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ACCESSION NBR: 8704140255 DOC. DATE: 87/04/07 NOTARIZED: NO DOCKET #
FACIL: 50-315 Donald C. Cook Nuclear Power Plant, Unit 1, Indiana & 05000315

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SUBJECT: LER 86-021-01: on 860925, RHR sys may have been in analyzed configuration re LOCA analysis. Caused by misunderstanding of Tech Spec 3.5.2. Caution tags placed on safety injection & RHR sys to prevent isolation. W/870407 ltr.

DISTRIBUTION CODE: IE22D COPIES RECEIVED: LTR 1 ENCL 1 SIZE: 8
TITLE: 50.73 Licensee Event Report (LER), Incident Rpt, etc.

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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 7

TITLE (4) Lack of Specificity in Technical Specification Requirements Resulted in Operation With Unanalyzed Emergency Core Cooling System Configuration (Residual Heat Removal)

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	2 5	8 6	8 6	— 0 2 1 —	0 1	0 4	0 7	8 7	D. C. Cook, Unit 2	0 5 0 0 0 3 1 6
										0 5 0 0 0

OPERATING MODE (9): 1

POWER LEVEL (10): 0 9 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)	
K. R. Baker, Operations 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	MANUFACTURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	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 25, 1986, we determined that the Residual Heat Removal (RHR) System may have been in an unanalyzed configuration in respect to the loss-of-coolant accident (LOCA) analysis. This determination was made as a result of an investigation prompted by Licensee Event Report (LER) 316-86-026 concerning Emergency Core Cooling System (ECCS) cross-tie valve requirements.

We believe the cause of the event was a legitimate misunderstanding of Technical Specifications (T/S) 3.5.2 regarding ECCS operability. We had assumed that this T/S was written to bound the safety analyses. Our incorrect assumption resulted in this event as well as the event cited in the Notice of Violation described in NRC Inspection Report 50-316/86042 and associated escalated enforcement action. Administrative controls have been placed on the cross-tie valves and other identified ECCS valves on both units to prevent isolation.

Evaluations have been performed by our fuel vendors to analyze a large-break LOCA with the RHR cross-tie closed. Based on this analysis our Unit 2 vendor judged that the events in this LER did not constitute a significant safety concern with regard to the requirements of 10 CFR 50.46. Our Unit 1 vendor did not have a firm technical basis for evaluating operation with both cross-ties closed (Safety Injection and RHR), but did conclude that with only one cross-tie closed we did not exceed the limits of 10 CFR 50.46.

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

Conditions Prior to Occurrence

Unit 1 - Mode 1 (Power Operation) - 90 percent reactor thermal power.
Unit 2 - Mode 1 (Power Operation) - 80 percent reactor thermal power.

Description of Event

On September 25, 1986, we determined that the Residual Heat Removal (RHR) System (EIIS-BP) may have been in an unanalyzed configuration in respect to the loss-of-coolant accident (LOCA) analysis. This determination was made as a result of an investigation prompted by Licensee Event Report (LER) 316-86-026.concerning Emergency Core Cooling System (ECCS) cross-tie valve (EIIS-MOV) requirements.

In the plant FSAR the large-break LOCA analysis requires that RHR injection be available to all four Reactor Coolant System(RCS) (EIIS-AB) cold legs.

If a single RHR pump (EIIS-BPP) were to fail, these flow paths are assured by maintaining the motor-operated valves IMO-314 and IMO-324 in the cross-tie between two pumps in the open position. An internal investigation revealed that past surveillance and maintenance valve configurations have resulted in injection paths available to only two of the four cold legs.

In researching past surveillance practices, we determined that both safety injection (SI) (EIIS-BQ) and RHR cross-ties have been closed several times to test a complete subsystem simultaneously. It was later determined that the simultaneous testing of an entire subsystem was not necessary. The procedures were revised in April 1985 to discontinue the simultaneous testing. Subsequent to the procedure revision, there were tests performed in which only one cross-tie (Either RHR or SI but not both) was closed. (Drawing on page 3).

