate readings for Reactor pressure, and Reactor water level from the YARWAY level instrumentation. At about this same time, all the Electromatic Relief Valves were closed to give a rough indication of pressure every time the Target-Rock valve lifted.

It was approximately 5:50 a.m. at this point when a GSEP Unusual Event was declared. The SCRE promptly called the On-Call Duty Person, and notified him of the situation. The On-Call Duty Person quickly called the Station Director. An Operations Director was already on site in the person of the Senior Operating Engineer. All required personnel were notified wihtin approximately 15 minutes of declaring the GSEP event.

At 5:55 a.m., the Primary Containment (Drywell) pressure reached 2 psig. It was discovered later this was caused by leaking gaskets on the discharge line flanges of the main steam relief valves. Also, blind flanges that were in place awaiting future installation of vacuum relief valves on the discharge lines were loose which also caused steam to be blown into the Drywell.

Two psig Drywell pressure is an ECCS initiation signal at which the HPCI System began to inject water into the vessel. The Senior Operating Engineer noticed the rise in water level and informed the Control Room.

It is significant to mention that the HPCI System auto-injected at its proper setpoint and continued to function properly without operator assistance or the Diesel Generator 1/2 operating. The B Core Spray pump also auto-started as designed at the 2 psig initiation signal although the pump was not called upon to inject because Reactor pressure was greater than the 325 psig permissive setpoint for injection. The 2 psig Drywell pressure condition also tripped the running RHR Service Water pump, the Drywell coolers, and the Reactor Building Closed Cooling Water (RBCCW) pumps.

During the previous 30 minutes, Operating personnel were returning T22 to service and at 5:56 a.m. this was accomplished. By 6:04 a.m. all 4 KV buses were being fed from T22 and off site power. Instrument Bus power was also restored. The Diesel Generator 1/2 was inspected and it was discovered to have tripped on underexcitation. The Diesel was successfully restarted at 8:20 a.m. after the lockout relay was manually reset.

Reactor pressure was continued to be controlled by operation of the ERVs in the proper sequence to disperse the heat load in the Suppression Pool. By approximately 6:15 a.m., Suppression Pool cooling was re-established. The pool water temperature had not exceeded 110^{O}F. At approximately 7 a.m. Drywell cooling and RBCCW flow to the coolers was re-established to reduce Drywell pressure to below 2 psig. Throughout the remainder of the morning, Reactor pressure was decreased and proper cool-down rates were established. Cold shutdown was achieved at approximately 5 p.m.

That afternoon, an On-Site Review Committee meeting was held to identify the equipment that required repair and testing prior to startup.

This equipment included the 203-3C Electromatic Relief Valve which did not open when actuated. The valve was replaced with a spare and successfully tested during startup. The Acoustic Monitor on the 203-3A Target-Rock relief valve did not function when the valve opened. Repairs were completed and the monitor was properly tested before and during startup. An investigation of the 1/2 Diesel Generator trip was completed and the removal of the underexcitation relays resulted. Steam leaks on the relief valve discharge lines were apparent during this event. All gaskets on the flanged elbows on the discharge of the relief valves were replaced. The flanges on the discharge line, instead of the vacuum breakers, were found to be loose, and were subsequently tightened. All five relief valves were successfully tested during startup and no Drywell pressure spikes were experienced. The thermocouples on the discharge of the Electromatic Relief Valves 203-3E and 203-3B were repaired. Proper response was acknowledged prior to startup.

All repairs and testing were completed by Thursday, June 24, at 1:20 p.m., at which time control rod withdrawal commenced for Reactor startup. The Reactor was made critical at 2:45 p.m. on June 24, 1982.

C. Investigation of the Event

The investigative and review process pursuant to the event was begun and followed-through prior to Reactor startup, to identify significant items and affect timely corrective action. These items are resolved as summarized below:

1. Fuse-pull error: A review of the QOP 6100-3 procedure has found it to be clear, concise, and properly written to perform its intended function. The in-plant labeling of the fuse cabinets was also found to be adequate. A detailed investigation of the error is to be carried out and documented by senior Operations Department management, with a report submitted to the CECo Production Performance Policy Committee.
2. Diesel Generator 1/2 Overload Trip: Investigations as to the cause of the trip of the 1/2 Diesel Generator indicated that the trip was due to actuation of underexcitation relay 140-DG 1/2/CEH. Underexcitation relay targets were present after the trip at the local Diesel Generator 1/2 relaying and metering cabinet. The relay was installed in 1981 as a modification (M-4-1/2-77-29) as recommended by the Commonwealth Edison Company System Planning and Station Nuclear Engineering Departments. A test was performed on June 23, 1982, which duplicated the conditions of the Diesel Generator 1/2 trip on the previous morning. The Diesel was auto-started, loaded to 300 KW, and an RHR Service Water pump was manually started. The underexcitation relay actuated, and confirmed the cause of the problem. The On-Site Review Committee determined that the underexcitation relays should be removed from the Diesel Generator trip circuitry for all three Diesel Generators. Station Nuclear Engineering concurred with this decision. Removal of these relays does not affect the auto-start and loading functions of the Diesel Generator in an emergency situation. Dresden Station was informed of this phenomenon. Details of the Diesel Generator 1/2 trip event are documented by LER/RO 82-12/01T-0, a copy of which is attached to this report.

3. Loss of the Instrument Bus: Following the trip of Diesel Generator 1/2, 480V Bus 28 was lost, as it was being fed from 4 KV Bus 23-1. Motor Control Center (MCC) 28-2, which is fed by Bus 28, is the normal feed to the Unit Two Instrument Bus. MCC 25-2 is the reserve supply for the Instrument Bus, and there is an automatic transfer to the reserve supply upon loss of the normal feed. This transfer did not take place during the event because MCC 25-2 (fed from 480V non-vital Bus 25) was de-energized. As was provided to the NRC in our response to NRC I.E. Bulletin 79-27, loss of the Instrument Bus results in the unavailability of numerous instrument indications in the Control Room; these are: Feedwater flow, steam flow, GEMAC narrow-range and wide range Reactor water level, Drywell pressure, Suppression Chamber pressure, Suppression Pool level, RHR and Core Spray flow, Core Spray pressures, and various Recirc and Cleanup system instrumentation. Although these indications were lost, the ability to place the Reactor in a safe Cold Shutdown condition was not impaired. ECCS functions remained intact, and the YARWAY Reactor water level instrumentation was available. Further, as per procedure QOA 6800-1, local YARWAY Reactor water level instrumentation at instrument racks 2202-5 and 2202-6 was used to verify Reactor water level. Reactor pressure and Drywell pressure indication was also available at rack 2202-5. Loss of the Recirc, Cleanup, Core Spray, and RHR instrumentation was not significant; in that the Recirc pumps were off, the Cleanup System was isolated, and Core Spray and RHR were not injecting because Reactor pressure was above the 325 psig permissive setpoint for

injection. The Quad-Cities Station response to Bulletin 79-27 is attached to this report for information.

