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'82 SEP 20 A10:48

OFFICE OF SECRETARY
DOCKETING & SERVICE
BRANCH

September 13, 1982

Mrs Mary P Sinclair
5711 Summerset Street
Midland, MI 48640

MIDLAND PROJECT
MIDLAND DOCKET NOS 50-329, 50-330
RESPONSE TO INTERROGATORIES

Dear Mrs Sinclair

Enclosed are Consumer Power Company's responses to the "Discovery Questions For Consumers Power Company On New Contentions Accepted by Board Order, August 14, 1982."

In response to these interrogatories, Consumers Power has made the following interpretations:

1. Interrogatory II 5 has been interpreted as referring to the NRC Staff.

2. Interrogatory II 13 includes a reference to a statement attributed to a Dr Epstein. We do not know whether or not this quote is accurate. In any event, such a statement is purely argumentative and totally inappropriate as a part of a discovery request. Notwithstanding, since the statement was made it should be noted that whatever statement was made by Dr Epstein should be considered in light of the transcript of the construction permit hearing, at pp 8314-8348, 3660-3661.

3. Interrogatory II 1 is a request for production of documents under 10CFRS2.741, rather than an interrogatory. Consumers Power Company will therefore respond to the request within the 30-day period prescribed by 10CFRS2.741(d).

Very truly yours

Philip P Steptoe

Philip P Steptoe

oc0982-0233a100

8209210276

DS03

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

ATOMIC SAFETY AND LICENSING BOARD

In the Matter of CCNSUMERS POWER COMPANY (Midland Plant, Units 1 and 2)	Docket No 50-329 OM 50-330 OM Docket No 50-329 OL 50-330 OL

September 10, 1982

AFFIDAVIT OF LOUIS S GIBSON

My name is Louis S Gibson. I am the Section Head, Safety and Analyses Section, of the Midland Project Licensing Department. In this capacity, my responsibilities are to review or conduct certain plant analyses for the Midland Plant.

I am primarily responsible for providing a response to Interrogatory I, Questions 1, 2 and 3 concerning Mary Sinclair Contention 3. To the best of my knowledge and belief, the above information and the responses to the above interrogatory(ies) are true and correct.

L S Gibson

Sworn and Subscribed Before Me This 10th Day of Sept 1982

Pamela J. Griffin
Notary Public
Jackson County, Michigan

My Commission Expires Sept 8, 1984

mi0782-0162n100

Mary P Sinclair

Interrogatory I

Contention 3 questions the adequacy of the methodology in the DES for determining the possibility of severe accidents at the Midland nuclear plants, and recommends NUREG/CR/2497, as a better basis.

Questions

1. Have any accidents occurred at Palisades and Big Rock that were a part of the data base for NUREG/CR/2497, June, 1982?

2. If so, describe them.

3. If so, explain if they were initiated by:

a. operator error

b. non-safety equipment impacting on safety equipment

c. equipment malfunction

d. not believing readings of non-safety grade instruments

e. instruments giving the wrong reading

f. maintenance during operation that disabled the safety systems

g. minor mishaps

h. failure of safety systems

i. lack of QA control during operation

Responses

1. Nine events have occurred at Palisades and Big Rock that were a part of the NUREG/CR 2497 data base.

2. The events mentioned above are described in the attached Licensee Event Reports "LERs" and also the attached relevent pages from NUREG/CR 2497.

3. The authors of NUREG/CR 2497 have described how the events were initiated under the heading "cause" and in the "failure sequence description" in the attached pages from NUREG/CR 2497.

CP Co's own characterizations of the initiating causes of these events are found in the attached LERs. CP Co has not performed any study attempting to recharacterize these events in terms of the categories shown in Discovery Question 3. In addition, CP Co believes it would be potentially misleading to use the terms suggested by Ms Sinclair.

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 132958

DATE OF LER: December 21, 1977

DATE OF EVENT: December 11, 1977

SYSTEM INVOLVED: off-site power

COMPONENT INVOLVED: switchyard bus "R"

CAUSE: spurious stripping relay signal

SEQUENCE OF INTEREST: loss of offsite power

ACTUAL OCCURRENCE: reactor trip with loss of offsite power

REACTOR NAME: Palisades

DOCKET NUMBER: 50-255

REACTOR TYPE: PWR

DESIGN ELECTRICAL RATING: 805 MWe

REACTOR AGE: 6.8 yr

VENDOR: Combustion Engineering

ARCHITECT-ENGINEERS: Bechtel

OPERATORS: Consumers Power Co.

LOCATION: 5 miles south of South Haven, Mich.

DURATION: N/A

PLANT OPERATING CONDITION: 100% power

SAFETY FEATURE TYPE OF FAILURE: (a) inadequate performance; (b) failed to start; (c) made inoperable; (d) ________________

DISCOVERY METHOD: during operation

COMMENT: See also 132943

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 132958

Date: December 11, 1977

Title: Complete Loss of Offsite Power Occurs at Palisades

The failure sequence was:

1. With the reactor at 100% power, the "R" switchyard bus de-energized because of a spurious signal from the bus stripping relay, resulting in a complete loss of offsite power and consequent loss of main condenser cooling.
2. The reactor was manually tripped.
3. Both diesel generators started and provided power to safety-related equipment.

Corrective action:

1. The "R" bus tripping scheme was modified to minimize loss of the bus due to spurious action of the "R" bus stripping relay. The specific cause of the stripping relay trip had not been determined.

Design purpose of failed system or component:

1. Off-site power provides the preferred source of electric power to plant equipment when the unit generator is not in operation. The condenser circulating water pumps are normally powered from the off-site power source.

Unavailability of system per WASH 1400:* loss of offsite power: 2×10^{-5}/hr

Unavailability of component per WASH 1400:* —

*Unavailabilities are in units of per demand D^{-1}. Failure rates are in units of per hour HR^{-1}.

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 132943

DATE OF LER: December 16, 1977

DATE OF EVENT: November 25, 1977

SYSTEM INVOLVED: offsite power

COMPONENT INVOLVED: switchyard bus "R"

CAUSE: bus trip from unknown causes

SEQUENCE OF INTEREST: reactor trip with loss of offsite power

ACTUAL OCCURRENCE: reactor trip with loss of offsite power

REACTOR NAME: Palisades

DOCKET NUMBER: 50-255

REACTOR TYPE: PWR

DESIGN ELECTRICAL RATING: 805 MWe

REACTOR AGE: 6.8 yr

VENDOR: Combustion Engineering

ARCHITECT-ENGINEERS: Bechtel

OPERATORS: Consumers Power Co.

LOCATION: 5 miles south of South Haven, Mich.

DURATION: N/A

PLANT OPERATING CONDITION: 85% power

SAFETY FEATURE TYPE OF FAILURE: (a) inadequate performance; (b) failed to start; (c) made inoperable; (d) ___________________

DISCOVERY METHOD: during operation

COMMENT: See also 132958

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 132943

Date: December 16, 1977

Title: Complete Loss of Offsite Power Occurs at Palisades

The failure sequence was:

1. During normal operation with the reactor at 85% power, switchyard bus "R" became de-energized, causing a complete loss of offsite power and resulting in a loss of main condenser cooling water flow.
2. The reactor was manually tripped.
3. Both diesel generators started and provided power to safety-related loads.

Corrective action:

None; the cause of the "R" bus loss was under investigation.

Design purpose of failed system or component:

Offsite power provides the preferred source of electric power to plant equipment when the unit generator is not in operation. The condenser circulating water pumps are normally powered from the off-site source.

Unavailability of system per WASH 1400:* loss of offsite power: 2×10^{-5}/hr

Unavailability of component per WASH 1400:* —

*Unavailabilities are in units of per demand D^{-1}. Failure rates are in units of per hour HR^{-1}.

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 130119

DATE OF LER: October 18, 1977

DATE OF EVENT: September 24, 1977

SYSTEM INVOLVED: offsite power

COMPONENT INVOLVED: switchyard "R" bus

CAUSE: bus trip from unknown causes

SEQUENCE OF INTEREST: loss of offsite power

ACTUAL OCCURRENCE: reactor trip due to loss of offsite power

REACTOR NAME: Palisades

DOCKET NUMBER: 50-255

REACTOR TYPE: PWR

DESIGN ELECTRICAL RATING: 805 MWe

REACTOR AGE: 6.5 yr

VENDOR: Combustion Engineering

ARCHITECT-ENGINEERS: Bechtel

OPERATORS: Consumers Power Co.

LOCATION: 5 miles south of South Haven, Mich.

DURATION: N/A

PLANT OPERATING CONDITION: 100% power

SAFETY FEATURE TYPE OF FAILURE: (a) inadequate performance; (b) failed to start; (c) made inoperable; (d) ____________________

DISCOVERY METHOD: during operation

COMMENT: —

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 130119

Date: September 24, 1977

Title: A Complete Loss of Offsite Power Occurs at Palisades

The failure sequence was:

1. With the reactor at 100% power during an electrical storm, the switchyard "R" bus de-energized and caused a complete loss of offsite power and loss of main condenser cooling.
2. The turbine tripped on high condenser vacuum, and effected reactor and generator trips.
3. Both diesel generators started and supplied electric power to safety-related loads.

Corrective action:

none

Design purpose of failed system or component:

Off-site power provides the preferred source of electric power to plant equipment when the unit generator is not in operation. The condenser circulating water pumps are normally powered from the offsite power source.

Unavailability of system per WASH 1400:* loss of offsite power: 2×10^{-5}/hr

Unavailability of component per WASH 1400:* —

*Unavailabilities are in units of per demand D^{-1}. Failure rates are in units of per hour HR^{-1}.

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 97578

DATE OF LER: November 15, 1974

DATE OF EVENT: October 17, 1974

SYSTEM INVOLVED: Offsite Power

COMPONENT INVOLVED: Differential Relays

CAUSE: Improperly designed Differential Relay System

SEQUENCE OF INTEREST: Loss of offsite power

ACTUAL OCCURRENCE: Loss of offsite power during SIS testing

REACTOR NAME: Palisades

DOCKET NUMBER: 50-255

REACTOR TYPE: PWR

DESIGN ELECTRICAL RATING: 805 MWe

REACTOR AGE: 3.4 yr

VENDOR: Combustion Engineering

ARCHITECT-ENGINEERS: Bechtel

OPERATORS: Consumers Power Company

LOCATION: 5 miles south of South Haven, Mich.

DURATION: N/A

PLANT OPERATING CONDITION: hot standby

SAFETY FEATURE TYPE OF FAILURE: (a) inadequate performance; (b) failed to start; (c) made inoperable; (d) ______

DISCOVERY METHOD: testing

COMMENT: This event is the second of its type at Palisades. After the first event (NSIC 71694, 5/17/72), the differential relays were removed pending a design review. They were reinstalled in January 1974 after modification.

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 97578

Date: November 15, 1974

Title: Loss of Offsite Power During SIS Testing at Palisades

The failure sequence was:

1. With the plant in a hot standby condition, the right channel safety injection test button was pushed to initiate a quarterly test.
2. Offsite power was lost due to the inadvertent operation of the differential relay system.
3. The diesel generators started and powered safety-related loads.

