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FACIL:50-315 Donald C. Cook Nuclear Power Plant, Unit 1, Indiana M 05000315
AUTH.NAME AUTHOR AFFILIATION
REINIGER,J.A. Indiana Michigan Power Co. (formerly Indiana & Michigan Ele
BLIND,A.A. Indiana Michigan Power Co. (formerly Indiana & Michigan Ele
RECIP.NAME RECIPIENT AFFILIATION

SUBJECT: LER 92-001-00:on 920128,determined that slabs at elevation 621 ft,6-inches in west main steam encls could collapse in event of postulated steam line break.Caused by inadequate configuration control.Change implemented.W/920227 ltr.

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TITLE: 50.73/50.9 Licensee Event Report (LER), Incident Rpt, etc.

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INDIANA
MICHIGAN
POWER

February 27, 1992

United States Nuclear Regulatory Commission
Document Control Desk
Rockville, Maryland 20852

Operating Licenses DPR-58
Docket No. 50-315

Document Control Manager:

In accordance with the criteria established by 10 CFR 50.59 entitled Licensee Event Report System, the following report is being submitted:

92-001-00

Sincerely,

for
A. A. Blind
Plant Manager

/sb

Attachment

c: D. H. Williams, Jr.
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090053

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U.S. NUCLEAR REGULATORY COMMISSION

APPROVED OMB NO. 3150-0104
EXPIRES: 4/30/92

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Donald C. Cook Nuclear Plant - Unit 1	0 5 0 0 0 3 1 5	1 OF 0 4

TITLE (4) Slabs at El 621'6" in the West Main Steam Enclosures for Unit 1 (Unit 2) Did Not Meet Design Basis Requirements Due to Inadequate Configuration Controls in 1973

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 1	2 8	9 2	9 2	— 0 0 1 —	0 0	0 2	2 7	9 2	D.C. Cook Plant Unit 2	0 5 0 0 0 3 1 6
										0 5 0 0 0

OPERATING MODE (9): 1

POWER LEVEL (10): 1 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)
J. A. Reiniger - Senior Structural Engineer	6 1 4 2 2 3 - 1 0 0 0

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)

At 1650 on January 28, 1992, it was determined that the slabs at El 621' 6" in the west main steam enclosures for Unit 1 and Unit 2 (one slab per unit) would, based on design basis high energy line break (HELB) analysis results, collapse in an uncontrolled manner in the event of a postulated steam line break in the West Main Steam Enclosures.

A best estimate calculation of the steam enclosure pressures resulting from a steam line break was made using the RELAP5/MOD2 computer code, and a structural analysis of the slab was made using these values. Based on this analysis, it was determined that the slab would remain functional if a full circumferential rupture of a main steam line were to occur, although some cracking can be expected in the areas of high stress concentration.

Inadequate configuration control, design control and design change control practices in 1973 were the cause of this condition. During the 1992 refueling outages, a design change will be implemented to strengthen the floor slab so it accommodates the design basis loads. Current configuration control, design control and design change control practices are proceduralized in General Procedures. The lessons from this problem have already been previously learned and formalized; therefore, no new preventive action is planned at this time.

NRC FORM 366A (6-89)

U.S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

APPROVED OMB NO. 3150-0104
EXPIRES: 4/30/92

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
Donald C. Cook Nuclear Plant - Unit 1	0 5 0 0 0 3 1 5	9 2	— 0 0 1	— 0 0	0 2 OF 0 4

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

Description of Event

At 1650 on January 28, 1992 it was determined that the slabs (EIIS/NM-SPT) at El 621' 6" in the west main steam enclosures for Unit 1 and Unit 2 (one slab per unit) would, based on design basis high energy line break (HELB) analysis, collapse in an uncontrolled manner in the event of a postulated steam line break in the West Main Steam Enclosures (WMSE).

The inability of the slab to withstand design loads was discovered during the investigation of previously identified concerns. Initially (September 1990) water from a leaking flange was found to be dripping into the room under the flange's location. During the investigation of this problem, a discrepancy between the slab and the drawings was identified (October 1990); specifically, the slab had saw cuts that were not identified in the drawings.