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

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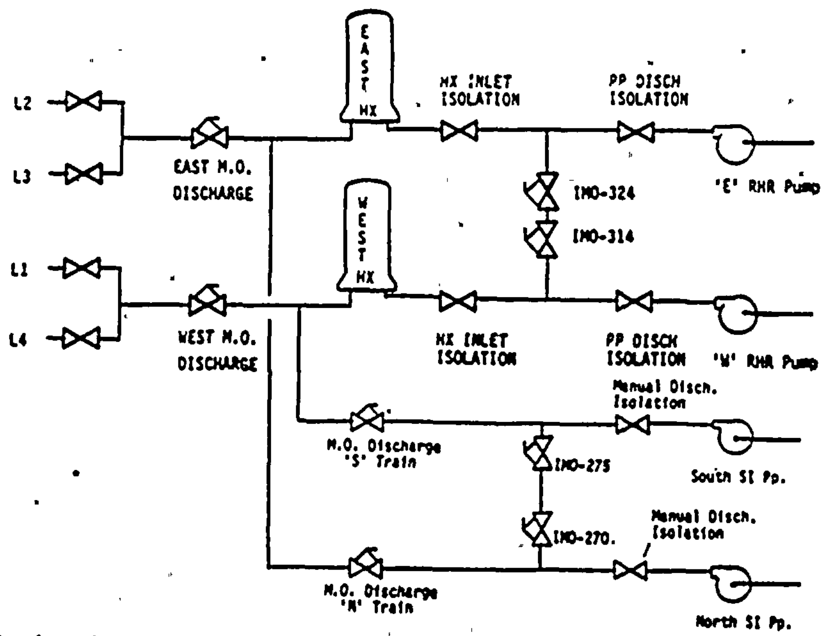
TEXT (If more space is required, use additional NRC Form 366A's) (17)



Cause of the Event

T/S 3.5.2 states that "two independent ECCS subsystems shall be OPERABLE with each subsystem comprised of:

a. One OPERABLE centrifugal charging pump,
b. One OPERABLE safety injection pump,
c. One OPERABLE residual heat removal heat exchanger,
d. One OPERABLE residual heat removal pump,
e. One OPERABLE flow path capable of taking suction from the refueling water storage tank on a safety injection signal and transferring suction to the containment sump during the recirculation phase of operation.

The T/S action statement allows one ECCS subsystem to be inoperable for up to 72 hours. This implies that there are two independent ECCS subsystems or trains with independent flow paths. It was assumed that the T/Ss were written to bound the safety analyses; however since the safety analyses assumes four injection points are available, our assumption was not valid.

Plant maintenance and testing procedures were written with the understanding that the ECCS consists of two independent subsystems. We believe this to be a legitimate misunderstanding of the requirements of the T/Ss. Further, from discussions with personnel from other nuclear plants and review of IE NOTICE 87-01 "RHR Valve Misalignment Causes Degradation of ECCS in PWRs", we believe this type of misunderstanding to be common within the PWR portion of the nuclear industry.

Analysis of Event

This event is considered reportable under the criteria set forth in 10 CFR 50.73(a)(2)(ii).

In order to evaluate the consequences of this event, we have contacted both of our fuel vendors, Westinghouse (Unit 1) and Advanced Nuclear Fuels (Unit 2). In support of one of the corrective actions associated with this LER and in support of operation with the RHR cross-tie only closed, both vendors are performing large-break LOCA evaluations assuming single train operation with the RHR cross-tie closed and the SI cross-tie open. The Westinghouse Unit 1 evaluation is presented below. Preliminary information from ANF also shows no problems with meeting the requirements of 10 CFR 50.46.

We also asked the vendors their opinion as to whether both cross-ties could be closed when operating on a single ECCS train. ANF responded that in their judgement the limits of 10 CFR 50.46 would not be exceeded with both cross-ties closed. Westinghouse stated they did not have a sufficient technical basis to form any conclusions at this time. Plant specific analyses with both cross-ties closed were not performed because of the excessive time and cost that were involved.