Corrective action:

The three-phase differential relays were removed from service pending a review and potential redesign of the system. Over-current protection devices remained installed to provide transformer protection.

Design purpose of failed system or component:

The differential relays provided overcurrent protection for the startup transformers.

Unavailability of system per WASH 1400:* Offsite power: 2×10^{-5}/hr

Unavailability of component per WASH 1400:* -

*Unavailabilities are in units of per demand D^{-1}. Failure rates are in units of per hour HR^{-1}.

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 71694

DATE OF LER: May 26, 1972

DATE OF EVENT: May 17, 1972

SYSTEM INVOLVED: Offsite power

COMPONENT INVOLVED: Startup transformer

CAUSE: Improperly chosen differential relay

SEQUENCE OF INTEREST: Loss of offsite power

ACTUAL OCCURRENCE: Loss of offsite power during testing

REACTOR NAME: Palisades

DOCKET NUMBER: 255

REACTOR TYPE: PWR

DESIGN ELECTRICAL RATING: 805 MWe

REACTOR AGE: 1.0 yr

VENDOR: CE

ARCHITECT-ENGINEERS: Bechtel

OPERATORS: Consumers Power Corp.

LOCATION: 5 miles south of South Haven, Michigan

DURATION: N/A

PLANT OPERATING CONDITION: Hot standby

SAFETY FEATURE TYPE OF FAILURE: (a) inadequate performance; (b) failed to start; (c) made inoperable; (d) ____________

DISCOVERY METHOD: During testing

COMMENT: The incompatible current transformer had been installed prior to criticality

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 71694

Date: May 26, 1972

Title: Loss of Offsite Power at Palisades

The failure sequence was:

1. With the plant in a hot standby condition, the left channel safety injection system test button was pushed to initiate a quarterly test.
2. This resulted in a loss of offsite power due to the spurious operation of a differential relay on the 1-2 startup transformer. The actuation of the relay was due to unbalanced sensing currents from a current transformer — a result of the incompatibility of the installed current transformer with the 345 KV - 2.4 KV step-down situation.

Corrective action:

The differential relays were removed from startup transformers 1-1 and 1-2. High-side overcurrent and instantaneous overcurrent relays were installed.

Design purpose of failed system or component:

The differential relays provided overcurrent protection for the startup transformer.

Unavailability of system per WASH 1400:* Offsite power: 2×10^{-5}/hr

Unavailability of component per WASH 1400:* —

*Unavailabilities are in units of per demand D^{-1}. Failure rates are in units of per hour HR^{-1}.

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 65969

DATE OF LER: September 16, 1971

DATE OF EVENT: September 8, 1971

SYSTEM INVOLVED: Reactor Coolant System, Reactor Protective System

COMPONENT INVOLVED: Electromatic Relief Valves

CAUSE: Opening of reactor protective system supply breakers resulted in opening of electromatic relief valves

SEQUENCE OF INTEREST: small LOCA (open electromatic relief valve)

ACTUAL OCCURRENCE: open electromatic relief valve

REACTOR NAME: Palisades

DOCKET NUMBER: 50-255

REACTOR TYPE: PWR

DESIGN ELECTRICAL RATING: 805 MWe

REACTOR AGE: .3 yr

VENDOR: Combustion Engineering

ARCHITECT-ENGINEERS: Bechtel

OPERATORS: Consumers Power Co.

LOCATION: 5 miles south of South Haven, Mich.

DURATION: N/A

PLANT OPERATING CONDITION: hot shutdown

SAFETY FEATURE TYPE OF FAILURE: (a) inadequate performance; (b) failed to start; (c) made inoperable; (d) failed open

DISCOVERY METHOD: During operation

COMMENT: —

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 65969

Date: September 16, 1971

Title: Unclear Wiring Diagrams Result in Depressurization at Palisades

The failure sequence was:

1. With the plant in hot shutdown, a technician de-energized breakers to the reactor protective system. This resulted in loss of power to the electromatic relief valve pilot valve solenoids opening the valves.
2. One of the electromatic relief valves was isolated because its isolation valve was closed; however, the open, unisolated relief valve permitted RCS blowdown to the quench tank.
3. Safety injection was initiated on both safety injection channels, however channel A was blocked by the operator.
4. The operator closed the electromatic relief valve isolation valve and started the third charging pump.

(Continued on attached sheet)

Corrective action:

1. The reactor protective system drawings were to be corrected to indicate the "as-built" plant condition using standard notation.
2. The electromatic relief valve control scheme was to be reviewed to determine if any changes were desirable to lessen the probability of a second incident.

Design purpose of failed system or component:

1. The electromatic relief valves provide RCS overpressure protection at a pressure below the RCS safety valve set points.

Unavailability of system per WASH 1400:* —

Unavailability of component per WASH 1400:* relief valve, failure to reseat: 10^{-2}/D

---

*Unavailabilities are in units of per demand D^{-1}. Failure rates are in units of per hour HR^{-1}.

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 61565

DATE OF LER: September 9, 1971

DATE OF EVENT: September 2, 1971

SYSTEM INVOLVED: offsite power; emergency on-site power

COMPONENT INVOLVED: switchyard breaker relay, diesel generator output relay

CAUSE: failure of switchyard breaker relay, failure to close for diesel generator breaker

SEQUENCE OF INTEREST: loss of offsite power

ACTUAL OCCURRENCE: Loss of offsite power and failure of a diesel generator to load

REACTOR NAME: Palisades

DOCKET NUMBER: 50-255

REACTOR TYPE: PWR

DESIGN ELECTRICAL RATING: 805 MWe

REACTOR AGE: .3 yr

VENDOR: Combustion Engineering

ARCHITECT-ENGINEERS: Bechtel

OPERATORS: Consumers Power Company

LOCATION: 5 miles south of South Haven, Mich.

DURATION: N/A

PLANT OPERATING CONDITION: not known

SAFETY FEATURE TYPE OF FAILURE: (a) inadequate performance; (b) failed to start; (c) made inoperable; (d) failed to load.

DISCOVERY METHOD: during operation

COMMENT: -

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 61565

Date: September 2, 1971

Title: Loss of Offsite Power and Failure of a Diesel Generator to Load at Palisades

The failure sequence was:

1. The Argenta No. 2 345KV line (one of three 345KV lines at the switchyard) tripped.
2. Failure of a "breaker failure relay" associated with the tripped line breaker resulted in the tripping of two other breakers on the ring bus, which caused a loss of offsite power.
3. Diesel generator No. 1 started and loaded its safety-related bus.
4. Diesel generator No. 2 started but its breaker did not close until the operator adjusted the synchroscope in preparation to close the breaker manually.

Corrective action:

Not specified (only LER abstract available).

Design purpose of failed system or component:

Offsite power provides electrical power to safety-related components when the unit generator is inoperable. The diesel generators provide a standby source of electric power for safety-related components when both the unit generator and the offsite power sources are not available.

Unavailability of system per WASH 1400:* Offsite Power: 2×10^{-5}/hr

Unavailability of component per WASH 1400:* Diesel-generator: 3×10^{-2}/D

*Unavailabilities are in units of per demand D^{-1}. Failure rates are in units of per hour HR^{-1}.

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 39024

DATE OF LER: March 3, 1972

DATE OF EVENT: January 25, 1972

SYSTEM INVOLVED: electric power

COMPONENT INVOLVED: relays and circuit breakers

CAUSE: line faults induced by a violent storm

SEQUENCE OF INTEREST: loss of offsite power

ACTUAL OCCURRENCE: Loss of offsite power

REACTOR NAME: Big Rock Point

DOCKET NUMBER: 50-155

REACTOR TYPE: BWR

DESIGN ELECTRICAL RATING: 72 MWe

REACTOR AGE: 9.3 yr

VENDOR: General Electric

ARCHITECT-ENGINEERS: Bechtel

OPERATORS: Consumers Power Company

LOCATION: Four miles NE of Charlevoix, Mich.

DURATION: N/A

PLANT OPERATING CONDITION: just scrammed

SAFETY FEATURE TYPE OF FAILURE: (a) inadequate performance; (b) failed to start; (c) made inoperable; (d) ______________

DISCOVERY METHOD: operational event

COMMENT: -

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 39024

Date: January 25, 1972

Title: Loss of Off Site Power and Other Failures At Big Rock Point

The failure sequence was:

1. "Galloping Conductors" caused line faults which resulted in the failure of the Gaylord 388 oil circuit breaker (OCB).
2. Other breakers at the substation acted to clear the fault. This, however, left the plant in a no load condition, resulting in a turbine trip on overspeed and subsequently in a reactor trip due to high neutron flux.
3. The 199 OCB was manually opened since the 138 kv line was de-energized intermittently (unspecified reasons) for 20 minutes.
4. The station transferred to the 46 kv back-up line, however, the breaker tripped. A stuck contact of an instantaneous overcurrent relay (1288 OCB) in conjunction with operation of the undervoltage bus fault relay caused the 46 kv line to de-energize. (see attached page)

Corrective action:

1. An inspection of the transmission lines indicated they had received no damage during event.
2. The faulty trip coil (388 OCB) and the faulty overcurrent relay (1288 OCB) were repaired.

Design purpose of failed system or component:

1. Offsite electric power (both normal and backup) provide power to the station when the unit operator is not inservice.
2. Relays and circuit breakers are provided to protect electrical components from excessive and insufficient current and voltage conditions.

Unavailability of system per WASH 1400:* LOOP: 2×10^{-5}/Hr.

Unavailability of component per WASH 1400:* circuit breakers 1×10^{-3}/D
relays 1×10^{-4}/D

*Unavailabilities are in units of per demand D^{-1}. Failure rates are in units of per hour HR^{-1}.

The failure sequence was: (continued)

5. Upon loss of both offsite lines the diesel generator started ar [illegible] ıded the 2-B bus.
6. The 138 kv line was re-energized approximately 20 minutes after the turbine trip. Attempts to close the 199 OCB failed due to false trip signals from the audio relay equipment.
7. The audio relay equipment was overridden and offsite power was restored.

7711-1530

Consumers Power Company

COPY

General Offices: 212 West Michigan Avenue, Jackson, Michigan 49201 • Area Code 517 788-0550

December 9, 1977

Mr James G Keppler
Office of Inspection and Enforcement
Region III
US Nuclear Regulatory Commission
799 Roosevelt Road
Glen Ellyn, IL 60137

DOCKET 50-255 - LICENSE DPR-20 -
PALISADES PLANT - ER-77-55 -
LOSS OF OFF-SITE POWER

On December 7, 1977 G Petitjean of our Palisades Plant staff contacted D Hunter and requested an extension of one week for the submittal of ER-77-55. Region III was notified of this event by TWX on November 28, 1977. The written Licensee Event Report will now be submitted on December 16, 1977.