4. Electromatic Relief Valve 3C Failure to Open/3A Relief Valve Acoustic Monitor Failure: The 3C electromatic relief valve did not open when manually actuated during the early stage of the event. A spare valve was on-hand, and was installed to replace the failed valve. Details of the investigation concerning this failure are contained in LER/RO 82-09/03L-0, which is to be submitted within 30 days of the event. The 3A relief valve (Target-Rock) acoustic monitor was repaired and functionally tested prior to startup. Deviation Report D-4-2-82-28 documents this failure and corrective actions.

5. Drywell Pressurization: Pressurization of the Drywell to 2 psig occurred shortly after the loss of Diesel Generator 1/2. The 2 psig signal resulted in loss of Suppression Pool cooling and Drywell cooling. After the unit was in the Cold Shutdown condition, a Drywell entry was made and a thorough inspection was made of the steam piping. The flanged elbows on the discharge lines from several of the relief valves had gasket leakage. The multiple relief valve actuations and the steam leakage from these lines caused the Drywell pressure to increase to a maximum of 4.3 psig during the event. Flange leaks were also found on blind flanges on the relief valve discharge lines that had been installed during the last refuel outage. As part of the Mark I containment fix modification, new larger size vacuum breakers are to be installed on the relief valve discharge lines. Since the vacuum breakers were not available during the refuel outage, blind flanges were installed in their place. The vacuum breakers are to be installed

and the blind flanges removed during the next refuel outage. All gaskets on the flanged elbows on the discharge of the relief valves were replaced. The blind flanges were tightened.

6. Essential Service Bus: The Essential Service System (ESS) Bus on Unit Two is powered by an Uninterruptible Power Supply (UPS). A description of the UPS and a block diagram are provided as attachments to this report. The effects of a postulated loss of the entire ESS Bus are described in our response to NRC I.E. Bulletin 79-27, which is also attached to this report. It was first thought that ESS had tripped, and the UPS had not auto-transferred to its 250V DC backup source. Detailed investigations by Station and Operational Analysis Department personnel concluded that the bus had never tripped during the event, and did in fact auto-transfer. If the ESS Bus had been lost, the operability of the HPCI System would not have been compromised. Although the automatic flow control function would have been rendered inoperative, the HPCI Motor Speed Changer (MSC) and Motor Gear Unit (MGU) are DC-powered, and would have allowed HPCI to auto-start and inject water into the reactor at the maximum flow rate. The HPCI protective turbine trip and auto-isolation functions would also be unaffected by a loss of ESS to the flow controller.

D. Conclusions and Recommendations

Further investigations have been performed relative to this event and possible future Auxiliary Power Transformer Maintenance. As previously committed to the NRC, it is our intent not to remove Transformers 12 or 22

from service during plant power operation, unless an emergency dictates this action, until the NRC has reviewed this report and released the Station from this commitment.

An evaluation has been conducted by the Station, the C.E.Co. Station Electrical Engineering and System Planning Departments, and General Electric Co. as to the possibility of performing maintenance on T22 during a unit shutdown with offsite power being supplied to the unit from T21 via Unit 2 Main Power Transformer (T2). Such an evolution would require that the copper links be pulled on the Unit 2 main Generator output, and that numerous jumpers be installed to bypass interlocks between and involving the generator, switchyard components, and the transformers.

The evaluation concluded that such a maneuver is possible; however, any decision to do this would be needed to be weighed against the impact of shutting the unit down for a scheduled 10-hour repair job on T22.

As has been described earlier, a GSEP Unusual Event condition was declared shortly after the DG 1/2 tripped. Although <u>all</u> off-site power was not lost to the Station as given by Emergency Action Level No. 10 in the procedure QEP 200-T1 table, this GSEP response to the event was conservative. The GSEP response to the inoperability of DG 1/2 as related to Unit 1 is described in LER/RO 82-12/01T-0.

Recommendations as a result of the event have been discussed at the Station and are as follows:

1. Procedure revisions have been submitted for various QGA and QOA procedures to add clarifying steps and add information which may be

helpful to operations personnel.