AEPSC filed Appendix O of the SAR for the Donald C. Cook Plant in June 1973. Appendix O deals with the consequences of high energy line breaks outside Containment, including the consequences of the postulated steam line and feedwater line breaks in the West Main Steam Enclosures. The supporting evaluation identified the as-built El. 621' 6" concrete slab as inadequate to withstand the peak pressure from the steam line break and resulted in remedial actions to prevent the uncontrolled collapse of the slab. The remedial actions consisted of introducing saw cuts in the cantilever portion of the slab (to increase ductility) and additional posts under the integral beam supporting the cantilever slab (to increase load carrying capacity). A review of these files yielded a report titled, "A Summary of the Findings of the High Energy Line Break Task Force on Modifications Required to Assure Safe Plant Shutdown Following a Postulated High Energy Line Rupture Outside of Containment." The slab was saw cut, based on the recommendations in the report; however, this information was not shown on the appropriate drawing. During the review of the files, it was also discovered that the calculation analyzing the slab and locating the saw cuts was apparently checked and approved, but the documentation of these tasks was not attached to the calculation. We had assumed that the conclusions and actions of the HELB Task Force of 1973 were correct and that the October 1990 condition was a documentation problem.

As stated above, this condition was considered to be a documentation deficiency, and an effort was undertaken to recreate the documentation. This effort included an analysis using as-built information. By January 28, 1992, the operability and design basis analyses for the Unit 2 WMSE El 621' 6" floor slab system had progressed to the point that the original assumptions had to be reassessed. A problem report was written to report our findings. The findings showed that the slab was not capable of withstanding the original design pressures, loadings due to a double-ended rupture of a main steam line above the slab. Because of the similarities between the two units, the same situation is presumed to exist in Unit 1.

A best estimate calculation of the steam enclosure pressures resulting from a steam line break was made using the RELAP5/MOD2 computer code, and a structural analysis of the slab was made using these values. Based on this analysis, it was determined that the slab would remain functional if a full circumferential rupture of a main steam line were to occur, although some cracking can be expected in the areas of high stress concentration.

LICENSEE EVENT REPORT (LER)
TEXT CONTINUATION

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
Donald C. Cook Nuclear Plant - Unit 1	0 5 0 0 0 3 1 5	9 2	— 0 0 1	— 0 0	0 3 OF 0 4

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

Description of Event (Cont'd)

This condition was reported to the NRC under the provisions of 10CFR50.72 (b)(1)(ii)(B), a condition that was outside the design basis of the plant.

Cause of Event

Inadequate configuration control, design control and design change control practices were the cause of this event.

Even considering that the tools available to the structural engineers performing the original design evaluations were not as user friendly as the ones available now, and that the standards of the time were not as high as today's standards, the design calculations were inadequate mainly due to the failure to implement self-checking and documentation practices. This may have caused the engineers to have overlooked or not recognized the shear problem in the beam.

The original shear reinforcement design for the beam was inadequate as the rebar stirrups were spaced at 12" on centers in a beam with an effective depth of less than 16", causing the beam to be unreinforced in shear. The maximum spacing is required to be half the beam's effective depth. It is not possible to tell at this point if the structural engineers reviewing the beam for the HELB effects considered the shear reinforcement present to be capable of contributing to the shear strength of the beam in proportion to stirrup size and spacing, unaware of the threshold spacing requirement.

Analysis of Event

The condition is considered reportable pursuant to 10CFR50.73 (a)(2)(ii)(b) as a condition outside the design basis of the plant.

The subject slab is a floor slab in the west main steam enclosure. Immediately under this slab is the west non-essential service water control and isolation valve area of the auxiliary building.

A best estimate analysis of the pressure that would be imposed on the slab was completed using the RELAP5/MOD2 computer code, and the calculation shows that the pressure would be 18.7 psia (a 4 psi differential across the slab since the lower room remains at atmospheric pressure). A structural evaluation of the slab has shown that, with a 4 psid loading, minor cracking of the top of the slab would occur at the points of high stress concentration; however, the slab would remain functional.

The pressures in the main steam enclosure were calculated assuming an instantaneous full area break of a 30-inch steam line (4.6 $ft.^2$ area). Failure of the steam line in this manner is not considered to be credible. This line is made of high quality materials and has operated for approximately 15 years without incident. The most likely failure mechanism would be a small detectable leak which would have minimal consequences on the pressure in the main steam enclosure.

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
Donald C. Cook Nuclear Plant - Unit 1	0 5 0 0 0 3 1 5	9 2	— 0 0 1 —	0 0	0 4 OF 0 4

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

Analysis of Event (Cont'd)

Based on the above, it is concluded that the current slab condition does not represent a significant safety concern, nor a significant hazard to the Health and Safety of the general public.

Corrective Actions

We are reconstructing the calculations supporting the as-found condition and will revise the design drawings to reflect the as-found condition.

During the 1992 refueling outages, a design change will be implemented to strengthen the floor slab so it can accommodate the design basis loads. The changes are documented in RFC DC-12-3100.

Current configuration control, design control and design change control practices are proceduralized in the AEPSC General Procedures (GPs). The lessons from this event have already been learned and formalized; therefore, no new preventive action is planned at this time.

Failed Component Identification

None

Previous Similar Events

LER 50-315/91-005