David P Hoffman (Signed)

David P Hoffman
Assistant Nuclear Licensing Administrator

CC: ASchwencer, USNRC

P77-156C

Consumers Power Company

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General Offices: 212 West Michigan Avenue, Jackson, Michigan 49201 • Area Code 517 788-0550

December 16, 1977

Mr James G Keppler
Office of Inspection and Enforcement
Region III
US Nuclear Regulatory Commission
799 Roosevelt Road
Glen Ellyn, IL 60137

DOCKET 50-255 - LICENSE DPR-20 -
PALISADES PLANT - ER-77-55 - LOSS OF
OFF-SITE POWER AND PLANT TRIP

Attached is a reportable occurrence related to the loss of off-site power and subsequent plant trip at the Palisades Plant.

David A Bixel (Signed)

David A Bixel
Nuclear Licensing Administrator

CC: ASchwencer, USNRC

2515 01514

LICENSEE EVENT REPORT

Palisades

CONTROL BLOCK: [| | | | |] (1–6)

(PLEASE PRINT ALL REQUIRED INFORMATION)

	LICENSEE NAME	LICENSE NUMBER	LICENSE TYPE	EVENT TYPE
01	M A P A L 1	0 0 - 0 0 0 0 0 - 0 0	4 1 1 1 1	0 1

		REPORT TYPE	REPORT SOURCE	DOCKET NUMBER	EVENT DATE	REPORT DATE
01 CON'T	* *	T	L	0 5 0 - 0 2 5 5	1 1 2 5 7 7	1 2 1 6 7 7

EVENT DESCRIPTION

02 During normal steady state operation, the "R" bus became de-energized causing a com-

03 plete loss of off-site power and resulting in a loss of main condenser cooling water.

04 The reactor was manually tripped. The primary plant was stabilized in the hot con-

05 dition and was borated. LCOs of Tech Spec 3.1.1, 3.7.1 were exceeded. Event similar

06 to ER-77-047. (Contd on attached sheet.)

	SYSTEM CODE	CAUSE CODE	COMPONENT CODE	PRIME COMPONENT SUPPLIER	COMPONENT MANUFACTURER	VIOLATION
07	E A	F	Z Z Z Z Z Z	Z	Z 9 9 9	Y

CAUSE DESCRIPTION

08 Cause of "R" bus loss not known and is still under investigation.

09

10

	FACILITY STATUS	% POWER	OTHER STATUS	METHOD OF DISCOVERY	DISCOVERY DESCRIPTION
11	E	0 8 5	N/A	a	N/A

	FORM OF ACTIVITY RELEASED	CONTENT OF RELEASE	AMOUNT OF ACTIVITY	LOCATION OF RELEASE
12	G	H	34.4 (Contd below.)	Secondary water to atmosphere.

PERSONNEL EXPOSURES

	NUMBER	TYPE	DESCRIPTION
13	0 0 0	Z	N/A

PERSONNEL INJURIES

	NUMBER	DESCRIPTION
14	0 0 0	N/A

PROBABLE CONSEQUENCES

15 N/A

LOSS OR DAMAGE TO FACILITY

	TYPE	DESCRIPTION
16	Z	N/A

PUBLICITY

17 N/A

ADDITIONAL FACTORS

18 (Amount of Activity - Contd) microcuries of I-131; 29.2 microcuries of I-133.

19

Attachment to Licensee Event Report 77-055

Event Description (Contd)

During the incident, a primary coolant leak occurred in the letdown line. The leak was isolated. Both diesel generators operated normally to supply electrical power during the incident. Electrical power was restored in 2.3 hours.

The delay in reporting this event was discussed with Region III. See our letter dated December 9, 1977.

(ER-77-55)

Consumers
Power
Company

COPY

General Offices: 212 West Michigan Avenue, Jackson, Michigan 49201 • Area Code 517 788-0550

December 21, 1977

Mr James G Keppler
Office of Inspection and Enforcement
Region III
US Nuclear Regulatory Commission
799 Roosevelt Road
Glen Ellyn, IL 60137

DOCKET 50-255, LICENSE DPR-20 -
PALISADES PLANT - ER-77-054,
ER-77-057 and ER-77-058

Attached are three reportable occurrences for the Palisades Plant. Event Report 77-058 was a prompt reportable event that was identified by TWX dated December 12, 1977.

David P Hoffman (Signed)

David P Hoffman
Assistant Nuclear Licensing Administrator

CC: ASchwencer, USNRC

2515 0517

LICENSEE EVENT REPORT

Palisades

CONTROL BLOCK: | | | | | | | (PLEASE PRINT ALL REQUIRED INFORMATION)

01 LICENSEE NAME: M I P A L 1 | LICENSE NUMBER: 0 0 – 0 0 0 0 0 – 0 0 | LICENSE TYPE: 4 1 1 1 1 | EVENT TYPE: 0 1

01 CON'T | * * | REPORT TYPE: T | REPORT SOURCE: L | DOCKET NUMBER: 0 5 0 – 0 2 5 5 | EVENT DATE: 1 2 1 1 7 7 | REPORT DATE: 1 2 2 1 7 7

EVENT DESCRIPTION

02 During normal steady-state operation, the 'R' bus became de-energized, causing a
03 complete loss of offsite power and resulting in a loss of main condenser cooling
04 water. The reactor was manually tripped. The primary plant was stabilized in the hot
05 condition and was borated. Technical Specifications 3.1.1.a and 3.7.1 were violated
06 Both diesel generators operated normally to supply electrical power during the 1.5

(cont'd)

(ER-77-58)

07 SYSTEM CODE: E A | CAUSE CODE: F | COMPONENT CODE: Z Z Z Z Z Z | PRIME COMPONENT SUPPLIER: Z | COMPONENT MANUFACTURER: Z 9 9 9 | VIOLATION: Y

CAUSE DESCRIPTION

08 The cause of the 'R' bus loss is unknown at this time. To mitigate the consequences
09 of future 'R' bus losses, the tripping scheme for the 'R' bus has been modified so
10 that a signal from the stripping relay (486 s-x) will trip neither the 345 KV supply

(cont'd)

11 FACILITY STATUS: E | % POWER: 1 0 0 | OTHER STATUS: NA | METHOD OF DISCOVERY: a | DISCOVERY DESCRIPTION: NA

12 FORM OF ACTIVITY RELEASED: G | CONTENT OF RELEASE: H | AMOUNT OF ACTIVITY: 15 (cont'd) | LOCATION OF RELEASE: Secondary water to atmosphere

PERSONNEL EXPOSURES

13 NUMBER: 0 0 0 | TYPE: Z | DESCRIPTION: NA

PERSONNEL INJURIES

14 NUMBER: 0 0 0 | DESCRIPTION: NA

PROBABLE CONSEQUENCES

15 NA

LOSS OR DAMAGE TO FACILITY

16 TYPE: Z | DESCRIPTION: NA

PUBLICITY

17 NA

ADDITIONAL FACTORS

18 (Amount of Activity- cont'd) microcuries of I-131; [illegible].5 microcuries of I-133.

19

Event Description, (cont'd):

hours that off-site power was not available. This event similar to ER 77-055.

(ER 77-058)

Cause Description (cont'd):

breaker 27-R8 nor the low side breakers feeding the 2400 and 4160 volt buses. The 27-R8 breaker will trip from the transformer bank differential relays, and thermal trips for the feeder breakers have been retained. This scheme retains electrical fault protection, yet will prevent plant trips from spurious action of the 'R' bus stripping relay.

To determine the source of the 'R' bus stripping signals, the 486 S-X relay has been instrumented so that recorder traces can be obtained in the event future trips occur.

BC72-007.0

Consumers Power Company

COPY

General Offices: 212 West Michigan Avenue, Jackson, Michigan 49201 • Area Code 517 788-0550

March 3, 1972

Dr. Peter A. Morris, Director
Division of Reactor Licensing
United States Atomic Energy Commission
Washington, DC 20545

Re: Docket 50-155
DPR-6 (ZEK)

Dear Dr. Morris:

This is intended to apprise you of the loss of an off-site power incident which occurred at our Big Rock Point Nuclear Plant on January 25, 1972. During the outage caused by the loss of off-site power, a two-to-three-foot drop in the spent fuel pool water level was experienced and one of two d-c operated emergency condenser valve (MO-7063) failed after opening automatically as required by reactor pressure conditions.

The loss of off-site power was attributed to a combination of unusually severe weather conditions and several equipment failures.

On the evening of January 24, 1972, an intense storm system passed through the area which consisted of rain that later changed to heavy snow as the temperatures fell. High winds on the following day caused the ice laden power lines to dance and sway. This resulted in a phenomenon called "galloping conductors" in which line faults occurred as the lines move relative to one another. The location of these line faults was calculated to be approximately 10 miles from the Gaylord Substation and on the 138 kV power line (see attached sketch). The Gaylord 388 oil circuit breaker (OCB) operated 12 times to clear the momentary line faults. On the thirteenth fault, the trip coil burned out and the 388 OCB failed to operate. Relays in substations feeding into and out of the Gaylord Substation caused breaker operations at their respective locations to clear the fault. As a result, the Big Rock Point Plant became momentarily isolated from the rest of the system and with essentially no load on the generator the unit tripped off on overspeed (116 OCB opened). This occurred at approximately 1304. The reactor scrammed due to high flux.

Since the fault occurred on the Gaylord side of the Emmet 488 OCB, a trip signal was not sent to the Big Rock Point 199 OCB and a load rejection did not occur. However, the 199 OCB was opened manually since the 138 kV line was de-energized intermittently over

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a span of approximately 20 minutes. Normally, the station power would have automatically transferred on undervoltage to the 46 kV source to supply the station power equipment. However, a stuck contact of an instantaneous overcurrent relay in the Emmet 46 kV bus protection relay scheme coupled with the operation of the undervoltage (UV) bus fault detector relay (which would have reopened had the fault cleared within a few cycles - fault lasted for 69 cycles) caused the 1288 OCB to trip and thus de-energize the 46 kV line to Big Rock. The two relays are connected in series and both must be closed simultaneously for a few cycles to cause the breaker to trip. The line was de-energized for approximately two hours until repairmen, who were hampered by considerable blowing and drifting of snow, could make the essential repairs and return the system to normal. The overcurrent relay had failed closed prior to the undervoltage condition occurring; it should not have operated under the conditions of the incident.

Upon the loss of both the normal and backup supplies of station power, the diesel generator started and closed onto bus 2-B to provide power for operation of essential emergency equipment.

Approximately 20 minutes after the turbine trip (1324), full potential was restored to the 138 kV line. However, when attempts were made to reclose the 199 OCB, a false tripping signal (from the audio tone relay equipment) caused the breaker to immediately retrip. The audio tone control was then put into the "off" position defeating the tripping signal and the 199 OCB was then reclosed successfully. The tone controls were then reconnected and station power loads were returned to their normal station power supply at 1353. The diesel generator was not shut down until approximately 1507.

Extensive inspection of the line section between the calculated fault location and the Gaylord Substation did not produce any evidence of damage. The faulty trip coil (388 OCB) and the faulty overcurrent relay (1288 OCB) have been repaired. The false tripping signal observed in the tone relaying equipment is being investigated as is the trip scheme for the 199 OCB.