<u>Analysis Performed by Westinghouse Electric Corporation in Support of Corrective Action 4</u>

The D.C. Cook Unit One safety injection system consists of two residual heat removal (RHR) pumps, two centrifugal charging (CCP) pumps and two high head safety injection (HHSI) pumps. Each HHSI pump discharge line splits to deliver flow into two of the four cold legs, and a cross tie connects the two pump discharge lines enabling one pump to deliver flow to all four cold legs. The RHR pump discharge piping is configured the same way as the HHSI pumps. The design basis large break Loss-of-Coolant (LOCA) analyses assume that flow delivery is available through all four lines from each pump in the safety injection system.

The D.C. Cook Unit One licensing basis LOCA analysis includes both large and small break LOCA events. The large break LOCA result in not highly dependent on HHSI pump flow capability due to the rapid system depressurization below the RHR pump actuation pressure (114.7 psia). Recovery from a large break LOCA event is governed by the availability of pumped injection and accumulator water delivery during the reflood phase of the transient. Because the RHR pump delivers much more flow than the HHSI pump at low pressure, the reduction in the amount of total HHSI flow delivery due to cross-tie closure will not affect large break LOCA calculated ECCS performance as significantly as the large reduction caused by RHR pump cross tie closure.

* U S GPO 1986-0-624-538-455

The 10 CFR 50.46 analysis of the D. C. Cook Unit One large break LOCA is based upon 102% core power operation together with minimum safeguards and maximum safeguards safety injection (SI) flowrates. Minimum safeguards SI flowrates are (predicated) upon one charging pump, one high head safety injection pump and one low head safety injection (RHR) pump operating; appropriate assumptions are made about pump head/flow characteristic curves and system resistance to minimize delivery to the reactor coolant system (RCS). Maximum safeguards SI flowrates are computed assuming all SI pumps present operate under pump head/flow characteristic and system resistance conditions which maximize post-LOCA flow delivery to the RCS.

Full HHSI and CCP flow delivery will be available for the large break LOCA event even if two RHR delivery lines are unavailable due to cross tie closure.

We evaluated the limiting scenario with minimum safeguards SI flow with two RHR delivery lines unavailable during a large break LOCA event, i.e., the broken cold leg is one of the two which is aligned to receive flow from the one operative RHR pump. Under this scenario the pump can deliver only 180 lbs. per second to the RCS during the core reflood phase of the large break LOCA analysis, rather than the 370 lbs. per second modeled in the BART analysis presented in the Cook Unit One FSAR. Pumped safety injection is actuated during the latter part of the blowdown of the initial RCS inventory; it supplements the accumulators in providing the water necessary to refill the reactor vessel lower plenum and then the vessel downcomer. Review of the large break LOCA spectrum, including the limiting case break (C_D=0.6 DECLG) of minimum safeguards Cook Unit One cases, establishes that the vessel downcomer fills completely during accumulator injection in all cases.

Once the accumulators have emptied, the pumped safety injection must supply the water needed to maintain downcomer level during core reflood. The CCP and HHSI pumps taken together supply about 110 lbs. per second to replenish downcomer inventory; together with the RHR pump output which reaches the RCS through the one available delivery line, a total pumped injection rate of 290 lbs. per second is available in the minimum safeguards condition with the RHR cross-tie closed.

During the core reflood phase of the Cook Unit One C_D=0.6 DECLG minimum safeguards case, much of the pumped injection to RCS spills from the full downcomer out the break. Review of the predicted core reflood transient verifies that the 290 lbs/second available under minimum safeguards conditions, with two RHR lines presumed unavailable, is adequate to maintain the downcomer water level full. Therefore, the only impact the pumped flow reduction, due to two RHR lines being unavailable, exerts on the Cook Unit One large break LOCA performance is the delay incurred in vessel refilling while the accumulators are injecting. Reduction in total pumped injection

flow delays slightly the bottom-of-core recovery, the time at which the water level reaches the bottom of the fuel during the large break LOCA transient. This delay in turn extends the time interval during which adiabatic heatup of fuel rods must be computed per 10CFR50 Appendix K. Also, the pumped injection flow reduction will slightly delay the filling of the downcomer, giving a slightly diminished head of water at any point in time as the downcomer fills.

