REGULATORY INFORMATION DISTRIBUTION SYSTEM (RIDS)

ACCESSION NBR: 8710270344 DOC. DATE: 87/10/19 NOTARIZED: NO DOCKET #
FACIL: 50-315 Donald C. Cook Nuclear Power Plant, Unit 1, Indiana & 05000315

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RECIP. NAME	RECIPIENT AFFILIATION

SUBJECT: LER 87-020-00: on 870917, determined that in event of fault of balance-of -plant cables, loss of control power on both independent trians of related panels could occur. Caused by deficient design. Design changes implemented. W/871019 ltr.

DISTRIBUTION CODE: IE22D COPIES RECEIVED: LTR 1 ENCL 1 SIZE: 7
TITLE: 50.73 Licensee Event Report (LER), Incident Rpt, etc.

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	NRR/DEST/ADS	1	0	NRR/DEST/CEB	1	1
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	NRR/DRIS/SIB	1	1	NRR/PMAS/ILRB	1	1
	REG FILE 02	1	1	RES DEPY GI	1	1
	RES TELFORD, J	1	1	RES/DE/EIB	1	1
	RGN3 FILE 01	1	1			
EXTERNAL:	EG&G GROH, M	5	5	H ST LOBBY WARD	1	1
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NRC Form 366 (9-83)

U.S. NUCLEAR REGULATORY COMMISSION
APPROVED OMB NO. 3150-0104
EXPIRES: 8/31/88

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
D. C. Cook Nuclear Plant, Unit 1	0 5 0 0 0 3 1 5	1 OF 0 6

TITLE (4) Lack of Isolation Between Balance of Plant and Essential Safety System Loads Due to Potential Design Deficiency

EVENT DATE (5)			LER NUMBER (6)			REPORT DATE (7)			OTHER FACILITIES INVOLVED (8)	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR	FACILITY NAMES	DOCKET NUMBER(S)
0 9	1 7	8 7	8 7	— 0 2 0 —	0 0	1 0	1 9	8 7	D.C. Cook, Unit 2	0 5 0 0 0 3 1 6
										0 5 0 0 0

OPERATING MODE (9): 5

POWER LEVEL (10): 0 0 0

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: *(Check one or more of the following)* (11)

[ ] 20.402(b)	[ ] 20.405(c)	[ ] 50.73(a)(2)(iv)	[ ] 73.71(b)
[ ] 20.405(a)(1)(i)	[ ] 50.36(c)(1)	[ ] 50.73(a)(2)(v)	[ ] 73.71(c)
[ ] 20.405(a)(1)(ii)	[ ] 50.36(c)(2)	[ ] 50.73(a)(2)(vii)	[ ] OTHER *(Specify in Abstract below and in Text, NRC Form 366A)*
[ ] 20.405(a)(1)(iii)	[ ] 50.73(a)(2)(i)	[ ] 50.73(a)(2)(viii)(A)	
[ ] 20.405(a)(1)(iv)	[X] 50.73(a)(2)(ii)	[ ] 50.73(a)(2)(viii)(B)	
[ ] 20.405(a)(1)(v)	[ ] 50.73(a)(2)(iii)	[ ] 50.73(a)(2)(x)	

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (AREA CODE)
John R. Sampson - Safety & Assessment Superintendent	6 1 6 4 6 5 - 5 9 0 1

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	REPORTABLE TO NPRDS

SUPPLEMENTAL REPORT EXPECTED (14)

[ ] YES *(If yes, complete EXPECTED SUBMISSION DATE)* [X] NO

EXPECTED SUBMISSION DATE (15): MONTH | DAY | YEAR

ABSTRACT *(Limit to 1400 spaces, i.e., approximately fifteen single-space typewritten lines)* (16)

On September 17, 1987, during a review of Safety System Functional Inspection on our auxiliary feedwater system, it was determined that in the event of a fault in certain Balance Of Plant (BOP) cables, which would involve distribution panels from both independent trains, a loss of control power on both independent trains of related (ESS) panels could occur.

The ESS loads that may have been affected are certain containment isolation valves, reactor head vent valves, post-accident sampling valves, and steam generator stop valve dump valves.

The cause of this event was a potentially deficient design which could have caused insufficient breaker interrupting capability, in conjunction with a lack of isolation between BOP and ESS loads.