In summary, the simultaneous loss of the normal and backup off-site station power supplies was caused by extremely severe and unusual weather conditions and two equipment failures. The length of time that off-site power was lost was extended by difficulties in getting substation operators to the substations and further compounded by a false tripping signal in tone relaying equipment. It is not considered possible that a plant-initiated event would cause a loss of the normal station power supply because of the size of the plant with respect to the system size. The plant provides up to 71 MWe (net) to Michigan Power Pool system of approximately 10,000 MWe.

The 46 kV station power supply was installed in March 1968 to provide a redundant station power supply to the Big Rock Point Plant. Since that installation, the loss of off-site power experienced January 25, 1972 is the only instance where both off-site power supplies were not

0 2 4 9 6 0 6 9 5

available at the same time. From the review of this occurrence, it is considered highly unlikely that a similar incident will ever occur again.

During the reactor pressure transient resulting from the turbine overspeed trip on January 25, the two emergency condenser outlet valves (MO-7063 and MO-7053) opened automatically as designed. Approximately two and one half hours later, when an attempt was made to shut MO-7063, the valve failed to operate. The emergency condenser inlet valve MO-7062 in the affected loop was shut to maintain reactor pressure. MO-7053 operated normally.

The motor unit of MO-7063 was disassembled and taken to a local motor rewinding shop for inspection and possible repair. The insulation on shunt and series fields was found burned to the extent that it required rewinding. However, there were no grounds or shorts found in either winding which suggested the burnup may have been caused by excessive running of the motor beyond its shutoff limits. Upon reinstallation of the motor, the torque switches were inspected for proper settings. The open torque switch was adjusted to allow for a wider margin between the valve automatic stop and manual stop.

It was concluded that the failure of the valve operator motor was probably due to an improperly set torque switch. A replacement motor for the valve operator has been ordered. When it arrives, it will be installed and the present motor returned to the vendor for a detailed inspection. This motor failed and was rewound once previously. The previous failure occurred while maintenance was being performed on the motor and was due to a maintenance error.

The first indication of an abnormal spent fuel pool water level was at 0100 on January 26, 1972 when the Control Room Operator observed a gradual increase in the fuel pool area monitor readings (approximately 12 mR). A visual inspection followed. The fuel pool water level was discovered to be two to three feet below the normal overflow to surge tank level. Detailed investigations of this decrease in fuel pool water level revealed the cause to be a siphoning action created by the piping configuration and valve alignment.

At the time the outage occurred, the fuel pool was valved for recycle through the radwaste system. However, upon loss of the normal station power, the fuel pit, radwaste and treated waste pumps ceased to operate and the fuel pit to radwaste isolation valves CV-4027 and CV-4117 closed automatically.

The critical elevations in the piping sequence are as follows:

1. Discharge Pipe at Bottom of Pool - 601'-6"

2. Highest Elevation of Pipe at Pool Surface - 630'-6"

3. Elevation of Sphere Isolation Valves - 593'-6"

4. Elevation of Pipe Where It Opens Into Clean Waste Receiver Tanks - 590'

Thus, the hydraulic head was available to create a siphoning action (11'-6" head) from the fuel pool to the clean waste receiver tanks when the isolation valves were reopened. (See Attachments 2 and 3.)

With the loss of two to three feet of water, equipment hanging on the side of the pool such as flange protectors, sample specimen racks and vacuum hoses were exposed to some extent. A fuel bundle was located in the elevator which was at the top of its vertical travel. The elevator has a pneumatic stop at seven feet below normal water surface and a mechanical stop at six feet. The decrease in water shielding allowed an increase in radiation dose rate in the vicinity of the fuel pool.

After the discovery, the valve to radwaste ("T" handle valve in fuel pit pump room) was closed and the fuel pool level restored via the waste hold tanks.

Two to three feet of water represents approximately 7,800 to 11,700 gallons. Since water was being processed to the condensate storage tank and the reactor valved for blowdown, the increasing amount of water to the radwaste system from the fuel pool was not immediately evident. Each of two clean waste and waste hold tanks has a high level alarm set at approximately 90%. The condensate storage tank high level alarm annunciates at about 95%. The storage tank level is also recorded on a chart on the front panel of the control room. Since these tanks are monitored frequently, it is inconceivable that the tank levels would have continued to rise unnoticed. Even assuming that they did, the spent fuel pool area monitor is set to alarm at 15 mR. It was reading approximately 12 mR and rising slowly when low pool level was first detected.

To prevent any further reoccurrence of the problem, a siphon breaker has been added to the inlet piping of the spent fuel pool. Three 7/16" holes were drilled in a horizontal pattern on the vertical section of inlet piping about 2" below the normal fuel pool level. On February 2, a test of the siphon breaker was conducted under conditions similar to those of January 25 and the siphoning action ceased when the pool level reached the three holes.

The addition of the siphon breaker eliminates any potential for losing the spent fuel pool water level. However, assuming that a loss of fuel pool water level were to again occur, the spent fuel pool

0 2 4 9 6 0 6 9 7

area monitor remote indicators and alarms and radwaste system tank level indicators and alarms would provide early notice of the abnormal condition.

Yours very truly,

Ralph B. Sewell (Signed)