The peak clad temperature (PCT) impact of these delays is identified as the increase in temperature which occurs while they remain in effect; it is based upon the rate of adiabatic heatup at the end of lower plenum refill. For the scenario in which only one RHR pump delivery line is available to the RCS, RHR flow into the RCS will be virtually zero as long as the accumulators are delivering water. Assuming no RHR pump injection during blowdown and lower plenum refill, bottom-of-core recovery (BOCREC) will require an additional 0.2 seconds based upon the rate of fill prevalent at BOCREC time. Since the adiabatic heatup rate is 30°F/second at BOCREC, a 6°F increase in calculated PCT will be incurred. Considering also the impact of the slower downcomer fill, an overall penalty of approximately 10°F in calculated PCT is all that occurs in the limiting Cook Unit One minimum safeguards large break LOCA FSAR case, under the closed cross tie scenario. The established PCT for this case is 1937°F, so a large PCT margin remains to the regulatory limit of 2200°F.

The C_D=0.6 DECLG large break LOCA with maximum safeguards is the identified limiting case for Cook Unit One, at a calculated PCT of 2154°F. Since no failure of safety injection equipment is the basis for the maximum safeguards LOCA scenario, each RHR (or HHSI) pump delivers into two of the four cold legs whether the cross-tie is closed or open. The flow delivery remains as analyzed, so calculated PCT for this limiting case is unaffected by the cross-tie assumption.

Corrective Action

The following corrective actions have been taken:

1) Administrative controls, in the form of caution tags, were placed on the safety injection and residual heat removal systems in both units to prevent isolation of injection points.

2) An Operations Department memo, dated October 3, 1986, was issued to identify the affected valves.

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3) Guidance on entry into T/S 3.0.3 was issued to the plant by the Nuclear Safety and Licensing (NS&L) section of the American Electric Power Service Corporation (AEPSC) in a letter dated September 26, 1986. The letter stated that it was considered permissible by the NRC (both Region III and NRR) to voluntarily enter T/S 3.0.3 to perform planned activities. Train-related valve-stroking procedures were revised by December 24, 1986 to preclude isolation of injection points within the 72-hour action statement, but they do allow entry into T/S 3.0.3 for up to one hour.

4) We are presently in the process of submitting a series of safety evaluations which reanalyze the small and large-break LOCA scenarios for Units 1 and 2. One of those safety evaluations is included as part of this LER, for example purposes only. The new small-breaker LOCA analyses assume that one SI pump injects water into only two RCS loops, one of which is the break leg. The new large-break LOCA analyses assume that one RHR pump injects water into only two RCS loops (one of which spills), while one SI pump injects into all four RCS loops. If these analyses are accepted and the requested interpretation of T/Ss 3.5.2 and 3.5.3 is allowed, we will be able to isolate two injection points on the RHR or SI flow paths (not both simultaneously). The requested interpretation would allow for the needed flexibility to perform maintenance and surveillance procedures while in Modes 1-4. The initial letter (AEP:NRC:1024) to request this interpretation was submitted on March 23, 1987.

Failed Components Identification

None

Previous Similar Events

316-86-026 - This Licensee Event Report is being submitted as a result of a review conducted following the investigation of LER 86-026.

American Electric Power System
INDIANA & MICHIGAN ELECTRIC COMPANY
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April 7, 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 10CFR50.73 entitled Licensee Event Reporting System, the following report is being submitted:

86-021-01

A. Alan Blind

Sincerely,

W. G. Smith, Jr.
Plant Manager

/afh

Attachment

cc: John E. Dolan
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