For both Units 1 and 2, design changes were implemented to isolate the BOP loads from the ESS loads. Procedural changes, which had been implemented prior to the identification but after the occurrence of this problem, would now necessitate the review of qualification documentation and technical studies of the breaker capabilities.

8710270344 871019
PDR ADOCK 05000315
S PDR

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D. C. Cook Nuclear Plant, Unit 1	0 5 0 0 0 3 1 5	8 7	— 0 2 0	— 0 0	0 2 OF 0 6

TEXT (If more space is required, use additional NRC Form 366A's) (17)

Conditions Prior to Occurrence

Unit 1 and Unit 2 in Mode 5 (cold shutdown)

Description of Event

On September 17, 1987, at approximately 1730 hours, during a review of an internal Safety System Functional Inspection on our auxiliary feedwater system (EIIS/BA), it was determined that in the event of a fault in certain Balance Of Plant (BOP) cables (EIIS/CBL), which would involve distribution panels from both independent trains (see Attachment 1), a loss of control power on both independent trains of related Essential Safety System (ESS) distribution panels could occur.

The following distribution panels (EIIS/BL) utilize electrical circuit breakers (EIIS/BKR) manufactured by the Heinemann Electric Company (Series 0441).

(Refer to Attachment 1)

Unit 1:	Train A	1-CCV-CD	1-SSV-A1, 1-SSV-A2
	Train B	1-CCV-AB	1-SSV-B
Unit 2	Train A	2-CCV-CD	2-SSV-A1, 2-SSV-A2
	Train B	2-CCV-AB,	2-SSV-B

The breakers are used for 250 VDC service. It was discovered that the interrupting capability of the Heinemann breakers, as used for D. C. Cook, is not specified by the manufacturer at 250 VDC. Furthermore, with the specified lower voltage (125 VDC) interrupting curves (trip current vs. time), a comparison of the breakers with their upstream fuses (Gould-Shawmut Trionic, or Bussman FRN) (EIIS/FU) indicates a lack of coordination for fault currents above approximately 1500 amps. Investigation concluded that this condition has existed since the initial startup of the units.

There were no inoperative structures, components or systems that contributed to this event.

Cause of Event

The cause of this event was a potential deficient design which could have caused insufficient breaker interrupting capability in conjunction with a lack of isolation betewwn BOP and ESS loads.

Analysis of Event

This event was determined to be reportable per 10 CFR 50.73 (a)(2)(ii)(B), the condition represented a violation of our design basis in that both trains of ESS equipment may have been adversely affected by a failure in the BOP equipment. The actual safety significance, however, is believed to be small. For a safety problem to have existed, BOP cables from related panels would have had to completely fail and interact simultaneously with an accident which would have required the affected ESS equipment to function. This is considered highly unlikely.

The ESS loads that may have been affected are certain containment isolation valves, reactor head vent valves, post-accident sampling valves, and steam generator stop valve dump valves.

The affected containment isolation valves and reactor head vent valves would have failed to the safe position (i.e., closed) on a loss of control power. For the containment isolation valves, this would have assured containment isolation in the event of an accident. For the Reactor head vent valves, isolation assures that uncontrolled RCS leakage will not occur in the event of a loss of control power.

The reactor head vent valves would not need to be opened at the very beginning of an accident. Should opening the valves have been necessary, they most likely would have been repowered if conditions warranted it.

Like the reactor head vent valves, post-accident sampling and post-accident hydrogen monitoring valves would not be required to be open at the very beginning of an accident. Upon loss of control power, these valves, which are normally closed, fail to the closed position to ensure the systems are adequately isolated. Should they be required to be opened post-accident, they most likely would have been repowered if conditions warranted it.

The steam generator stop valve dump valves cause the fast closure of the steam generator stop valves. If power was lost to both trains of the dump valves, the steam generator stop valves would not have been capable of automatically closing from the fast closure trip. However, the stop valves could have still been driven closed though the hydraulic closure mechanisms. Controls for the hydraulic closures are located in the control room. The Emergency Operating Procedures instruct the operators to close the stop valves if there is no indication on the control boards of normal closure status following a trip signal. This would have been done by the use of the normal hydraulic closure control switch. Finally, as noted before, the probability of a failure of the type that would cause failure of both trains of dump valves concurrent with an accident which requires automatic closure of the valves, is considered to be small.