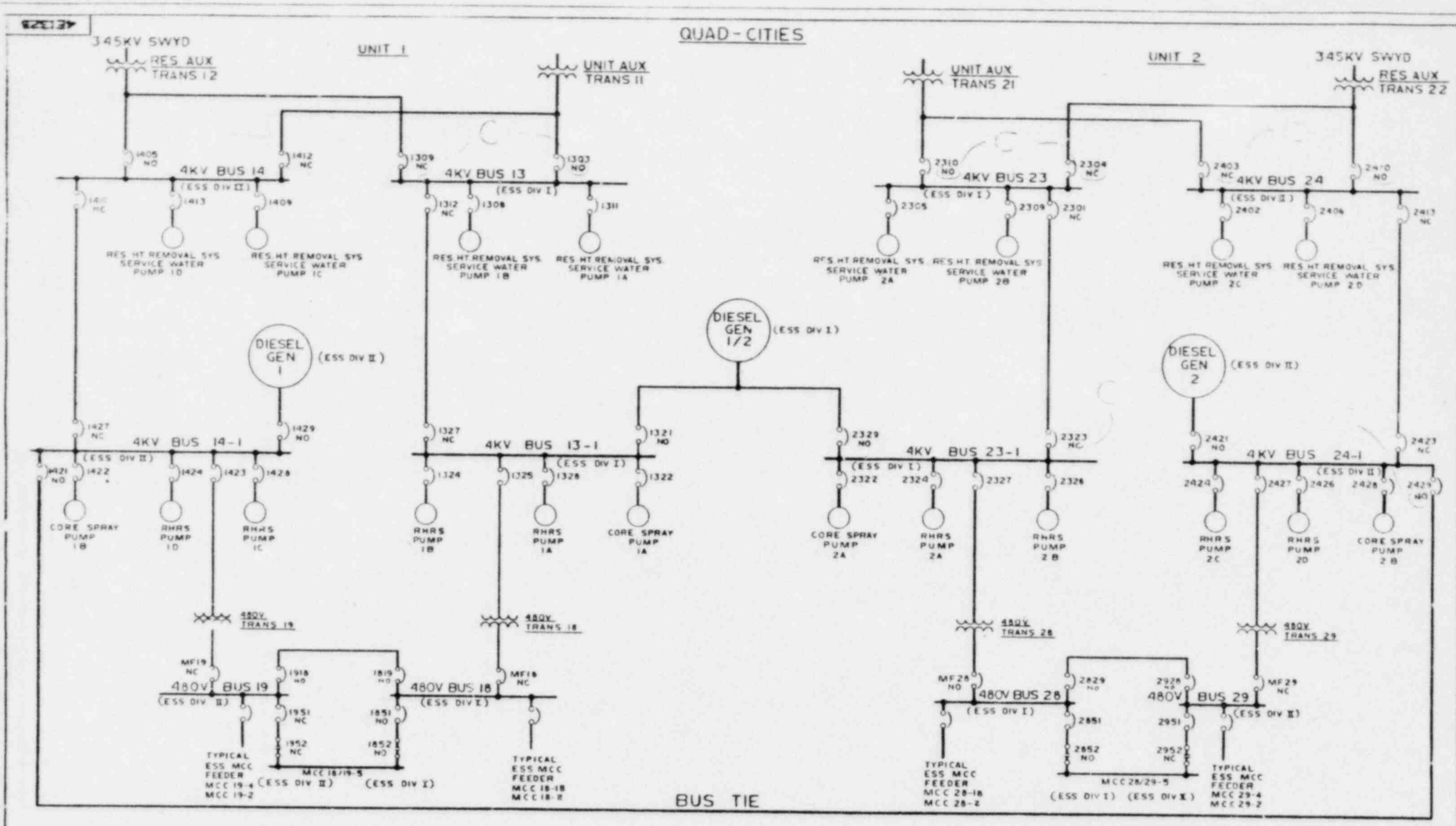
2. An Action Item Record (AIR-4-82-18) has been issued for the C.E.Co. Station Nuclear Engineering Department (SNED) to evaluate a proposed change to the Diesel Generators' auto-start circuitry to provide a seal-in circuit to allow all of the protective trips to be blocked until normal power is restored. This topic is discussed in LER/RO 82-12/01T-0.
3. An Action Item Record (AIR-4-82-19) has been issued to SNED to evaluate a proposed change to the Instrument Bus and ESS Bus to provide more reliable backup feeds to these buses. A possible solution would be to change the ESS Bus reserve feed from MCC 28-2 to MCC 29-2, and to have the Instrument Bus reserve feed changed from MCC 25-2 to MCC 29-2. The same would apply for Unit 1.
4. Modifications M-4-1(2)-80-31 are to be installed during the next refuel outages; which modify the Core Spray logic so that the drywell coolers and RBCCW pumps do not trip on a Core Spray initiation (e.g. 2 psig drywell pressure) if power is available to the emergency buses.

In conclusion, it is recognized that the event described in this report was significant considering the unavailability of two diesel generators at the Station and the loss of off-site power to Unit 2. However, the actions taken by the operators and their supervisors were timely and attentive to the adverse situations that were presented to them. Emergency and abnormal procedures were followed for the degraded electrical distribution conditions on both Units. The professional performance and cooperation of these people, as well as the

Senior Operating Engineer, the Shift Technical Advisor, and both NRC Resident Inspectors were exemplary. At no time during this event was there evidence of confusion or misunderstandings. The reactor was never out of control, and the health and safety of the Station employes and the general public were never in a state of compromise during the event.

Attachemnts

1. Single Line Diagram - Emergency Power System
2. LER/RO 82-12/01T-0
3. Response to NRC I.E. Bulletin 79-27
4. Unit 2 Essential Service UPS Block Diagram and System Description



NUCLEAR SAFETY RELATED
EQUIPMENT IS SHOWN ON THIS DRAWING.

REV.	DATE	DESCRIPTION
B		ADDED SAFETY RELATED EQUIP.
C		FOR RECORD DWG. CHAN. CECO REV.

SINGLE LINE DIAGRAM
EMERGENCY POWER SYSTEM
QUAD-CITIES STATION UNIT 1&2
GENERAL ELECTRIC CO.
FOR
COMMONWEALTH EDISON CO.
CHICAGO, ILLINOIS

SARGENT & LUNDY
INCORPORATED
ENGINEERS
CHICAGO

4E-328

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NJK-82-288

July 1, 1982

J. Keppler, Regional Director
Office of Inspection and Enforcement
Region III
U. S. Nuclear Regulatory Commission
799 Roosevelt Road
Glen Ellyn, IL 60137

Reference: Quad-Cities Nuclear Power Station
Docket Number 50-254, DPR-29, Unit One
Appendix A, Section 3.9.E.1

Enclosed please find Reportable Occurrence Report Number RO 82-12/01T-0 for Quad-Cities Nuclear Power Station. This occurrence was previously reported to Region III, Office of Inspection and Enforcement by telephone on June 22, 1982, and by telecopy on June 23, 1982.

This report is submitted to you in accordance with the requirements of Technical Specification 6.6.B.1.e.; malfunction of the functional requirements of the Diesel Generator System as used to cope with accidents analyzed in the SAR.

Very truly yours,

COMMONWEALTH EDISON COMPANY
QUAD-CITIES NUCLEAR POWER STATION

N K Tomlynson

N. J. Kalivianakis
Station Superintendent

NJK:GCT/bb

Enclosure

cc T. J. Rausch
N. Chrissotimos
INPO Records Center

8207130446

U.S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT

CONTROL BLOCK: [| | | | |] (1) (PLEASE PRINT OR TYPE ALL REQUIRED INFORMATION)

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LICENSEE CODE | LICENSE NUMBER | LICENSE TYPE | CAT

0 1 | REPORT SOURCE L (6) | DOCKET NUMBER 0 5 0 0 0 2 5 4 (7) | EVENT DATE 0 6 2 2 8 2 (8) | REPORT DATE 0 7 0 1 8 2 (9)

EVENT DESCRIPTION AND PROBABLE CONSEQUENCES (10)

Following a scram on Unit Two, from loss of off-site power, both the Unit Two and 1/2 Diesel Generators loaded to their respective emergency buses. The 1/2 Diesel Generator subsequently tripped when the "A" RHRSW pump was started. The Unit One Diesel Generator was out of service at this time. Therefore, contrary to Technical Specification 3.9.E.1, Unit One was operating without its Unit or Shared Diesel Generator. Safety implications were minimal since provisions were being initiated to shutdown the Reactor. Also, both on-site and off-site AC power were available to supply the Unit One loads. Plus, the Unit battery systems and the High Pressure Core Injection System were available.

