CATEGORY 1

REGULATORY INFORMATION DISTRIBUTION SYSTEM (RIDS)

ACCESSION NBR:9701020056 DOC.DATE: 96/12/23 NOTARIZED: NO DOCKET #
FACIL:50-315 Donald C. Cook Nuclear Power Plant, Unit 1, Indiana M 05000315
AUTH.NAME AUTHOR AFFILIATION
PISARSKY,F. American Electric Power Co., Inc.
BLIND,A.A. American Electric Power Co., Inc.
RECIP.NAME RECIPIENT AFFILIATION

SUBJECT: LER 96-006-01:on 961024,one h action statement requirements were not met for inoperable Unit 1 power operated relief block valve.Caused by incorrect assumption as to which valve factor was applicable.Valve was closed.W/961223 ltr.

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American Electric Power
Cook Nuclear Plant
One Cook Place
Bridgman, MI 49106
616 465 5901

AEP
AMERICAN ELECTRIC POWER

December 23, 1996

United States Nuclear Regulatory Commission
Document Control Desk
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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:

96-006-01

Sincerely,

A. Alan Blind

A. A. Blind
Site Vice President

/mbd

Attachment

c: A. B. Beach, Region III
E. E. Fitzpatrick
P. A. Barrett
S. J. Brewer
J. R. Padgett
D. Hahn
Records Center, INPO
NRC Resident Inspector

IE22 1/1

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)
One Hours Action Statement Requirements Not Met for Inoperable Unit 1 Power Operated Relief Block Valve

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
10	24	96	96	– 006 –	01	12	23	96	FACILITY NAME	DOCKET NUMBER

OPERATING MODE (9): 1

POWER LEVEL (10): 89

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
	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)	X	50.73(a)(2)(i)		50.73(a)(2)(viii)(B)		
	20.2203(a)(2)(v)		50.73(a)(2)(ii)		50.73(a)(2)(x)		

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (Include Area Code)
Frank Pisarsky, Production Engineering - Mechanical Components Supervisor	616/465-5901, x2607

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 October 24, 1996, at 1215 hours with Unit 1 in Mode 1 at 89 percent Rated Thermal Power, 1-NMO-152, the block valve for Power Operated Relief Valve (PORV) 1-NRV-152, was declared inoperable. It was subsequently determined that the valve had been inoperable since October 14, 1996, when the calculation run for 1-NMO-152 using a revised valve factor showed that the thrust required to close the valve under a design basis differential pressure was above the current thrust setting in the closed direction. As the valve was not declared inoperable until October 24, 1996, the 1 hour Action Statement requirements were not met. This LER is being submitted in accordance with 10CFR50.73(a)(2)(i)(B), to report a condition prohibited by the plant's Technical Specifications.

The root cause of this event was the incorrect assumption as to which valve factor was applicable.

The valve was closed and power removed from the actuator. Actuator capability, valve structural limits, and valve thrust setting will be increased before the end of the upcoming Unit 1 refueling outage in early 1997. In addition, a review is being performed of valve factor use to determine if other there are others valves utilizing similarly derived valve factors.

The event was evaluated for both impact on the UFSAR Chapter 14 accident analysis and Probabilistic Safety Assessment, and determined to be of minimal safety significance.

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

Conditions Prior to Event

Unit 1 was in Mode 1, Power Operations, at 89 percent Rated Thermal Power.

Description of Event

On October 24, 1996, at 1215 hours with Unit 1 in Mode 1 at 89 percent Rated Thermal Power, 1-NMO-152, the block valve for Power Operated Relief Valve (PORV) 1-NRV-152, was declared inoperable. It was subsequently determined that the valve had been inoperable since October 14, 1996, when the calculation run for 1-NMO-152 using a revised valve factor showed that the thrust required to close the valve under a design basis differential pressure was above the current thrust setting in the closed direction.

The valve factor (0.4) used in the original calculation for 1-NMO-152, which was derived from EPRI test loop data of a similar valve, had been previously evaluated and accepted by the NRC. As part of the final evaluation the EPRI Performance Prediction (PPM) computer model was to be run to determine a bounding valve factor. This run indicated a more conservative value of 0.51 should be used. However, this valve factor of 0.51 was not applied at the time the information was received.

On October 23, 1996, during the NRC Generic Letter (GL) 89-10 closure inspection, the NRC questioned the use of the 0.4 valve factor in lieu of the EPRI PPM valve factor of 0.51. The NRC questioned the similarities of the EPRI valve to the Cook Plant valve and stated that there were enough differences between the valves that the test data was not entirely applicable to the Cook valves, and that the EPRI PPM valve factor should be considered best available data, and used as the basis for operability. After discussions with Engineering, it was determined that the valve factor should be applied. The 0.51 valve factor was subsequently applied to the thrust calculation and the valve was declared inoperable.

Cause of the Event

The root cause of the event was the incorrect assumption as to which valve factor was to be applied.

Analysis of Event

In accordance with 10CFR50.73(a)(2)(i)(B), this event is being reported as a condition prohibited by the plant's Technical Specifications. Information impacting the operability of the valve was available 10 days prior to the valve being declared inoperable, thereby exceed the 1 hour time limit set forth by the Technical Specifications.