RBS/dmb

Ralph B. Sewell
Nuclear Licensing
Administrator

CC: BHGrier,
USAEC

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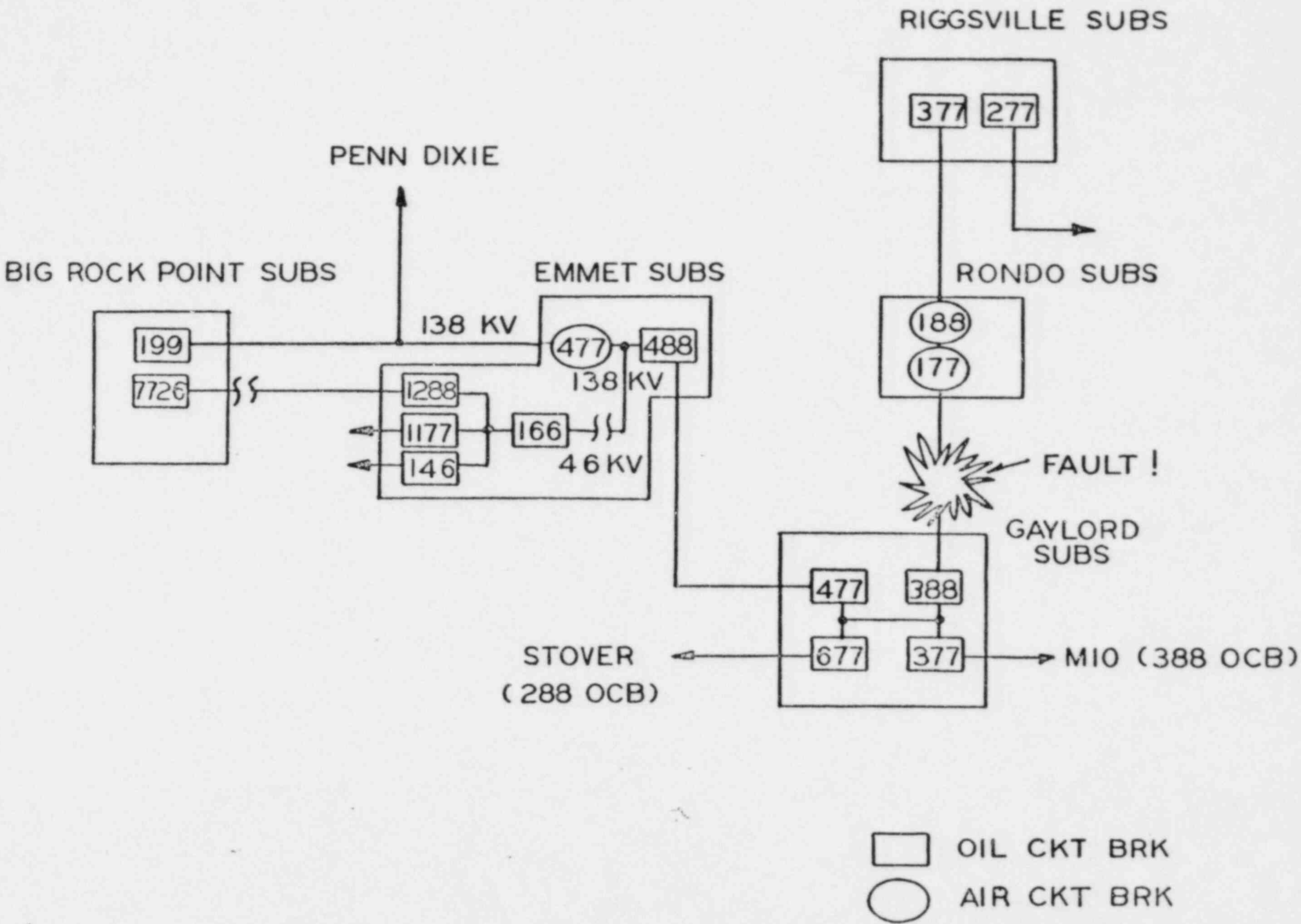
RIGGSVILLE SUBS
377
277
PENN DIXIE
BIG ROCK POINT SUBS
EMMET SUBS
RONDO SUBS
138 KV
199
477
488
188
7726
1288
138 KV
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1177
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FAULT !
GAYLORD SUBS
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STOVER
(288 OCB)
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M10 (388 OCB)
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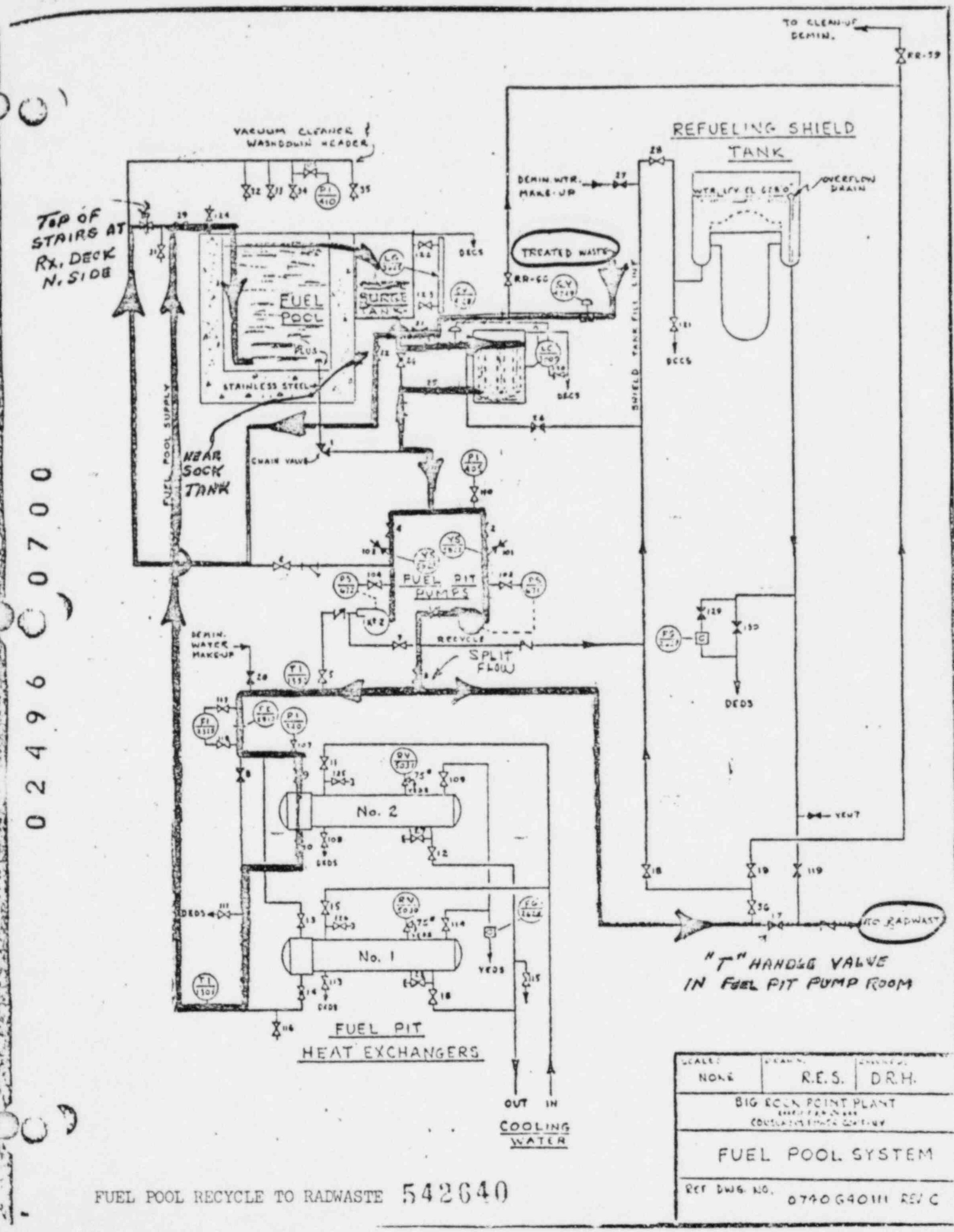
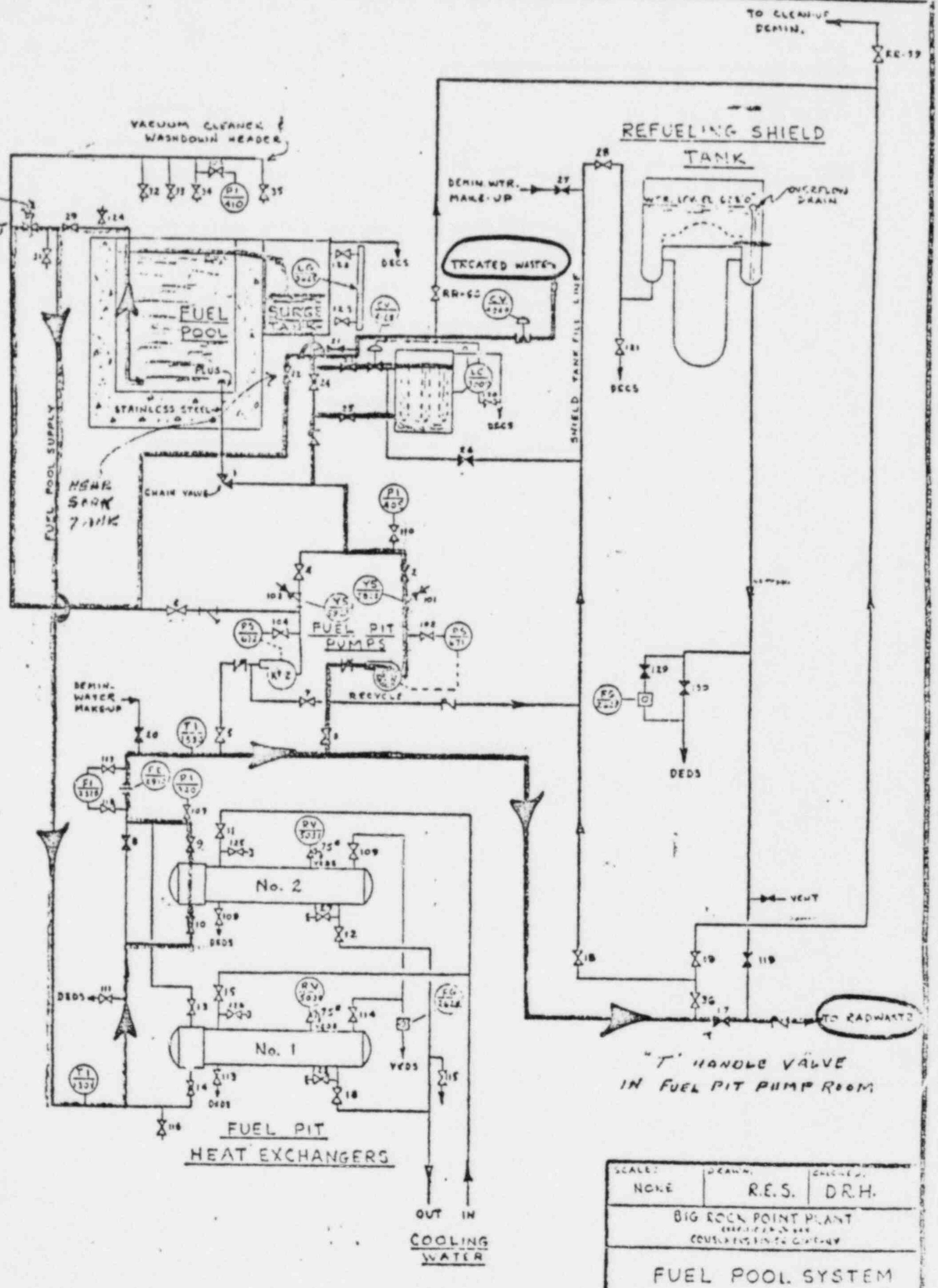


FUEL POOL RECYCLE TO RADWASTE 542640



SCALE:	DRAWN:	CHECKED:
NONE	R.E.S.	D.R.H.

BIG ROCK POINT PLANT
CONSUMERS POWER COMPANY

FUEL POOL SYSTEM

REF. DWG. NO. 0740G40111 REV C

ABNORMAL SYPHONING FLOW TO RADWASTE

542641

02496 0701

P71-016C

Consumers Power Company

COPY

General Offices: 212 West Michigan Avenue, Jackson, Michigan 49201 • Area Code 517 788-0550

September 9, 1971

Dr. Peter A. Morris, Director
Division of Reactor Licensing
US Atomic Energy Commission
Washington, DC 20545

Re: Docket No 50-255
License No DPR-20

Dear Dr. Morris:

This letter is written to apprise you of a recent incident involving the emergency power supply at this facility.

At the time the incident occurred, the plant was in a cold shutdown condition with primary coolant system at refueling boron concentration and atmospheric pressure. The shutdown cooling system was in operation.

The 345 kV Argenta No 2 line tripped at 0706 on September 2, 1971. A breaker failure relay operation on the 27R8 air blast breaker tripped breakers 29H9 and 25R8, clearing the 'R' 345 kV bus and resulting in the loss of power to the start-up transformers. (Refer to attached Drawing E-501.) The main bank has not been routinely energized during the current low power operation resulting in a single source of off site power.

The emergency diesel generators both started and quickly achieved rated speed with the 1-1 unit properly closing in on the dead 2400 V bus and picking up load as designed. (Refer to attached Drawing E-1.) However, the 1-2 diesel generator unit failed to close in on the dead bus automatically until the operator turned the synchroscope plug in preparation for closing the breaker manually.

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CAUSE OF INCIDENT

The cause of the loss of power to the station was determined to be a faulty breaker failure relay on the 27R8 breaker which caused the 'R' bus to be isolated. The exact cause of the diesel generator failure to close automatically on the dead bus is still under investigation at this time.

CORRECTIVE ACTION PLANNED

A review of the wiring scheme and actual installation is in progress to determine the exact mode of failure. Test operation under simulated loss of offsite power will be conducted to determine any deficiency and/or demonstrate the operability of the complete system.

Yours very truly,

Robert L. Haueter

LMH/ERC/mho

Robert L. Haueter
Electric Production
Superintendent - Nuclear

CC: BHGrier
USAEC Compliance, COO

P71-0170

Consumers Power Company

COPY

General Offices: 212 West Michigan Avenue, Jackson, Michigan 49201 • Area Code 517 788-0550

September 16, 1971

Dr. Peter A. Morris, Director
Division of Reactor Licensing
US Atomic Energy Commission
Washington, DC 20545

Re: Docket No 50-255
License No DPR-20

Dear Dr. Morris:

This letter is written to apprise you of a recent incident involving the primary coolant system at this facility.

At the time the incident occurred, the plant was in a hot shutdown condition with the primary coolant system at refueling boron concentration, 532° F and 2100 psia. Three of the four primary coolant pumps were in operation and preparations underway to bring the reactor critical for operator training.

At 1335 on September 8, 1971, a technician de-energized the breakers to the reactor protective system to install a minor modification. This act de-energized the feed to the electromatic relief valve pilot valve solenoids allowing the valves to open.

The primary system pressure decreased to a low point of approximately 1280 psia over a period of 2 - 3 minutes until the blowdown was terminated by closure of the motor operated valves used to isolate the open relief valve.

The system pressure and temperature were back to normal in approximately one hour.

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The rate of depressurization of the reactor vessel and the pressurizer approximated the design rate for a pressurizer safety valve operation. The number of occurrences allowed for in the design was 200. Therefore, we conclude that the integrity of the primary coolant system has not been compromised by this incident.

SEQUENCE OF EVENTS

1335 Breaker #13 on panels Y10 and Y30 de-energized to allow for a minor modification to the reactor protective system. Refer to attached drawing E-8.
Relays K1 and K3 which feed the electrically operated pressurizer relief valve bi-stables (RV1042B and RV1043B) de-energized.

1335 RV1043B opened and released steam to the quench tank starting a decrease of primary system pressure. RV1042B had previously been isolated by means of the motor operated isolation valve and did not pass steam during the incident. Refer to attached drawing M-201.

1336 Safety injection signal (SIS) actuated on both channels, however the A channel was blocked by action of the operator upon realization of the cause of the pressure drop. Safety injection pumps started.

1336 Operator started MOV1043A closing.

1337 Charging pump P55C manually started.

1337 System pressure decay turned around at a low point of 1280 psia as MOV1043A closed stopping flow of steam to the quench tank via RV1043B.