SYSTEM CODE (11)	CAUSE CODE (12)	CAUSE SUBCODE (13)	COMPONENT CODE (14)	COMP SUBCODE (15)	VALVE SUBCODE (16)
E E	B	A	G E N E R A	A	

(17) LER/RO REPORT NUMBER	EVENT YEAR		SEQUENTIAL REPORT NO.		OCCURRENCE CODE	REPORT TYPE		REVISION NO.
	8 2	—	0 1 2	/	0 1	T	—	0

ACTION TAKEN (18)	FUTURE ACTION (19)	EFFECT ON PLANT (20)	SHUTDOWN METHOD (21)	HOURS (22)	ATTACHMENT SUBMITTED (23)	NPRD-4 FORM SUB. (24)	PRIME COMP SUPPLIER (25)	COMPONENT MANUFACTURER (26)
F	F	Z	Z	0 0 0 0	Y	N	A	G 0 8 0

CAUSE DESCRIPTION AND CORRECTIVE ACTIONS (27)

Investigation of the 1/2 Diesel Generator trip revealed it had tripped on under-excitation. This trip should have been locked-out after the Diesel had auto-started on bus undervoltage. A design deficiency was discovered, in that, the Auto-Start Relay should have sealed-in which would have blocked the underexcitation trip. To prevent re-occurrence, the underexcitation relays on all three Diesel Generators were temporarily removed until a permanent design change could be completed. Action Item Record 4-82-18 was written to the Station Nuclear Engineering Department to support this work.

FACILITY STATUS (28)	% POWER (29)	OTHER STATUS (30)	METHOD OF DISCOVERY (31)	DISCOVERY DESCRIPTION (32)
E	0 5 3	NA	A	Operational Event

ACTIVITY RELEASED (33)	CONTENT OF RELEASE (34)	AMOUNT OF ACTIVITY (35)	LOCATION OF RELEASE (36)
Z	Z	NA	NA

PERSONNEL EXPOSURES

NUMBER (37)	TYPE (38)	DESCRIPTION (39)
0 0 0	Z	NA

PERSONNEL INJURIES

NUMBER (40)	DESCRIPTION (41)
0 0 0	NA

LOSS OF OR DAMAGE TO FACILITY (43)

TYPE (42)	DESCRIPTION
Z	NA

PUBLICITY

ISSUED (44)	DESCRIPTION (45)
N	NA

8207130455

NRC USE ONLY

NAME OF PREPARER Daryl G. Clark

I. LER NUMBER: LER/RO 82-12/01T-0

II. LICENSEE NAME: Commonwealth Edison Company
Quad-Cities Nuclear Power Station

III. FACILITY NAME: Unit One

IV. DOCKET NUMBER: 050-254

V. EVENT DESCRIPTION:

At 0526 hours on June 22, 1982, the Unit Two Reactor tripped from a Reactor Feedwater pump trip and subsequent low water level. The pump trip resulted from the loss of Bus 22 while the reserve auxiliary transformer 22 was being removed from service for maintenance. The Unit Two and Unit 1/2 Diesel Generators automatically started on a bus undervoltage signal and loaded to their respective buses. At approximately 0547 hours, Bus 23 was energized from Bus 23-1 in order to initiate the Containment Cooling mode of the Residual Heat Removal System (RHR). When the Operator attempted to start the 2A RHR Service Water pump, the Unit 1/2 Diesel Generator tripped. Several attempts were made to restart the Diesel Generator, but without success. Off-site power to Unit Two was restored at 0555 hours and Bus 23-1 was energized by normal means at 0604 hours. At 0805 hours, when the Unit Two scram recovery had returned to normal, the Unit 1/2 Diesel Generator was declared inoperable. During the time of this event, the Unit One Diesel Generator had been inoperable while the Diesel Cooling Water pump was being overhauled. Due to the degraded mode of the Unit One emergency AC power system, a Generating Station Emergency Plan Unusual Event was declared and the Nuclear Regulatory Commission personnel, both on-site and in Bethesda, Maryland, were notified. Technical Specification 3.9.E.1 states that when both the unit and shared Diesel Generator are inoperable, the Reactor must be in the Cold Shutdown condition within 24 hours. The Diesel Generator trip was being investigated and provisions were being initiated to shutdown the Unit One Reactor. Immediately following the unusual event notification, the Generator lock-out relay for the Unit 1/2 Diesel Generator was found to be tripped. At 0820 hours, the lock-out relay was reset; and the Diesel Generator was started and loaded to Bus 23-1 for a two hour operability surveillance.

The operability surveillance was successfully completed, and the unusual event and provisions for Unit One shutdown were cancelled.

VI. PROBABLE CONSEQUENCES OF THE OCCURRENCE:

Between the time that the Unit 1/2 Diesel Generator tripped and off-site power was restored to Unit Two, the Unit Two Diesel Generator performed as designed to supply power to its designated equipment in order to safely shutdown the Reactor. This fact is emphasized in the Final Safety Analysis Report 8.2.3.1, stating that one Diesel Generator is capable of sustaining the necessary power for a safe shutdown of one unit in case of a loss of off-site power. Power to Division I equipment, which is fed from Bus 23-1,

VI. PROBABLE CONSEQUENCES OF THE OCCURRENCE: (Continued)

was de-energized for only 17 minutes, after which time off-site power was restored. All of the Unit Two battery systems were also available and supplied power to their associated equipment over the duration of the event.

Both on-site and off-site power were available to supply the Unit One loads while the Unit 1/2 Diesel Generator was inoperable. Incoming power was available from four outside lines; and the reserve auxiliary transformer was available at all times. The Unit One battery systems were available to operate the High Pressure Coolant Injection System, Reactor Core Isolation Cooling System, and the Automatic Blowdown System. Station procedures exist to place the Reactor in a Hot Shutdown condition without any AC power available. Following the restoration of power to the Unit Two reserve auxiliary transformer, it would have been possible to feed Unit One Division II AC power with the Unit Two Diesel Generator through the Bus 14-1/Bus 24-1 cross-tie. Therefore, had Unit One lost off-site power at this point, a Diesel Generator would have been available to safely shutdown the unit.

