CATEGORY 1

REGULATORY INFORMATION DISTRIBUTION SYSTEM (RIDS)

ACCESSION NBR:9702200157 DOC.DATE: 97/02/11 NOTARIZED: NO DOCKET #
FACIL:50-315 Donald C. Cook Nuclear Power Plant, Unit 1, Indiana M 05000315
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RECIP.NAME RECIPIENT AFFILIATION

SUBJECT: LER 97-001-00:on 970112,while in mode 1 at 99.8% rated thermal power,SI pump discharge cross-tie train "B" shutoff valve was closed to support filing SI sys accumulators. Caused by incorrect sequence of procedure step.W/970211 ltr.

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

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AEP
AMERICAN ELECTRIC POWER

February 11, 1997

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 50.73 entitled Licensee Event Report System, the following report is being submitted:

97-001-00

Sincerely,

A. Alan Blind

A. A. Blind
Site Vice President

/mbd

Attachment

c: A. B. Beach, Region III
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IE22/1

NRC FORM 366 (5-92) U.S. NUCLEAR REGULATORY COMMISSION

APPROVED BY OMB NO. 3150-0104 EXPIRES 5/31/95

ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS INFORMATION COLLECTION REQUEST: 50.0 HRS. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE INFORMATION AND RECORDS MANAGEMENT BRANCH (MNBB 7714), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555-0001, AND TO THE PAPERWORK REDUCTION PROJECT (3150-0104), OFFICE OF MANAGEMENT AND BUDGET, WASHINGTON, DC 20503.

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET NUMBER (2)	
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TITLE (4)
Technical Specification 3.03 Entered On Loss of Four Loop Injection Capability Due to Incorrect Procedural Guidance

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 NAME	DOCKET NUMBER
01	12	97	97	– 001 –	00	02	11	97	FACILITY NAME	DOCKET NUMBER

OPERATING MODE (9)	1
POWER LEVEL (10)	99.8

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

	20.2201(b)		20.2203(a)(3)(i)		50.73(a)(2)(iii)		73.71(b)
	20.2203(a)(1)		20.2203(a)(3)(ii)		50.73(a)(2)(iv)		73.71(c)
	20.2203(a)(2)(i)		20.2203(a)(4)		50.73(a)(2)(v)		OTHER
	20.2203(a)(2)(ii)		50.36(c)(1)		50.73(a)(2)(vii)		(Specify in Abstract below and in Text, NRC Form 366A)
	20.2203(a)(2)(iii)		50.36(c)(2)		50.73(a)(2)(viii)(A)		
	20.2203(a)(2)(iv)		50.73(a)(2)(i)		50.73(a)(2)(viii)(B)		
	20.2203(a)(2)(v)	X	50.73(a)(2)(ii)		50.73(a)(2)(x)		

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (Include Area Code)
Robert Gillespie, Operations Superintendent	616/465-5901, x2535

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)			EXPECTED SUBMISSION DATE (15)	MONTH	DAY	YEAR
YES	X	NO				

ABSTRACT (Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines) (16)

On January 12, 1997, with Unit 1 in Mode 1 at 99.8 percent Rated Thermal Power, the Safety Injection (SI) pump discharge cross-tie train "B" shutoff valve was closed to support filling the SI system accumulators. The East and West Residual Heat Removal (RHR) pump discharge cross-tie valves were already closed at this time. With the unit in this configuration, all four loop injection was lost and Technical Specification 3.0.3 was entered. This condition existed for approximately one minute. The event was terminated when the RHR cross-tie valves were opened per procedure.

The problem was discovered on January 16, 1997 during a subsequent simulator training scenario on four loop injection where it was determined that procedure steps had been incorrectly sequenced. A one hour notification was made in accordance with 10 CFR 50.72(b)(1)(ii)(B), as a condition outside of the design basis, at 1653 hours on the same day.

This event was a result of incorrect procedural guidance. The affected procedures were corrected to direct the proper sequence for operation of the SI pump and RHR pump discharge cross-tie valves. Other actions were taken to ensure that source documentation, performance of 10CFR50.59 reviews and functional reviews are properly performed.

This event was evaluated against the analyses for large break LOCA, small break LOCA, and containment, and found to have no safety significance.

Conditions Prior to Occurrence

Unit 1 in Mode 1 at 99.8 percent Rated Thermal Power.

