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Operations

Entergy Operations, Inc.

December 27, 1990

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U. S. Nuclear Regulatory Commission
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SUBJECT: Arkansas Nuclear One - Unit 1
Docket No. 50-313
License No. DPR-51
Licensee Event Report 50-313/90-016-00

Gentlemen:

In accordance with 10CFR50.73(a)(2)(i), attached is the subject report concerning Reactor Protection System and Emergency Feedwater Initiation and Control System setpoints which were not calibrated in accordance with Technical Specifications requirements due to personnel error.

Very truly yours,

for
James J. Fisicaro
Manager, Licensing

JJF/RHS/sgw
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NRC Form 366
(6-89)

U.S. Nuclear Regulatory Commission
Approved OMB No. 3150-0104
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LICENSEE EVENT REPORT (LER)

FACILITY NAME (1) Arkansas Nuclear One, Unit One	DOCKET NUMBER (2)	PAGE (3)
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TITLE (4) Reactor Protection System and Emergency Feedwater Initiation and Control System Setpoints Not Calibrated in Accordance with Technical Specifications Requirements Due to Personnel Error

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 5 0 0 0
1 1	2 7	9 0	9 0	0 1 6	0 0	1 2	2 7	9 0		0 5 0 0 0

OPERATING MODE (9)	N	THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more of the following) (11)			
POWER LEVEL (10)	0 0 0	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)	X 50.73(a)(2)(i)	50.73(a)(2)(viii)(A)	
		20.405(a)(1)(iv)	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	
Richard H. Scheide, Nuclear Safety and Licensing Specialist	5 0 1	9 6 4 - 5 0 0 0

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

SUPPLEMENT REPORT EXPECTED (14)	EXPECTED SUBMISSION DATE (15)	Month	Day	Year
Yes (If yes, complete Expected Submission Date) X No				

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

On November 27, 1990, it was identified that the tolerances stated in the Reactor Protection System Channel Test procedures for the Reactor Power/Power Imbalance/Flow bistables would allow the bistables to be calibrated to trip at a power level slightly greater than allowed by the Technical Specifications (TS). On November 29, it was identified that procedural tolerances would allow the bistables which enable the automatic actuation of the Emergency Feedwater System (EFW) on loss of 4 reactor coolant pumps to be calibrated to actuate at a power level slightly greater than allowed by the TS (10%). Additionally, it was identified that procedural tolerances would allow the bistables which enable the low steam generator pressure actuation of EFW to be calibrated to actuate at a pressure greater than allowed by the TS (750 psig). The root cause of this event was determined to be personnel errors resulting in the procedure deficiencies. The affected calibration procedures were revised to ensure compliance with TS. A program is being developed to document the basis for existing setpoints and to provide procedural control and documentation of setpoint determinations and changes.

A. Plant Status

At the time the conditions discussed in this report were identified, Arkansas Nuclear One, Unit 1 (ANO-1) was in the Cold Shutdown condition. Refueling outage 1R9 was in progress.

B. Event Description

Between November 27 and November 29, 1990, during the performance of a Technical Specifications surveillance procedure review, three deficiencies were identified in which procedural setpoint tolerances would have allowed the bistables associated with the Reactor Protection System and Emergency Feedwater Initiation and Control System to be calibrated to actuate at greater than their Technical Specifications stated limits. Technical Specification Figure 2.3-2 establishes the maximum allowable setpoints for reactor thermal power by considering reactor power imbalance (power in the upper half of the core minus power in the lower half of the core) as compared to the various acceptable reactor coolant pump operating configurations. The Reactor Power/Power Imbalance/Flow bistables are calibrated to initiate a reactor trip prior to exceeding the limits established by this figure in order to ensure that the integrity of the fuel cladding is maintained. On November 27, 1990, it was identified that the setpoints specified in the Reactor Protection System (RPS) [JC] Channel Test procedures for the Reactor Power/Power Imbalance/Flow bistables were the actual limits as indicated by Figure 2.3-2. The allowable procedural tolerance for this setpoint made it possible for the bistables to be left at a trip setpoint slightly greater than allowed by the Technical Specifications.

Technical Specifications require that the automatic actuation of the Emergency Feedwater System (EFW) on loss of 4 reactor coolant pumps be operable above 10 percent reactor power. Below 10 percent power, this actuation signal may be bypassed. On November 29, 1990, it was identified that procedural tolerances would allow the bistables which automatically enable the automatic actuation of EFW on the loss of 4 reactor coolant pumps to be calibrated to actuate at slightly greater than 10 percent reactor power.

The EFW system is designed to be automatically actuated when steam generator (SG) pressure decreases to 600 psig. During plant cooldowns and depressurization when SG pressure is below 750 psig, this initiate signal may be manually bypassed. However, Technical Specifications requires that the bypass be automatically removed when SG pressure is greater than 750 psig. On November 29, 1990, it was identified that procedural tolerances would allow the bistables which automatically enable the Low SG Pressure Initiate to be calibrated to actuate at slightly greater than 750 psig.

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

C. Root Cause

The root cause of this event was determined to be personnel errors resulting in the procedure deficiencies. Personnel responsible for development of the procedures failed to recognize that the tolerances allowed by the procedures could result in setpoints which did not satisfy the TS requirements.

D. Corrective Actions

The affected calibration procedures were revised to ensure that the setpoints discussed in this report remain within the limits established by the Technical Specifications.

The review which identified the deficiencies discussed in this report was initiated in October, 1990 as a result of the discovery of a similar condition (LER 50-368/90-021-00) in an ANO-2 procedure. The review looked at procedures which adjust and verify Technical Specifications setpoints for the ANO-2 Plant Protection System [JC] and the ANO-1 RPS, Engineered Safeguards Actuation System [JE], and Emergency Feedwater Initiation and Control System to ensure that they complied with the applicable Technical Specifications limits. This review has been completed and no other significant deficiencies were identified.

A Plant Setpoint Control Program is being developed by ANO. This program will document the basis for existing setpoints for selected components as well as provide procedural control and documentation of setpoint determinations and changes. This program, which is an ANO Business Plan item, is expected to be fully implemented by June 5, 1992, and should prevent the occurrence of similar events.

E. Safety Significance

Although the RPS Channel Test procedures allowed the Reactor Power/Power Imbalance/Flow bistables to be calibrated at slightly greater than the Technical Specifications limit, this condition would not have resulted in the setpoint exceeding any of the analytical limits which were used in the safety analysis calculations.

The setpoints for the EFW initiate bypass resets discussed in this report are somewhat arbitrary in nature since no analytical limits were established for them in the safety analysis. In addition, the amount these setpoints were allowed by procedure to exceed the Technical Specifications stated limits was negligible and of minimal consequence.

Considering the above, ANO believes that the conditions discussed in this report are of minimal safety significance.

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

F. Basis for Reportability

Since plant procedures allowed the instruments discussed in this report to be calibrated to actuate at values greater than the Technical Specifications allowed, these conditions are considered reportable pursuant to 10CFR50.73(a)(2)(i)(B) as operation prohibited by the plant's Technical Specifications.

G. Additional Information

Previous similar events in which procedures allowed setpoints to be calibrated above the limits established by Technical Specifications were reported in LERs 50-368/86-016-00 and 50-368/90-021-00. The deficiencies discussed in this report were identified as a result of actions initiated in response to the identification of the conditions reported in LER 50-368/90-021-00.

Energy Industry Identification System (EIIS) codes are identified in the text as [XX].