The Unit 1/2 Diesel Generator was proven operable four and one-half hours following the trip; thus, the 24 hour Cold Shutdown requirement in Technical Specification 3.9.E.1 was not required to be implemented. Due to the large amount of redundancy in the electrical distribution system, the battery powered high pressure ECCS systems and the short duration of the Diesel Generator outage, safe shutdown of the Unit One Reactor could have been achieved.

VII. CAUSE:

The cause of this occurrence was a design error in the Diesel Generator control logic system. The Diesel Generator trip occurred from an actuation of the underexcitation relay in the Diesel exciter cabinet. The underexcitation relay monitors the phase angle of the Generator output leads. A large phase angle, or low power factor in an AC Generator is a good indication of abnormally low field excitation. When the Operator attempted to start the RHR Service Water pump, a high starting current and extremely low power factor was experienced. This phenomenon is typical when starting a large AC induction motor. The low power factor, approximately 0.40 instead of the normal 0.80, caused the underexcitation relay to actuate and trip the Generator lock-out relay.

An automatic start signal to the Diesel Generator will block all the protective trips except for the phase differential current trip. The Diesel Generator automatically started when the undervoltage relays on Bus 23 & Bus 23-1 energized. After the Generator was loaded to the Bus 23, normal voltage was restored and the undervoltage relays de-energized. This caused the automatic start relay to reset, thus engaging all of the Diesel Generator protective trip functions. The underexcitation relay is

VII. CAUSE: (Continued)

designed to protect the Generator during testing when the Diesel Generator is loaded to an energized bus. When the bus is not energized, the power factor will drop and can cause the underexcitation relay to actuate when a large motor is started.

When the Diesel Generator tripped, several attempts to restart the Diesel were made by the Control Room Operator. The Diesel would not start because the Generator lock-out relay had not been manually reset. The Equipment Operator had been dispatched to the switchyard to expedite the restoration of off-site power to Unit Two. For this reason, the Generator lock-out relay was not immediately reset.

VIII. CORRECTIVE ACTION:

The immediate corrective action was to restore normal off-site power to Unit Two. Power was restored 17 minutes after the Diesel Generator tripped. The Generator lock-out relay was reset and the two hour operability surveillance was successfully completed four and one-half hours following the trip. These actions satisfied the Limiting Conditions for Operation in Technical Specification 3.9.E.1, and Unit One continued normal operation.

To correct the design error in the Diesel Generator automatic start control logic, Action Item Record 4-82-18 has been initiated to provide a seal-in circuit to the automatic start relay. The seal-in circuit will allow all the protective trips to be blocked until normal power is restored. Until this change is completed, the underexcitation relays on all three Diesel Generators have been removed. The removal of these relays will have no affect on the emergency capability of the Diesel Generators. This temporary change will adequately correct the design error and prevent any chance of recurrence.

QUAD-CITIES UNITS 1 AND 2

RESPONSE TO IE BULLETIN 79-27

1. Based on review of this Bulletin, the following buses are addressed in response to the Bulletin, for each unit:

   - Reactor Protection Bus A and B
   - Essential Service Bus
   - Instrument Bus
   - 24/48 VDC Bus A and B

Reactor Protection System (RPS) Bus A and B

Each 120 V.A.C. RPS bus is supplied normally by an M-G set, which in turn is fed from a 480 V Motor Control Center (MCC 18-2 and 19-2 for Unit 1, MCC 28-2, 29-2 for Unit 2). These MCC's are supplied from emergency 480V buses 18 and 19 (28 and 29), which may be fed from their respective emergency diesel generators. A reserve supply to each RPS bus exists from 480V MCC 15-2 (25-2). This reserve supply is interlocked with the normal supply by use of a keylock switch to prevent simultaneous feed from both sources.

A. Loss of power to either RPS bus, even momentarily, will give a half-scram condition, and partial Groups II and III primary containment isolations (PCI). Control Room alarm CHANNEL A(B) MANUAL SCRAM would annunciate, indicating loss of power to the A(B) RPS bus. Numerous other control room alarm indications would be present, including all of those associated with the particular RPS trip system that was de-energized. If both RPS buses are lost on the unit, a full reactor scram will occur. Group II and Group III PCI will also take place, as will half the permissive for a Group I isolation. Numerous control room alarms, as well as the reactor and turbine-generator parameter changes associated with a scram, will more than adequately alert the operator to a loss of power to the RPS buses. Indications exist in the Auxilairy Electrical Equipment Room for Bus Amps, Bus Volts, and Generator Volts.

B. The loads on the RPS buses are as follows:

Bus 1A(2A) - (1) Process Radiation monitoring panel 901-10 (902-10)

(2) PCI Division I, relay panel 901-40 (902-40)

(3) RPS Division I relay panel 901-15 (902-15)

(4) Power Range Neutron Monitoring panel 901-37 (902-37)

Bus 1B (2B) - (1) PCI Division II relay panel 901-41 (902-41)

(2) RPS Division II relay panel 901-17 (902-17)

(3) Power Range Neutron Monitoring panel 901-37 (902-37)

(4) Process Radiation Monitoring Panel 901-10 (902-10)

As mentioned above, loss of power to both of the RPS buses will result in a full reactor scram and Groups II and III isolations. From this logic, a mechanical vacuum pump trip and SJAE suction valve closure will also occur. A Group I isolation will not occur, since the redundant power supply to the isolation relays will still be energized from the Essential Service Bus. Further, the D.C. Solenoids for the MSIV's and the solenoids for the Primary Sample Valves fed from the Instrument Bus will remain energized, and the valves will remain open. Thus, ability to achieve and maintain a cold shutdown condition is not impaired, since the reactor has been shutdown, all rods are inserted, normal condensate-feedwater and associated level control are available, and the main condenser is available as a primary heat sink. All ECCS also would be available in this condition. With the reactor in this condition, there is no need to rely on the power range neutron monitoring panel (901-37, 902-37). Loss of power to panel 901-37 (902-37) will disable the Reactor Recirculation Loop Flow instrumentation. This is of minimal concern, since the Recirculation Pumps would be at minimum speed. Pump speed, jet pump flow, and total core flow instrumentation would still be available to monitor Recirculation System performance. Loss of RPS power to panel 901-10 (902-10) would result in a Reactor Building Vent Isolation, Vent Fan Trip, and SBGTS Auto-Start from loss of power to the Reactor Building Vent and Fuel pool Radiation Monitors. Loss of power to the Off-Gas SJAE Radiation Monitors will initiate the 15-minute isolation timer. The timer can be reset to inhibit an off-gas isolation so the main condenser would remain available. No Main Steam Line High Radiation Monitor Group I isolation trip will occur, since the B and D monitors are supplied from the Essential Service Bus. The above evolutions do not affect the ability to achieve a cold shutdown condition. A loss of a single RPS bus would be less significant, since redundant monitors would still be operating.