1338 Stopped safety injection pumps.

1343 Breaker #13 on panels Y10 and Y30 closed and power restored to relief valve control circuits.

1344 Returned charging and letdown to normal.

1400 As the system pressure recovered to a point above 1700 psia, the operator reset the safety injection channel B and returned all equipment to normal.

1430 Primary coolant system pressure and temperature back to normal. The coolant temperature and pressurize level swings of 3° F and 4 percent respectively are considered insignificant.

CAUSE OF INCIDENT

The basic cause of this incident was the non-standard designation of contacts (as used by the architect engineer on the drawing) in the control circuit to the power operated relief valves. The technician was misled by the 'a' contact designation as shown on the drawing when in fact the circuit is wired using 'b' contact(s). This led him to believe the relief valves would not open when the power was removed by opening breaker #13 on the panels Y10 and Y30.

CORRECTIVE ACTION PLANNED

The drawing(s) will be corrected to indicate the "as built" condition using standard notations.

A review of the control scheme design will be conducted by Consumers and the Reactor Supplier personnel to determine if any changes are desirable in order to lessen the probability of a second incident.

The plant operators will be furnished with revised procedures indicating steps to be utilized to minimize the blowdown if it should occur in the future.

Yours very truly,

R. L. Haueter

LMH/ERC/mho

Robert L. Haueter
Electric Production
Superintendent - Nuclear

CC: BHGrier
USAEC Compliance, COO

P71-019C

Consumers Power Company

COPY

General Offices: 212 West Michigan Avenue, Jackson, Michigan 49201 • Area Code 517 788-0550

September 29, 1971

Dr. Peter A. Morris, Director
Division of Reactor Licensing
US Atomic Energy Commission
Washington, DC 20545

Re: Docket No 50-255
License No DPR-20
Palisades Plant

Dear Dr. Morris:

This letter will confirm our recent discussion with the USAEC Compliance Office, Chicago, and provide supplemental information for our letter dated September 9, 1971 concerning the incident involving the emergency power supply at Palisades.

An investigation of the incident revealed that all four 345 kV lines into the substation were energized at the time of the incident and that lightning faulted the 345 kV Argenta No 2 Line. The Argenta end of the line and one breaker at Palisades cleared properly. The 27R8 air blast breaker at Palisades failed to trip by primary relaying and after a few cycles the breaker failure relays operated to clear the 27R8 breaker and "R" bus. Relaying for these conditions was correct. This is contrary to our earlier information as reported in our September 9, 1971 letter.

The 27R8 air blast breaker circuitry was tested and the breakers repeatedly tripped in an effort to again initiate the malfunction. This malfunction could not be repeated. If further testing proves futile, certain components of the primary relays on this circuit will be replaced as a precautionary measure to possibly prevent a future occurrence of the above.

A wiring error was found to have caused the failure of the number 1-2 diesel generator to automatically close in on the dead 2400 volt 1D bus after power was lost from the startup transformer. Contrary to the electrical drawings, the permissive contacts to breaker number 152-213 (diesel generator) were wired to

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"movable" type contacts rather than stationary contacts in breaker cell number 152-203 (station power to 1D bus). Thus, continuity was broken to the diesel start circuitry with breaker number 152-203 in the "racked out" position. *

Breaker number 152-203 (station power to 1D bus) was placed in the test position and caution tagged thus, placing the "movable" contacts into the circuit that will allow the diesel generator to close into the 1D bus normally.

The wiring error will be corrected before power operation.

Yours very truly,

R. L. Haueter

Robert L. Haueter
Electric Production
Superintendent - Nuclear

ERC/mho

CC: BHGrier
USAEC Compliance, COO

* At the time of this incident no power was being fed from the main transformer bank via the station power transformers because of plant status and the low power demand for station auxiliaries.

P74-081C

Consumers Power Company

COPY

General Offices: 212 West Michigan Avenue, Jackson, Michigan 49201 • Area Code 517 788-0550

November 15, 1974

Mr. James G. Keppler, Director
Region III, Directorate
of Regulatory Operations
US Atomic Energy Commission
799 Roosevelt Road
Glen Ellyn, Illinois 60137

Re: Docket 50-255
License DPR-20
Palisades Plant
UE-2-74

Dear Mr. Keppler:

Attached is an Unusual Event Report (UE-2-74) covering a problem with the start-up transformer differential current protection relays. This unusual event is similar to the one reported on May 26, 1972. A preliminary investigation of the present problem has indicated that our recent revision of the differential relay arrangement has not completely resolved the original problem. The investigation is continuing and we are confident that an adequate revision to the differential relaying system can be made and spurious operation eliminated.

Adequate protection of the start-up transformer is being provided by the overcurrent protective device.

Yours very truly,

Ralph B. Sewell (Signed)

DAB/map

Ralph B. Sewell
Nuclear Licensing Administrator

CC: Directorate
of Licensing
USAEC
Washington, DC

2514 0347

7912150022

UNUSUAL EVENT REPORT
Palisades Plant

1. Unusual Event: UE-2-74

2A. Report Date: November 15, 1974

2B. Event Date: October 17, 1974

3. Facility: Palisades Plant, Covert, Michigan

4. Identification of Occurrence: Start-Up Transformer Trip During SIS Test

5. Condition Prior to Occurrence: The plant was at hot standby.

6. Description of Event: The quarterly safety injection system (SIS) test was in progress with the left channel part of the test having been successfully completed. When the right channel test was initiated, off-site power was lost. The diesel generators started and automatically closed to provide plant power. Off-site power was restored in about one half hour.

7. Designation of Apparent Cause of Occurrence: The installation of the current transformers associated with the differential relays appears to be the cause of the problem. This is complicated because of incompatability problems associated with 345 kV to 2.4 kV step-down situation.

8. Analysis of Occurrence: Operation of the SIS would have been dependent on the plant emergency diesel generators. The SIS is designed to operate off of the diesel generators and would have performed properly.

9. Corrective Action: The three-phase differential relays have been removed from service pending an investigation by our Relay Protection Department. Overcurrent protection devices remain installed and provide adequate transformer protection. Even so, because of the added transformer protection available utilizing a differential relay system, we plan to reinstall the differential relay system when a suitable design can be achieved.

The Safety Audit and Review Board will conduct a review of the differential relay system and any appropriate testing program prior to plant operation (at power) with differential relays in service on the start-up transformer.

The SIS test has been successfully completed with the differential relays out of service.

10. Failure Data: This same failure occurred in May 1972 under similar conditions. At that time, the differential relay system was removed from service and was not reinstalled until after it was modified in January 1974.

Consumers Power Company

COPY

General Offices: 212 West Michigan Avenue, Jackson, Michigan 49201 • Area Code 517 788-0550

October 18, 1977

Mr James G Keppler
Office of Inspection and Enforcement
Region III
US Nuclear Regulatory Commission
799 Roosevelt Road
Glen Ellyn, IL 60137

DOCKET 50-255 - LICENSE DPR-20 -
PALISADES PLANT - ER-77-047 -
LOSS OF OFF-SITE POWER

Attached is a 14-day reportable occurrence relating to the loss of off-site power at the Palisades Plant on September 24, 1977. The report due date was extended one week as previously discussed with Mr R Warnick.

An attachment to the LER is provided for additional details.

David P Hoffman (Signed)

David P Hoffman
Assistant Nuclear Licensing Administrator

CC: ASchwencer, USNRC

ATTACHMENT TO LER 77-047

The cause of the 'R' Bus loss is not known. Loss of off-site power causes a loss of main condenser cooling water. Thus, the main turbine tripped on high back pressure and tripping of the main generator and reactor occurred. Both emergency diesel generators started immediately, loaded properly and performed satisfactorily throughout the incident. The secondary system was isolated, primary system stabilized and the plant functioned as designed during the incident. Atmospheric dumps were operated as necessary to maintain PCS temperature. During the incident, power was lost to the Security System and extra Security personnel were called into the plant. Technical Specifications 3.1.1 and 3.7.1 were violated.

3.1.1 At least one primary coolant pump or shutdown cooling pump shall be in operation whenever a change is being made in the boron concentration of the primary coolant.

It is conservative and prudent to borate the plant to shutdown condition after a reactor trip. Boron samples during the incident and after restoring primary coolant flow were as anticipated and verified that no stratification occurred.

3.7.1 The primary coolant system shall not be heated above $325^{\circ}F$ or maintained above $325^{\circ}F$ if the following electrical systems are not operable.

a. Station power transformer 1-2.

b. Start-up transformer 1-2.

l. 2400 'V' Bus 1-E.

Throughout the incident the plant was maintained in hot shutdown. Power was restored after 4 hours and 45 minutes. It was prudent to maintain the PCS hot rather than add to the Operators workload as power was restored shortly, and plant conditions were stabilized. This is permitted by C.E. Standard Technical Specifications. Changes to 3.1.1 and 3.7.1 will be submitted for approval.

The plant operated as analyzed and no failures of safety-related systems occurred as a result of this incident.

LICENSEE EVENT REPORT

Palisades

CONTROL BLOCK: | | | | | | | (1–6)

(PLEASE PRINT ALL REQUIRED INFORMATION)

01	LICENSEE NAME	LICENSE NUMBER	LICENSE TYPE	EVENT TYPE
7 8	M I P A L 1 (9–14)	0 0 – 0 0 0 0 0 – 0 0 (15–25)	4 1 1 1 1 (26–30)	0 1 (31–32)

01 CON'T	REPORT TYPE	REPORT SOURCE	DOCKET NUMBER	EVENT DATE	REPORT DATE
* * (57–58)	T (59)	L (60)	0 5 0 – 0 2 5 5 (61–68)	0 9 2 4 7 7 (69–74)	1 0 1 8 7 7 (75–80)

EVENT DESCRIPTION

02 During an electrical storm, the 'R' Bus was de-energized causing a complete loss

03 of off-site power, resulting in a loss of main condenser cooling water and ultimately

04 a plant trip. Primary plant was stabilized in the hot condition and was borated.

05 Event nonrepetitive. Tech Specs 3.1.1 and 3.7.1 were violated. Electrical power

06 was restored after 4.75 hours and the plant returned to Tech Spec limits. (ER-77-47)

07	SYSTEM CODE	CAUSE CODE	COMPONENT CODE	PRIME COMPONENT SUPPLIER	COMPONENT MANUFACTURER	VIOLATION
7 8	E A (9–10)	C (11)	Z Z Z Z Z Z (12–17)	Z (43)	Z 9 9 9 (44–47)	Y (48)

CAUSE DESCRIPTION

08 Exact cause of loss of electrical power is not known. A change to the Tech Specs

09 will be made to make this type of event a nonviolation.