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TEXT (If more space is required, use additional NRC Form 366A's) (17)

For the reasons detailed above, the condition reported in this LER is not considered to represent a significant risk to public health and safety.

Corrective Action

(Refer to Attachment 2)

For both Units 1 and 2, design changes were implemented to isolate the BOP loads from the ESS loads. The BOP loads were removed from panels CCV-AB and CCV-CD and relocated to a BOP panel, VDAB-2. BOP loads on panel SSV-A1 and SSV-B were transferred to panel SSV-A2. The existing ESS load cables on panel SSV-A2 were pulled back and reconnected to panel SSV-A1. A Class 1E feeder fuse, added to panel SSV-A2, would now isolate faults in the BOP circuitry from the Class 1E circuitry.

Procedural changes, which had been implemented prior to the identification of this problem but after the occurrence of this problem, would now necessitate the review of qualification documentation and technical studies of the breaker capabilities.

Failed Component Identification

No component failures were identified during the investigation of this event.

Previous Similar Events

None.

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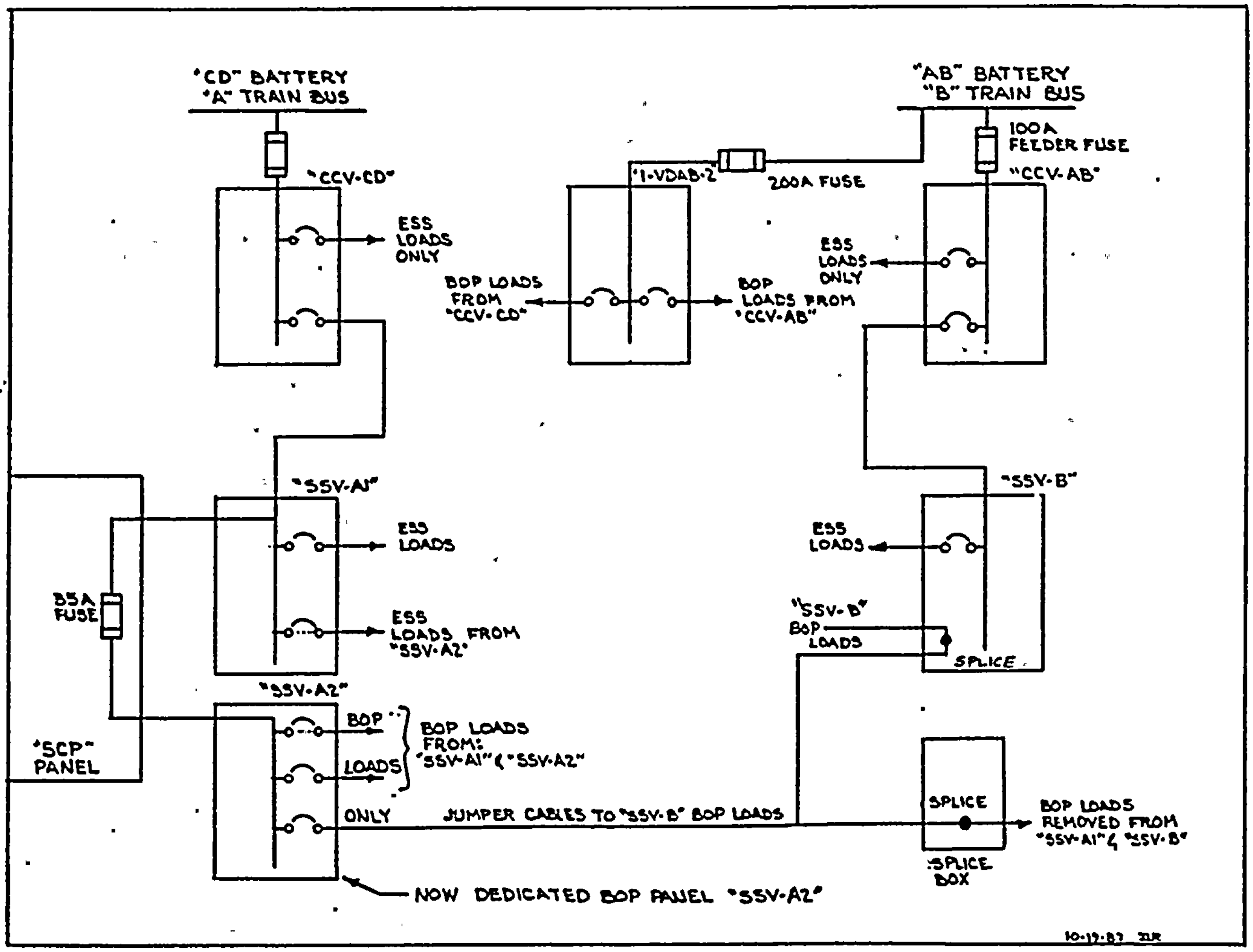
TEXT (If more space is required, use additional NRC Form 366A's) (17)