C. No modifications are proposed based on the above review.

<u>Essential Service Bus</u>

The 120/240 V.A.C. Essential Service System (ESS) Bus is normally fed from MCC 18-2 (28-2), with a reserve feed from a M-G Set. The M-G Set may be powered from 480 V.A.C. Bus 18 (28), or from 250 V.D.C. Bus 1 (2). An auto-bus transfer exists from MCC 18-2 (28-2) and the M-G Set in the event the MCC is lost. If A.C. power to the M-G Set is lost (i.e. loss of Bus 18 (28)) the D.C. Motor is energized to continue running the M-G Set.

A. The following alarms persuant to the ESS Bus exist in the control room:

120/240 VAC ESS BUS LOW VOLTAGE
ESS M-G SET A.C. FEED TRIP
120/240 VAC ESS BUS ON M-G SET
ESS M-G SET D.C. FEED TRIP

Since the B and D Main Steam Line Radiation Monitors are powered by the ESS Bus, Alarms will come up for CHANNEL B REACTOR SCRAM, CHANNEL B MAIN STEAM HIGH RADIATION and GROUP I ISOLATION CHANNEL B TRIP if ESS is lost. Alarms for loss of power to the process computer, recirc flow limit, RPIS inoperative, and low vessel water level will also come up. Indications exist in the Auxiliary Electrical Equipment Room for the following:

AC Motor Amps and Volts
Transfer Volts
AC Generator Amps and Volts
Frequency
Bus Amps and Volts

B. The following loads are connected to the ESS Bus:

Control Room Panel 901-4 (902-4)
Control Room Panel 901-5 (902-5)
Turbine and Auxiliary Panel 901-7 (902-7)
Process Radiation Monitor Panel 901-10 (902-10)
RPS Division II Relay Panel 901-17 (902-17)
FW and Recirculation Panel 901-18 (902-18
Hydrogen and Stator Cooling Panel 2251-7 (2252-7)
SBGT Local Panel 2212-29A (2212-29B)
Main Chimney Instrument Rack 2212-5
RPIS Relay Panel 901-27 (902-27)
Computer Panel 901-45 (902-45)
Computer Main Feed Cabinet
Computer Peripheral and Transducer Power Supply Cabinet
FW Controls, Panel 901-5 (902-5)
Radiation Recorder, Panel 912-4
SRM, IRM, APRM Recorders, Panel 901-5 (902-5)
Control Room Panel 901-3, (902-3)
Recirculation M-G Set Protection Auxiliary Relay Panel 2201-25A (2202-25A)
HPCI FLow Control Unit 2340-1
Main Chimney Sample Vacuum Pump
Control Room Panel 901-6 (902-6)
Feedwater Instrument Rack 2251-9 (2252-9)
B and D Main Steam Line Radiation Monitors
HPCI Signal Converter 2386
Off-Gas Panel 901-54 (902-54)
Safety/Relief Valve Acoustic Position Monitor

A total loss of ESS for a sustained period of time is very unlikely due to the diverse backup power sources available, and the reliability of these sources. These sources are switched over automatically, and the bus is intended to remain in continuous service even if all other sources of A.C. electrical power (including the diesel generators) should fail. Both A.C. power supplies came from an emergency 480 V bus, supplied by a Diesel Generator. The 250 VDC source is reliable, since it comes from the Station Battery System.

A loss of ESS for a period of time will not directly cause a reactor scram, although a shutdown will probably occur from either reactor vessel low or high water level. This would be due to the loss of the feedwater level control system. The feedwater pump minimum flow valves will also fail open. Reactor water level narrow-range GEMAC indication will be available, but the associated level recorder and YARWAY narrow and wide-range indicators will not. Local manual operation of the feedwater regulating valves could still be accomplished.

Reactor pressure, feedwater flow, and steam flow indication would still be available. The recirculation pumps' speed control system and indication would be lost. HPCI and RCIC flow control devices would be unavailable. No PCI would take place, although half of the logic permissive would be satisfied. An Off-Gas isolation and SJAE suction valve closure would occur. The chimney activity recorder would be lost, but the chimney and SJAE monitors would still be available. Although one SBGTS train would be unavailable, the redundant train would be available. ADS and all low pressure ECCS would be intact and available, as would the Shutdown Cooling Mode of RHRS. Of relatively minor significance in achieving cold shutdown, the following items would not be available:

- RCIC testable check valve test feature
- CRD flow control (Scram function not affected)
- APRM & RBM indication lights
- SRM, IRM, APRM recorders (Indication would still be available).
- Turbine overspeed trip test feature
- Stator cooling and Seal Oil local instrumentation (Runback circuit not affected)
- RPIS, Rod Select, RWM
- Off-Gas system instruments and valve indications
- RHR SW instrumentation
- Process computer
- Recorders for feedwater flow, steam flow, & reactor pressure (Indicators still available)

Based on the availability of water for the vessel, the availability of ADS and low pressure ECCS, the operability of CRD scram and rod block functions, the initial availability of the main condenser, and the use of Shutdown Cooling, a sustained loss of ESS, although extremely improbable, would not effect the ability of the reactor to achieve a cold shutdown condition.

C. No additional modifications are proposed based on the above review.

INSTRUMENT BUS

The 120/240 V.A.C. instrument bus is noramlly fed from MCC 18-2 (28-2), with a reserve supply from MCC 15-2 (25-2). There is an automatic transfer to the reserve supply upon loss of the normal feed.

A. Alarms exist for the following in the Control Room:

120 VAC INSTRUMENT BUS LOW VOLTAGE
120 VAC INST. BUS TRANSFER TO EMERGENCY SUPPLY

Bus voltage indication is available at the bus cabinet in the auxiliary electrical equipment room. Alarms would likely come up indicating heater drain valve closure, MSDT and moisture separator high level, and 24/48 VDC battery charger trip upon loss of the instrument bus.