10

11	FACILITY STATUS	% POWER	OTHER STATUS	METHOD OF DISCOVERY	DISCOVERY DESCRIPTION
7 8	E (9)	1 0 0 (10–12)	N/A (13–44)	A (45)	N/A (46–80)

12	FORM OF ACTIVITY RELEASED	CONTENT OF RELEASE	AMOUNT OF ACTIVITY	LOCATION OF RELEASE
7 8	Z (9)	Z (10)	N/A (11–44)	N/A (45–80)

PERSONNEL EXPOSURES

13	NUMBER	TYPE	DESCRIPTION
7 8	0 0 0 (9–11)	Z (12)	N/A (13–80)

PERSONNEL INJURIES

14	NUMBER	DESCRIPTION
7 8	0 0 0 (9–11)	N/A (12–80)

PROBABLE CONSEQUENCES

15 N/A

LOSS OR DAMAGE TO FACILITY

16	TYPE	DESCRIPTION
7 8	Z (9)	N/A (10–80)

PUBLICITY

17 N/A

ADDITIONAL FACTORS

18 N/A

19

P72-030C

Consumers Power Company

COPY

General Offices: 212 West Michigan Avenue, Jackson, Michigan 49201 • Area Code 517 788-0550

May 26, 1972

Mr. E. J. Bloch, Director
Directorate of Licensing
United States Atomic Energy
Commission
Washington, DC 20545

Re: Docket 50-255
License No DPR-20

40-72-19
20

Dear Mr. Bloch:

This is to apprise you of two incidents which occurred recently at the Palisades Plant. The first occurred at 1808 on May 17, 1972 and involved the 1-2 start-up transformer protective relays and the second occurred at 0120 on May 18, 1972 and involved emergency diesel generator No 1-1.

At the time both of these incidents occurred, the plant was in a hot standby condition with the primary system at 530°F, 2100 psia and the coolant boron content at refueling concentration.

At 1808 on May 17, 1972, the safety injection system (SIS) test button was pushed to initiate a quarterly test on the left channel of the safety injection system. This resulted in a loss of outside power as a differential relay on the 2400-volt start-up transformer operated and cleared the 345 kV "R" bus in the switchyard.

The diesel generator started automatically but required manual synchronization to the 2400-volt safeguard buses 1C and 1D. This was as designed and due to the test button depression during the auto cycle.

Preparations were made to back feed off-site power through the normal station power transformers while the tripping of the differential relay was being investigated.

The actuation of the differential relay was spurious and was due to unbalanced sensing currents from the current transformer with load on the start-up transformer. This unbalance was due to the incompatibility of the installed current transformer to the 345 kV to 2.4 kV step-down situation. This unbalance plus the starting currents of the pump motors initiated by the test signal caused the relay to operate.

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At the time the relay operated, the load on start-up transformer No 1-2 was 4.4 MVA plus the starting current of 1750 horsepower of the motors. From test data and the above load data, the 345 kV current transformers require approximately .8 ampere of excitation current to support the voltage requirement of the current transformer. With .8 ampere excitation current, the current transformer had 32% error. The 32% error was enough to cause the differential relay to operate.

To provide protection for start-up transformer No 1-2, the differential relays have been removed and high side overcurrent and instantaneous overcurrent relays have been installed. The high side overcurrent relays will not trip incorrectly for any current transformer saturation that occurs.

The protection for start-up transformer No 1-1 was reviewed and the same saturation problem was found to exist for its differential relay. The same changes were made for the Nc 1-1 transformer as for the No 1-2 transformer.

The protection for station power transformer No 1-1 and No 1-2 has been reviewed. Current transformer saturation is not considered to be a problem in the application of differential protection to these banks, because of the different voltage ratio of these transformer banks.

The load that existed on the No 1-2 start-up transformer at the time of trip was the maximum expected during plant operation. The SIS test has been conducted successfully several times before but at the time they were conducted, loads on the transformer were less.

Prior to returning the plant to service, a safety injection system test will be performed which involves simultaneous actuation of both halves of the safety injection system with the plant in a hot standby condition, thereby subjecting the starting power source to the maximum loads they will see.

At 0120 on May 18, 1972, the diesel generator No 1-1 tripped off due to loss of fuel supply. The engine had operated for over seven hours but ran out of fuel as the level switch on the engine fuel reservoir failed and prevented the flow of fuel from the day tank.

The level switch which failed and prevented continued operation of the diesel generator No 1-1 has been replaced. In addition, the monthly operational tests on the diesel generator unit have been revised to include a functional check of the level switches which actuates the fill valve and the low-level alarm. This is accomplished by shutting off the manual valve and allowing the reservoir level to drop until the switches are actuated.

Yours very truly,

Ralph B. Sewell (Signed)

RBS/dmb

Ralph B. Sewell
Nuclear Licensing
Administrator

CC: BHGrier,
USAEC

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UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

ATOMIC SAFETY AND LICENSING BOARD

In the Matter of CONSUMERS POWER COMPANY (Midland Plant, Units 1 and 2)	Docket No 50-329 OM 50-330 OM Docket No 50-329 OL 50-330 OL

September 10, 1982

AFFIDAVIT OF DAVID A SOMMERS

My name is David A Sommers. I am a Section Head in the Midland Safety and Licensing Department. In this capacity, my responsibilities are supervising and coordinating the review of environmental licensing and radiological safety issues for the Midland Project.

I am primarily responsible for providing a response(s) to Interrogatory II, Questions 5, 6, 7, 8, 9, 10, 11 and 13 concerning Mary Sinclair Contention 5. To the best of my knowledge and belief, the above information and the responses to the above interrogatory(ies) are true and correct.

David A Sommers

Sworn and Subscribed Before Me This 15th Day of Sept 1982

Pamela J. Griffin
Notary Public
Jackson County, Michigan

My Commission Expires Sept 8, 1984

mi0982-2670a168

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

ATOMIC SAFETY AND LICENSING BOARD

In the Matter of CONSUMERS POWER COMPANY (Midland Plant, Units 1 and 2)	Docket No 50-329 OM 50-330 OM Docket No 50-329 OL 50-330 OL

July 12, 1982

AFFIDAVIT OF DONALD H EVANS

My name is Donald H Evans. I am an Engineering Supervisor for Bechtel Associates Professional Corporation. In this capacity, my responsiblities are for Geotechnical support to the Ann Arbor Power Division for Hydraulic, Hydrologic and Hydrothermal Analyses.

I am primarily responsible for providing a response to Interrogatory Questions 2, 3, 4 and 12 concerning Mary Sinclair Contention 5. To the best of my knowledge and belief, the above information and the responses to the above interrogatory are true and correct.

Donald H Evans

Sworn and Subscribed Before Me This 10th Day of Sept 1982

Pamela J. Griffin
Notary Public
Jackson County, Michigan

My Commission Expires Sept 8, 1984

mi0782-2268a100

Mary P Sinclair

Interrogatory II

Contention 5 deals with questions about the adequacy of the basis of the data in the Monthly Cooling Pond Performance Tables on the cooling pond provided in the DES (4-7, 4-8).

Questions

1. Provide all internal correspondence on the cooling pond between Consumers Power Co., Bechtel and the NRC.

2. What data base did you rely on to arrive at 880 acres as an adequate size for a cooling pond for the Midland plants?

3. Who supplied this data base?

4. What experience in cooling pond operation are you relying on for your assurance that this pond is workable and adequate to its tasks?

5. What qualifications did the staff have that you relied on for determining the adequacy of this pond? Provide information on their backgrounds in terms of not only their education, but also their experience for this evaluation.

6. Have you sent any observers to the Dresden pond which was identified by the NRC staff as most closely identical in performance to the Midland plants because of climate and meteorological conditions?

7. If so, when did they make their observations?

8. If so, please provide their report.

9. What are the differences in size between the Dresden pond and the Midland plant?

10. Are their one or two reactors at Dresden?

11. What will be the differences in heat load during full operation between the Midland nuclear plants and Dresden?

12. Explain the source of data for each factor in the Monthly Cooling Pond Performance Tables.

13. What studies have been made to determine the effect of the fogging on the people in the area and the Bullock Creek elementary school? (Dr. Edward Epstein, the meteorologist from the University of Michigan who was our expert witness on fogging, discussed this at a seminar of the nuclear engineering department in October, 1972, and said, "I don't know how those people are going to live.")

Responses

2. Cooling pond sizes are set by the allowable condenser inlet temperature, heat rejected by the plant, and natural heat energy entering and leaving the pond considering the varying meteorological conditions at the pond location. Analytical and physical model studies were performed for the Midland Plant Units 1 and 2 to support the adequacy of the cooling pond size. Listed below are the more significant studies that give the data base to arrive at 880 acres as an adequate size for the Midland cooling pond. These reports contain various data bases and assumptions used for evaluating cooling pond performance.

a. Cooling System Study (May 1968)

The cooling pond alternative was recommended by Bechtel against other types of plant cooling systems.

b. Cooling Pond Calculations for Midland Plant (February 1969)

A Bechtel thermal performance comparison between 625 acre and 880 acre cooling pond sizes provided factors considered in pond sizing.

c. Midland Condenser Optimization Study (March 1969)

Within this Bechtel study the 880 acre pond size was recommended based on optimizing of the condenser performance.

d. Midland Cooling Pond Model Studies (May 1971)

Recommendations based on analysis of preliminary cooling pond model studies performed at Alden Research Laboratories to support the design of the Midland cooling pond.

e. Cooling Pond Thermal Performance Summary Report (August 1973)

This is a combined progress and thermal performance report summarizing what Bechtel has done on cooling pond design since 1968. It also summarizes the thermal performance of the 880 acre cooling pond.

Additional studies have been conducted since these were published which evaluate pond thermal performance but were not used in selecting the 880 acre pond size.

3. Refer to the list of studies provided in response to question 2. The thermal performance studies performed by Bechtel to support the cooling pond size were based on meteorological data from the following:

U.S. Weather Bureau
Michigan Data for
Midland, Flint, East Lansing and/or Tri-City Airport

Alden Research Laboratories provided data from physical model studies under subcontract to Bechtel.

4. The principles and methodology used in the Midland cooling pond thermal performance analysis are well accepted in practice and have been verified through operation or tested against field data at a minimum of eight

plants. These plants, some of which are located in Nebraska, Virginia, Illinois, South Carolina and South Dakota, cover a wide range of climatic conditions and geographic locations, including the Dresden Plant in Illinois.

In addition to many detailed analytical evaluations of the Midland cooling pond, summarized in part by the Reference, a physical model study was performed. Physical models provide a better understanding of the physical phenomena in a complex environment. When physical model studies are used in conjunction with analytical evaluations they provide added assurance that the design will accomplish the desired results.

Simple analytical techniques, similar to those used in the early stages of evaluating the Midland pond thermal performance, were applied to field data from the Four Corners plant. One approach included the use of temperature distribution information developed by the physical model studies of the Midland cooling pond. The results are reported in the Reference and are reasonable for this limited comparison. The methodology used to determine the Midland cooling pond transient thermal performance was no way affected, adjusted or altered based on the above comparisons with the Four Corners field data. Experiences in cooling pond operation at the Four Corners plant has no affect on the Midland cooling pond thermal performance analyses.

Reference: "Cooling Pond Thermal Performance, Summary Report" Midland Plant Units 1 & 2, Bechtel Incorporated, for Consumers Power Company, August 1973.

5. Consistent with the definitions provided by the intervenor in the preface to her interrogatories, CP Co interprets this question as referring to the NRC Staff, not the CP Co Staff; hence we have no direct response.