ATTACHMENT NO. 2
DESIGN CHANGES



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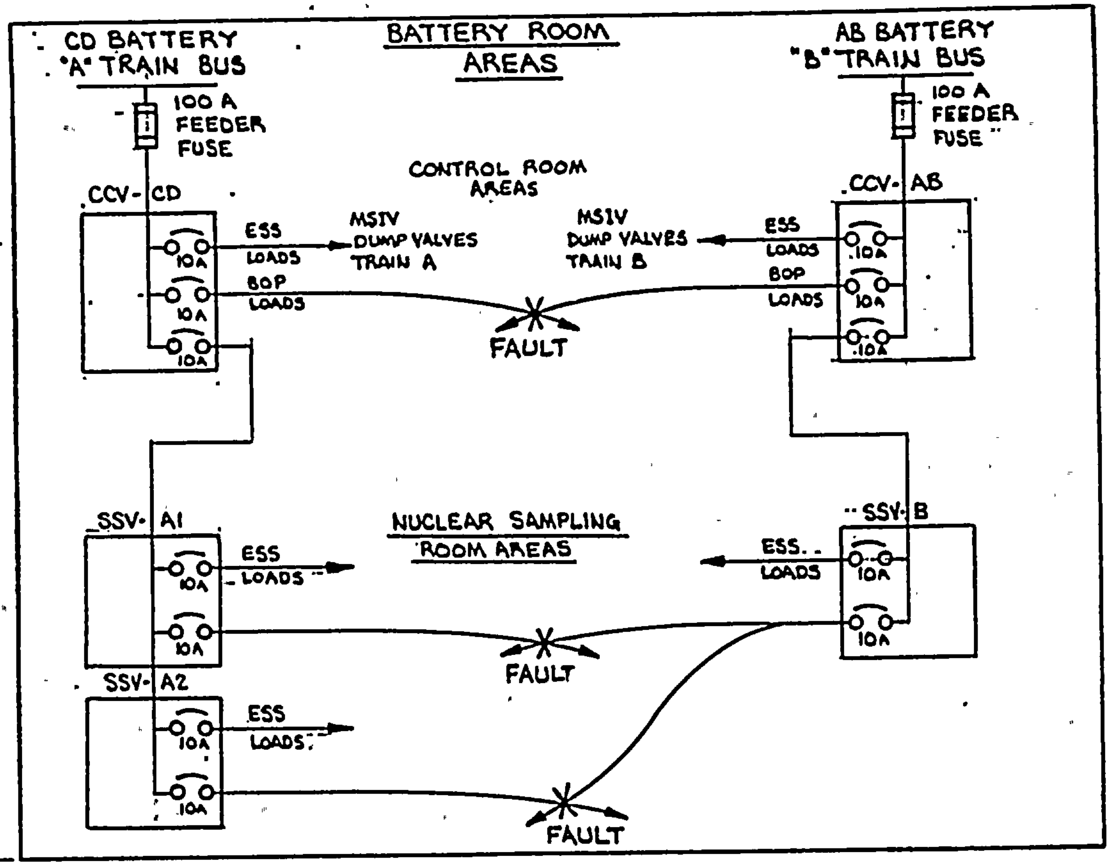
TEXT (If more space is required, use additional NRC Form 366A's) (17)

ATTACHMENT NO. 1
DEMONSTRATION OF FAULT CONDITIONS



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INDIANA MICHIGAN POWER

October 19, 1987

United States Nuclear Regulatory Commission
Document Control Desk
Washington, D.C. 20555

Operating License DPR-58
Docket No. 50-315

Document Control Manager:

In accordance with the criteria established by 10 CFR 50.73 entitled Licensee Event Reporting System, the following report is being submitted:

87-020-0

Sincerely,

W. G. Smith, Jr.
Plant Manager

WGS:afh

Attachment

cc: John E. Dolan
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