B. The following loads are connected to the instrument bus:

Rod control relay panel 901-28 (902-28)
Area radiation monitors, panel 901-11 (902-11)
Generator voltage regulator cubicle
Protection relay panel 901-29 (902-29)
Process instruments, panel 901-19 (902-19)
Cleanup system instrument rack 2201-2 (2202-2)
Containment cooling benchboard 901-3 (902-3)
Area leak detection temperature monitoring panel 901-21 (902-21)
EHC panel 901-31 (902-31)
Control Room Panel 901-4 (902-4)
Control Room Panel 901-5 (902-5)
Feedwater and recirculation panel 901-18 (902-18)
Accumulator panels 2201-16 & 20 (2202-16 & 20)
Turbine panel 901-7 (902-7)
SRM/IRM drive control panel 2201-14 (2202-14)
Jet pump panel 901-38 (902-38)
HPCI panel 901-39 (902-39)
24/48 VDC Battery Chargers A and B
FW heater racks 2251-14A, 14B, 16 (2252-14A, 14B, 16)
Process radiation monitors & panel 912-4
Radwaste control panel 2212-4
ECCS relay panels 901-32 & 33 (902-32 & 33)
Pressure suppression panel 901-3 (902-3)
Recirc. M-G Set panels 2201-25A & B (2202-25 A. & B)
Annunciator input relay cabinet 901-34 (902-34)
Oxygen Analyzer panel 912-7
345 KV meter panels 912-400 & 912-402
345 KV frequency recorder, panel 912-2
Service building telephone room service
PCI panels 901-40 & 41 (902-40 & 41)

shutdown the reactor would remain intact. Since the YARWAY level instrumentation would be available, close monitoring of the reactor vessel level could still be accomplished. Of less relative importance, the following items would be disabled upon loss of the instrument bus:

Generator ground detection
Indication for HPCI drain valves
Test feature for HPCI and RHR testable check valves
Area radiation monitors
Turbine Lift Pumps
Pressure/level monitoring of CRD accumulators
SBLC tank level indication
Condenser Seal trough valves
Condensate make-up pump auto-start feature
Area leak detection
Reset feature of Group I, II, and III PCI
SJAE suction valve position indication
Process radiation recorders (indicator and trip units would still be available)
Cleanup drain flow regulator
Control Room Vent Isolation feature
Makeup Demineralizers
Stack gas sampling
Radwaste Instrumentation and Valves
AO 5406 valve position

C. No modifications are proposed based on the above review.

24/48 V.D.C. Buses

A. The 1A (2A) and 1B (2B) distribution panels are supplied by the 24/48 V.D.C. batteries and battery chargers. Each battery is rated at 80-ampere hours. Voltage indication is provided in the Control Room. An alarm exists in the Control Room for battery undervoltage. Alarm indication for neutron monitor low supply voltage would come up in the event of a loss of 24/48 VDC. Local indications exist in the Unit 1(2) Battery Charger Room for: Charger Volts and Amps, Bus A Positive and Negative Volts, and Bus B Positive and Negative Volts.

B. The following loads are connected to the 24/48 V.D.C. system:

Startup Range Neutron Monitoring (SRM/IRM) Panel 901-36 (902-36)

Process Radiation Monitor Panel 901-10 (902-10)

Stack Gas Monitor Panel 912-4

Loss of the entire 24/48 V.D.C. system will disable the process liquid radiation monitors (RBCCW, Service Water, Radwaste Effluent)

125 VDC turbine bldg panel 1A (2A) ground detection
Recirc sample valves AO-220-44 and AO-220-45
Control Room Fire Protection HVAC panel 2212-89
Control Room Panel 901-8 (902-8)
Panel 2251-45 (2252-45)
Panel 2251-45 (2252-45)
Heating boiler pressure indicator, panel 912-5
Refuel bridge telephone
FW regulating valves' position indication
Off-gas panel 901-54 (902-54)
Relief and safety valve acoustic position monitors

A complete loss of the instrument bus would not directly cause a reactor scram, but would likely cause a loss of all low pressure and high pressure feedwater heater level control. The normal drains would fail closed, and the extraction steam dump valves would open. This would probably result in a high moisture separator level turbine trip and subsequent reactor scram. Of added significance is that the reactor manual control system would incur rod blocks, and indicating lights for accumulator trouble, scram, and scram discharge volume valve position would be lost.

Scram could be verified from the rod position indication system and a power decrease. Indications in the control room for feedwater regulating valve position, feedwater flow, steam flow, and GEMAC reactor water level would be lost. The feedwater controllers, regulating valves, and both narrow range YARWAY level indications would be available. Of significance is the loss of the 24/48 V.D.C. battery chargers. However, the batteries themselves would provide their designed function. Numerous other indications would be unavailable, such as RHR flow, Core Spray flow, Core Spray pressures, wide-range reactor water levels, drywell pressure, suppression chamber pressure, suppression chamber level, recirc pump and M-G set temperatures, CRD pressures and flows, jet pump flows, recirc pump seal pressures, cleanup flow and temperature, cleanup conductivity, recirc pump DP and flows, etc. Both SBGTS trains would start and the reactor building vent system would isolate. The recirc sample valves 220-44 and 220-45 would go closed and the TIP ball valves would close, if open. The reactor building-to-suppression chamber vacuum breakers would open, but in-line check valves would preclude venting the suppression chamber to the reactor building atmosphere. Although many indications would be lost, the ability to place the reactor in a safe cold shutdown condition would not be impaired. Normal feedwater, the shutdown cooling mode of RHRS, ECCS, and the main condenser heat sink would be available. No PCI would take place, and the ability to

and the main chimney effluent monitors. Since the trip logic for the SJAE off-gas radiation monitors comes from the 24/48 VDC System, the isolation function from high radiation would be disabled. However, the Main Steam Line Radiation Monitors and trips would be available, as would indication from the SJAE monitors. Loss of the 24/48 V.D.C. system will also disable the SRM and IRM Systems. If not in the RUN Mode, this will de-energize the associated IRM protective relays and cause a reactor scram. The indications, recorders, and lights would also be lost. Loss of the entire system is very unlikely, and a partial system loss (A or B) would have minimal implications as far as obtaining a cold shutdown condition. Even for a complete 24/48 V.D.C. system loss, the reactor can still be brought to a cold shutdown condition due to the availability of normal feedwater, ECCS, the main condenser, reactor normal control, and the shutdown cooling mode of RHRS.