6. Neither Consumers Power Company nor Bechtel has sent personnel to the Dresden cooling pond for the purpose of observing climatic or meteorological conditions associated with pond operation. However, Consumers Power has retained Murray and Trettel Inc, for the Midland fog and ice monitoring studies. Mr J P Bradley of Murray and Trettel was a principal investigator/project manager for Commonwealth Edison's steam fog impact studies for Dresden. In this capacity, Mr Bradley participated in the field observations at Dresden.

7. As noted above, no field observations were made by Consumers Power or Bechtel personnel. The Murray and Trettel observations, in which J P Bradley participated, occurred during the periods of December 1971 to March 1973 and November 1977 to March 1978.

8. Murray and Trettel has issued the following two reports documenting their field observations at Dresden:

1. Report on Meteorological Aspects of Operating the Cooling Pond and Sprays at the Dresden Nuclear Power Station, Murray and Trettel Inc, Chicago, IL, August 1973 1001-1005.

2. Report on Steam Fog Impact Engineering at Dresden Nuclear Power Station Report #1183 Murray and Trettel Inc, Chicago, IL, May 1978.

9. The Dresden Station cooling lake is 1275 acres (Dresden FES-OL, Section 3.4.3.a, November 1973).

The Midland Plant cooling pond is 880 acres (Midland FES-OL, Section 4.2.4.2, July 1982).

10. There are three nuclear reactors at Dresden. Unit 1, which does not have its heat load rejected to the cooling pond, was shut down in late 1978 or 1979 and has not operated since then. Units 2 and 3 are the other two reactors at Dresden, which do have their heat load rejected to the cooling pond.

11. During full operation of the Midland Plant Units 1 and 2, the heat load rejected to the cooling pond can vary from 7.69 to 9.05 x 10^9 Btu/hr dependent on the amount of process steam being sent to Dow Chemical Company (Midland FES-OL, Table 4.2, July 1982). This variance in heat load rejection represents the range of plant operations from maximum guaranteed process steam load to back-end limited on Unit 1 with Unit 2 valves wide open.

The heat load rejected to the Dresden cooling lake by Dresden Units 2 and 3 at design conditions is 11.2 x 10^9 Btu/hr (Dresden FES-OL, Section 3.4, November 1973).

Comparing the respective design heat loads to cooling pond acreage, Midland can vary from 8.74 to 10.28 x 10^6 Btu/hr/ac as contrasted with Dresden at 8.78 x 10^6 Btu/hr/ac.

12. The factors referred to in the question are condenser inlet temperature; average pond surface temperature; total evaporation; percent

imposed heat load lost by evaporation. All of these are determined from the principles of cooling pond thermal performance. These well accepted principles are based on a balance of energy added to, subtracted from, and stored in the water body.

Energy added to a cooling pond consists of net solar radiation, net atmospheric radiation and plant heat load rejected to the circulating water at the condenser. Energy removed from the cooling pond consist of back radiation and evaporation and conduction processes. This outgoing energy is a function of the water surface temperature.

Data used as input to the computations that evaluated the energy balance for the Midland cooling pond include:

- dry bulb temperature
- dew point temperature (or relative humidity)
- wind speed
- cloud cover
- barometric pressure
- solar radiation

These data were available from Michigan weather bureau data at Midland (Dow Chemical), Flint, East Lansing or the Tri-City Airport.

A more complete discussion of the methods of analysis, data, and studies are presented in the referenced report.

Reference: "Cooling Pond Thermal Performance, Summary Report", Midland Plant Units 1 & 2, Bechtel Incorporated, for Consumers Power Company, August 1973.

13. Consumers Power Company has had two separate analytical studies conducted on cooling pond fog and its environmental impact in the vicinity of the Midland Plant cooling pond. These studies, their mode of publication or discussion and their availability are listed below.

a. Bechtel Company, The Environmental Effects of the Midland Plant Cooling Pond: Interim Report, June 1971. This study was included as an attachment to Section 4.3 of the Applicant's Supplemental Environmental Report (ASER) and was available since published in the ASER in October 1971.

The final version (Summary Report) was issued on April 28, 1972. The Bechtel study (both interim and summary reports) were available at and thoroughly discussed (even by the intervenor's witness) at the ASLB Construction Permit Hearings in 1972.

b. D J Portman and M R Weber, Fog and Plumes From Power Plant Cooling Systems in the Tri-Cities-Saginaw Bay Area, pp 1-5, 44-65 and 68-71, June 1975. This study was included as Appendix 5.1C of the Environmental Report ER-OLS and has been available since published in the ER-OLS in April 1978.

The issue of cooling pond and its environmental impact has also received coverage in the following other documents listed below.

a. Consumers Power Company, Applicant's Supplemental Environmental Report, Section 4.3 "Fogging Effect of Cooling Pond Operation," Section 5.1.3.1.K "Environmental Costs (Items 11.1 and 11.2), October 19, 1971 (as amended through January 7, 1972). This information has been available since published in October 1971 and was thoroughly discussed at the ASLB Construction Permit Hearings in 1972.

b. US Atomic Energy Commission, Final Environmental Statement Related to the Construction of Midland Plant Units 1 and 2, Section V.A.2 "Fogging and Icing," Section VII "Adverse Impacts That Cannot Be Avoided," Section XI.B.3 "Summary of Cost-Benefit Analysis: Impact on Air and Land," Section XII.B "Discussion of Comments Received...: Fog Occurrence and Extent," March 1972. This information has been available since published in March 1972 and was thoroughly discussed at the ASLB Construction Permit Hearings in 1972.

c. Consumers Power Company, ER-OLS, Section 5.1.4.1 "Frequency of Fog Occurrence, Section 6.1.3.1.8 "Fog and Ice Monitoring," Section 6.2.3.1.2 "Operational Fog and Ice Monitoring Program," Section 6.2A-3.1.1.2.3 "Fog and Ice Formation," Section "NRC Questions and Responses" Met 2, Met 3, Met 4, Met 5, Met 6, Met 7, Met 8, Met 9, Met 10, Met 11, Met 12, submitted for docketing April 12, 1978 (as amended through Revision 13 - December 1981, January 4, 1982). This information has been available since published in April 1978.

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

ATOMIC SAFETY AND LICENSING BOARD

In the Matter of CONSUMERS POWER COMPANY (Midland Plant, Units 1 and 2)	Docket No 50-329 OM 50-330 OM Docket No 50-329 OL 50-330 OL

September 8, 1982

AFFIDAVIT OF PHILIP A DI BENEDETTO

My name is Philip A Di Benedetto. I am the Engineering Manager at Nutech Engineers-Bethesda. In this capacity, I am presently involved in providing technical consulting services to several utilities on equipment qualification program development.

I am primarily responsible for providing a response to Interrogatory III, Question 3, concerning Contention 7. To the best of my knowledge and belief, the above information and the responses to the above interrogatory are true and correct.

Philip A. Di Benedetto

Sworn and Subscribed Before Me This 8 Day of Sept 1982

Beverly A Bross
Notary Public
Washtenaw County, Michigan

My Commission Expires November 30, 1982

mi0982-0225a100

BEVERLY A. BROSS
NOTARY PUBLIC, WASHTENAW CO., MICH
MY COMMISSION EXPIRES NOV.30, 1982

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

ATOMIC SAFETY AND LICENSING BOARD

In the Matter of CONSUMERS POWER COMPANY (Midland Plant, Units 1 and 2)	Docket No 50-329 OM 50-330 OM Docket No 50-329 OL 50-330 OL

September 10, 1982

AFFIDAVIT OF PETER W JACOBSEN

My name is Peter W Jacobsen. I am a Senior Engineer in Technical Services Section of Midland Design Production Department. In this capacity, my responsibilities include coordination of the environmental qualification program.

I am primarily responsible for providing a response to Interrogatory III, Questions 1 and 2 concerning Mary Sinclair Contention 7. To the best of my knowledge and belief, the above information and the responses to the above interrogatory are true and correct.

Peter W Jacobsen

Sworn and Subscribed Before Me This 10th Day of Sept 1982

Pamela J. Griffin
Notary Public
Jackson County, Michigan

My Commission Expires Sept 8, 1984

mi0782-2265c100

Mary P Sinclair

Interrogatory III

Contention 7 deals with the effects of low doses of radiation on polymer cable insulation and jacketing and the synergistic effects of radiation and temperature in degrading those materials.

Questions

1. Send all correspondence with the NRC on this study and its implications for the Midland nuclear plants.

2. What percentage of the plant's electrical wiring system is not accessible to inspection once the plant is started?

3. For what percentage of the estimated lifetime of these plants can the cable insulation be expected to retain its integrity before its degradation by synergistic effects makes it unsafe?

Responses

1. There has been no formal technical correspondence with the NRC on this study and its implications for the cable used at Midland to date. However a letter from J E Brunner to M Wilcove dated August 30, 1982 indicating we would be requesting information from the NRC Staff has already been copied to Ms Sinclair. The letter requesting technical information on the Sandia Reports NUREG/CR-2156 and NUREG/CR-2157 is being prepared by Consumers Power Company. As part of normal distribution, Ms Sinclair and all other persons on the service list will be sent copies of this letter.

2. A detailed assessment of all electrical cables in the plant to determine accessibility for inspection has not been conducted. However, a review of all cable types most likely to be affected by the phenomenon reported in the Sandia Study, in their worst case application (ie, highest potential for accelerated damage due to synergistic effect as discussed in the Sandia Reports) has shown that these cable types are

accessible for inspection with varying degrees of difficulty. Such difficulties include work in high radiation areas, possible need for plant shutdown, whether or not the cable can be inspected in place, etc, and depend on the method of inspection.

3. It is anticipated that all Class 1E cable within the Midland plant will maintain its electrical integrity throughout the 40 year life of the plant. This is based on the testing and analyses performed, as well as operating experience throughout the nuclear industry.

Testing performed for Consumers Power Company on actual cable used in safety circuits has demonstrated a qualified life of 40 years and has further demonstrated the ability of the cable to perform its safety function following simulation (by test) of a design basis event (e.g., loss of coolant accident). This testing did not take into account the synergistic effect of low dose rate and mildly elevated temperature suggested by the Sandia Reports. However, a preliminary analysis of the Sandia Reports indicates that although some accelerated degradation is caused by the low dose rate and mildly elevated temperature synergistic effect, the total amount of degradation does not impact the Midland cables' integrity or their ability to perform their safety function during or after the design basis events. On this basis, we expect to show that although some additional degradation is attributable to the synergistic effect of low dose rate and mildly elevated temperature its impact on the overall life of the cable is not significant.

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CERTIFICATE OF SERVICE

I hereby certify that copies of the attached responses of Consumers Power Company to Discovery Questions of Interviewer Mary P Sinclair were sent by U S Mail, first class, postage prepaid, to the attached service list this 15th day of September, except for Mary Sinclair, who was served by Federal Express.

Philip P Steptoe

Philip P Steptoe