The single Chapter 14 accident analysis which requires PORV operability is the Steam Generator Tube Rupture (SGTR) with a loss of offsite power. In this event, 1 PORV is required to reduce primary system pressure during the recovery phase of the accident to slightly above the secondary side pressure. A second PORV is required to be operable for single failure considerations. In the 10 day period for which the PORV block valve was considered inoperable, but in the fully open position, the associated PORV was available to perform this safety function. A review of the valve factor used for 1-NMO-152 showed that the torque switch setting would allow operation below an RCS pressure of 1500 psig. Since an operator would have depressurized the RCS to the secondary side pressure as part

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

Analysis (cont'd)

of the SGTR recovery, the block valve would have been operable at the reduced RCS pressure. In the event that the PORV that was used to depressurize failed to reclose, the operator could have successfully closed the associated block valve. The inoperability of 1-NMO-152 would not have prevented the associated PORV from performing its safety function for the SGTR accident or prevented the block valve from closing after the RCS pressure was reduced. Therefore there is no impact on the safety consequences of this accident resulting from 1-NMO-152 remaining open while inoperable.

Technical Specification 3.4.9 requires either a) 2 PORVs or b) 1 PORV and the RHR safety valve to be operable with appropriate lift settings for the Low Temperature Overpressure Protection (LTOP) system. One PORV (or RHR safety valve) provides adequate overpressure protection for this system. The second valve is required for single failure concerns. The 2 PORVs that are part of the LTOP system are 1-NRV-152 and 1-NRV-153 with associated block valves 1-NMO-152 and 1-NMO-153. Since 1-NMO-152 was still open during the period in question, its associated PORV was available to function for the LTOP system. In addition, the LTOP can also use 1 PORV and the RHR safety valve to provide overpressure protection. Thus the LTOP safety function was still provided.

The main safety function of the PORV block valve is to isolate a PORV if the PORV fails in the open position. With 1-NMO-152 inoperable and fully open, this safety function was compromised. A small break LOCA (SBLOCA) could have occurred if the associated PORV was opened and then failed to reclose when system pressure was reduced. This is roughly the equivalent of a 3 inch hot leg break. The SBLOCA analysis for Cook shows that the 3 inch cold leg break is the most limiting break. Since the hot leg break is less severe than a cold leg break, the consequences resulting from the postulated failed open PORV are bounded by the existing SBLOCA analysis. In addition, as previously noted, 1-NMO-152 could function below an RCS pressure of 1500 psig. Plant operating procedures instruct the operator to close the PORV and the block valve if the PORV will not close. Therefore the postulated SBLOCA could have been manually terminated after RCS pressure dropped below 1500 psig.

The containment integrity analysis shows that the peak containment pressure results from a double ended guillotine break of the RCS pump suction piping, and the peak containment temperatures result from a Main Steam Line rupture. The consequences of the postulated failed open PORV are easily bounded by the consequences of these other scenarios.

An evaluation of the effect of operating with an inoperable PORV block valve in the fully open position was evaluated. The increase in the SBLOCA frequency results in an overall annual risk increase of 1.39E-07 per year. This value is within the 1E-06 Nuclear Energy Institute guidelines for acceptable risk increase, therefore, the impact of this postulated scenario on the Cook PSA is minimal.

In conclusion, an inoperable PORV block valve in the fully open position would not be capable of performing the safety function of isolating a failed open PORV until RCS pressure was reduced to less than 1500 psig. This scenario would require a pressure transient that would challenge the PORV setpoint or a spurious operation of the PORV, followed by the failure of the PORV to reclose on demand. This is roughly equivalent to a SBLOCA. The accident analysis performed for the SBLOCA demonstrates that the limiting SBLOCA is a 3 inch cold leg break, for which the results are more severe than the postulated failed open PORV. Therefore, the safety consequences of operating with an inoperable PORV block valve in the open position are bounded by the SBLOCA analysis, and the risk significance of operation in this condition during the time period in question has minimal impact.

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

Corrective Action

The valve was closed and power removed from the actuator. An analysis has been performed to allow use of this valve under 1500 psid for LTOP service. Actuator capability and the valve structural limits will be modified, and the valve thrust setting will be increased by the end of the upcoming Unit 1 refueling outage in early 1997. These actions will provide sufficient capability so that the actuator can be set to accommodate a valve factor of 0.51 and be in full compliance with Generic Letter 889-10 closure criteria. The other 5 NMOs - 2 in Unit 1 and 3 in Unit 2 - were verified to be operable with the use of the 0.51 valve factor.

To ensure that additional instances of incorrect assumptions have not occurred, a review of the position paper on valve factors was performed. The review noted one additional valve model where the valve factor is based solely on EPRI testing of similar valves. The remaining GL 89-10 valve factors are based on more than one type of test information , i.e., differential pressure testing at other nuclear plants, testing at Cook, EPRI test a data. Additional basis for the valve factor of the other valve model will be developed either by testing under design basis conditions or application of the EPRI PPM.

Failed Component Identification

None

Similar Event

None