Description of Event

On January 12, 1997, with Unit 1 at 99.8 percent Rated Thermal Power, during performance of 01-OHP 4021.008.004, "Adjusting the Level of Accumulators", four loop injection capability was momentarily lost due to the simultaneous closure of the Safety Injection (SI) (EIIS-BQ) and Residual Heat Removal (RHR) (EIIS-BO) discharge cross-tie valves. Procedure Step 3.8.1 directed the operators to close at least one SI pump discharge cross-tie valve. The valve selected was 1-IMO-275, SI pump discharge cross-tie train "B" shutoff valve. At this time, the East and West RHR pump discharge valves, 1-IMO-314 and 1-IMO-324, were already closed. The resultant configuration precluded four loop injection capability and placed the unit in Technical Specification 3.0.3. Step 3.8.2 opened both 1-IMO-314 and 1-IMO-324 restoring four loop injection capability. Four loop injection capability was unavailable for approximately one minute.

This condition was discovered on January 16, 1997, by the same Operations shift who had performed the accumulator level adjustment, during a subsequent simulator training scenario on four loop injection. During this simulator training it was determined that the steps in 01-OHP 4021.008.004 had been incorrectly sequenced.

Cause of Event

This event is attributable to incorrect procedure guidance resulting from personnel error on the part of the procedure writer and supporting reviewers.

Analysis of the Event

This event is being reported under 10 CFR 50.72(b)(1) (ii)(B), as a condition outside of the design basis.
With both sets of cross-ties closed, the SI system was in a configuration where four-loop injection capability was not available in the event of a worst-case single failure. In this configuration, each SI and RHR pump would flow to the RCS through its own header only, two-loop injection for each pump. This is acceptable if both trains of ECCS are operating. However, if one train of ECCS were to fail, then flow would only be delivered to two loops, and at a flow consistent with the piping resistance of one header, as the piping resistance of one header causes the operating point of the pump to shift to a lower flowrate. This results in a reduced ECCS flow during injection from a given SI pump than would be available if that pump were open to both headers.

Large Break LOCA (LBLOCA) Analysis

The Unit 1 LBLOCA was analyzed at several different sets of operating conditions. One case was analyzed at 3250 MWt with the RHR cross-tie valves closed and the SI cross-tie valves open. The ECCS configuration that existed during the performance of the accumulator fill procedure had both sets of cross-ties closed for approximately one minute. The LBLOCA was not analyzed for the case of all cross-ties closed. Therefore the ECCS configuration on January 12, 1997 placed the plant in an unanalyzed condition with respect to LBLOCA.

NRC FORM 366A U.S. NUCLEAR REGULATORY COMMISSION

APPROVED BY OMB NO. 3150-0104
EXPIRES 5/31/95

ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS INFORMATION COLLECTION REQUEST: 50.0 HRS. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE INFORMATION AND RECORDS MANAGEMENT BRANCH (MNBB 7714), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555-0001, AND TO THE PAPERWORK REDUCTION PROJECT (3150-0104), OFFICE OF MANAGEMENT AND BUDGET, WASHINGTON, DC 20503.

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

Analysis of the Event (cont'd)

Even though the ECCS configuration was outside of its design limit, there are conservative assumptions in the LBLOCA analysis which would help offset the reduced flow condition associated with the SI cross-ties closed. The LBLOCA analysis assumes that the SI pumps are degraded by 10%. Actual plant test data plotted against the SI pump head curve show that the SI pumps have very little degradation. This is important since the flow rate through one header at higher RCS pressures, early in the LBLOCA transient, is in the portion of the pump head curve where a 10% degradation results in a significant drop-off in delivered flow. The result is that the SI pump could have delivered more flow than the assumptions in the accident analysis would have allowed, particularly earlier in the blowdown before the RHR pumps begin to deliver flow to the RCS. In addition, since the RHR pumps are also assumed to be degraded by 10%, their nominal flow would have also been higher than assumed in the accident analysis, and could have been a significant source of additional water. The RHR pumps cannot begin to deliver flow until the RCS depressurizes below 170 psig, so the extra flow from these pumps will be delayed. Since the peak cladding temperature (PCT) does not occur until approximately 65 seconds after the break, and the RHR pumps begin delivering flow at approximately 24 seconds, they will be at full flow well before the PCT is reached. Finally, our current analysis of record for this case results in a calculated PCT of 2075 °F. Therefore there is some additional margin to the maximum allowable PCT of 2200 °F.

It should also be noted that the Appendix K evaluation models have substantial built-in margin that allow the analysis to envelope all expected operating and plant conditions. In addition, the analysis models postulate a worst case single failure, loss of one ECCS train with a loss of offsite power, that further limits the analysis ECCS flow.