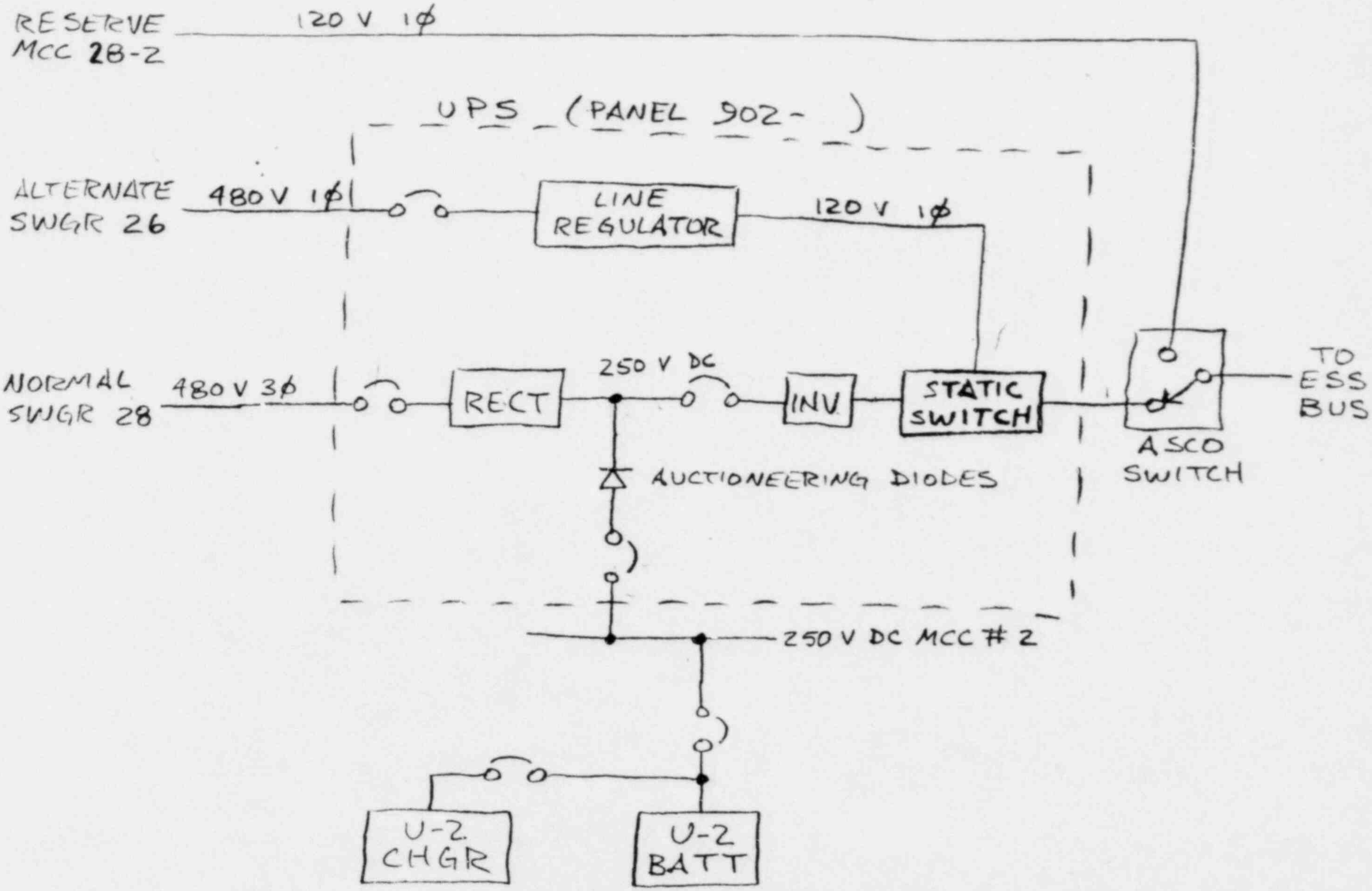
C. No modifications are proposed based on the above review.

2. Emergency operating procedures exist pertaining to loss of power to the aforementioned buses. These procedures include appropriate symptoms, alarms, and operator actions to be taken to restore power to the appropriate bus. Based on the review performed in response to item 1 above, procedure revisions will be made to more clearly give the use of alternate indications and circuits from other buses. These revisions will also increase the scope of symptoms and operator actions, based on the findings and reviews performed persuant to item 1. Procedure revisions will be implemented by April 1, 1980.

3. IE Circular No. 79-02 has been reviewed. SCI inverters do not exist at Quad-Cities Station. However, modifications had been initiated prior to issuance of the Circular to replace the ESS M-G Sets with static inverters, and to install an inverter to power the Station process computer. These modifications had been issued to upgrade the 120 V.A.C. ESS system to improve its reliability, and to add a measure of flexibility to the system.

   After having initially received circulation 79-02, a copy was sent to the Commonwealth Edison Company Station Nuclear Engineering Department, who was in the process of designing and procuring the inverters in accordance with the modifications. At present, the design and procurement of the new inverter systems is being finalized. Those review items and recommendations given in Circular 79-02 will be re-examined and implemented as necessary prior to completion of the modifications and placing the inverters into service.

UNIT 2 ESSENTIAL SERVICE UNINTERRUPTIBLE POWER SUPPLY
BLOCK DIAGRAM



FK 9/29/81

Unit 2 Essential Service System Description

The Unit 2 Essential Service System (ESS) bus is supplied by an Uninterruptable Power Supply (UPS). The UPS consists of an inverter, rectifier, static switch and AC line regulator.

The Normal supply for the UPS is 480V 30 power from Bus 28. The rectifier converts it to 250 V DC for the inverter. The inverter generates 120 V AC for the ESS bus.

If the Normal AC source fails, auctioneering diodes in the rectifier instantaneously change the supply for the inverter to the Unit 2 250V station battery. When Normal AC is restored to the rectifier, the diodes change the inverter supply back to the rectifier.

If the inverter fails, the static switch changes the ESS bus to the Alternate AC source, Bus 26, through the AC line regulator. The regulator converts 480 V from Bus 26 to 120 V AC for the ESS bus. The static switch will transfer to the Alternate source in about 5 milliseconds. On restoration of power from the inverter, the static switch will transfer back after a delay of one second.

If both Normal and Alternate sources fail, or if the entire UPS fails, the Asco switch will transfer the ESS bus to the Reserve source, MCC 28-2. When the Normal or Alternate source is restored, the Asco switch must be reset manually.

The UPS has a Manual Bypass switch to override the static switch for testing and shutdown of the UPS.