The risk to which Unit 1 was exposed to due to the configuration of the ECCS is very limited due to the short time involved. An informal estimate of the increase in core damage probability for a 15 minute period with one train of ECCS inoperable is approximately 1×10^{-8}. Loss of an entire train of ECCS would easily bound the loss of flow from the SI pumps with the SI cross-ties closed. Therefore, the risk impact to the core for one minute was minimal. In addition, Technical Specification 3.0.3 requires that the plant initiate actions within one hour to place the plant in at least Hot Standby within the next 6 hours if the requirements of an LCO or its Action statement cannot be met. This requirement recognizes the relative risk associated with a condition that violates a Technical Specification, and the rapid shutdown of the plant, by allowing a certain amount of time to initiate orderly actions to shut the plant down. The risk due to design basis accidents inherent in the one hour time allotted to initiate the shutdown of the unit is accepted in the Technical Specifications, and bounds the 1 minute exposure time in which the SI cross-ties were closed.

In conclusion, the configuration of the ECCS system with both cross-ties closed is outside of the LBLOCA analysis assumptions for Unit 1 of the Donald C. Cook nuclear plant. However, the conservative assumptions in the LBLOCA analysis help to offset the reduction in flow that results from the loss of four loop injection capability given a postulated worst-case single failure. In addition, the brief period of time that the plant was in this configuration limits the actual risk to which the plant was exposed to within the window recognized in the Technical Specifications for loss of ECCS capability.

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Analysis of the Event (cont'd)

Small Break LOCA (SBLOCA) Analysis

The analysis for the SBLOCA for Unit 1 includes a case in which the SI cross-tie is closed and the RHR cross-tie is open. A similar case was also analyzed for the main steam safety valve setpoint tolerance relaxation from 1% to 3%. The event described in the condition report stated that both sets of cross-ties were closed for one minute. Since the analysis models the SI cross-tie closed, there is no difference in the ECCS configuration with respect to SI flow. The analysis also modeled the RHR cross-tie open. However, the RHR pumps do not have a significant impact on the analysis results since the peak cladding temperature for all of the cases analyzed occurs at RCS pressures much higher than the RHR pump shutoff head. Therefore, the configuration of the ECCS system with the SI and RHR cross-ties closed does not significantly affect the results of the SBLOCA at 3250 MWt.

Containment Analysis

The long term containment pressure analysis assumes that minimum safeguards is available for core cooling. Other assumptions are that the RHR cross-tie valves are closed, the RHR and SI pumps are degraded by 10%, and the core power is at 3425 MWt. However, the peak containment pressure occurs after the ice bed meltout, which in turn is after the transfer to containment sump recirculation. The SI cross-ties are closed when recirculation is established. Since the SI cross-ties are closed well before the peak containment pressure occurs, their closure earlier has no significant affect on the long term containment pressure analysis. Note that before ice bed meltout, the containment pressure is at or below 8 psig, well below the containment design pressure of 12 psig.

The peak containment temperature analysis is based on a main steam line break inside of containment, and is not affected by this ECCS configuration.

Conclusion

The inadvertent closure of the SI cross-tie valves for approximately one minute placed the plant in an unanalyzed condition with respect to the LBLOCA analysis. However, the conservative assumptions in the LBLOCA analysis help to offset the reduction in flow that results from the loss of four loop injection capability (assuming a worst-case single failure. In addition, the brief period of time that the plant was in this configuration limits the actual risk to which the plant was exposed to within the one hour window recognized in the Technical Specifications (e.g. T/S 3.03) for loss of ECCS capability. The ECCS configuration does not significantly impact the results of the SBLOCA and the containment integrity analysis.

Based on the above evaluations, the event was determined to have no safety significance.

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Corrective Actions

The RHR pump discharge cross-tie valves were opened during the next procedural step.

Procedure 01/02-OHP 4021.008.004 was corrected to direct the proper sequence for operation of the SI pump and RHR pump discharge cross-tie valves. Appropriate administrative actions were taken to ensure that accurate guidance is implemented in the operating procedures. Training was provided to all Operations Department procedure writers on source document requirements, 10CFR50.59 reviews and functional reviews.

The reactor operator requalification program is currently providing training on the proper operation of the safety injection pump and residual heat removal pump discharge cross-tie valves. All operating crews were also provided with a 'Lessons Learned" training document describing barriers that failed to identify this event before it occurred. Operator Aids were placed on the main control boards to provide guidance to the operators on SI pump and RHR pump discharge cross-tie configuration control.

Failed Component Identification

Not Applicable

Previous Similar Events

315/95-004-00