LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	Arkansas Nuclear One, Unit One
DOCKET NUMBER (2)	0|5|0|0|0| 3| 1| 3
PAGE (3)	1 OF 0|5
TITLE (4)	Inadequate 10CFR50.59 Design Change Review Resulting in a Design Deficiency In Emergency Feedwater System

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	1|5	8|6	8|6	0|0|2	0|0	0|2	1|7	8|6		0|5|0|0|0|
										0|5|0|0|0|

OPERATING MODE (9): 1

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

POWER LEVEL (10): 0\|9\|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)	___ 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)
Dwight J. Johnson, Plant Licensing Engineer	5|0|1|9|6|4|-|3|1|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
X Yes (If yes, complete Expected Submission Date)	___ No		0|6	2|7	8|6

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

On 1/14/86 at 1400 hours, a Nuclear Regulatory Commission Safety System Functional Inspection team notified plant staff members of a potential design deficiency with the Emergency Feedwater System (EFW) design. The unit was operating at 89.5% power. The plant staff reviewed the concern and concluded on 1/15/86 at 1645 hours that there was a design deficiency in the EFW System. At this time the plant was proceeding to cold shutdown for an unrelated maintenance activity. The design error involved the deletion of two check valves from the original design requirements for the steam supplies to the steam turbine driven emergency feedwater pump (P-7A). These valves were to serve as steam supply line isolation components during a Main Stream Line Break upstream of the Main Steam Block Valves with a concurrent loss of red channel AC power. These valves were omitted from the final design because of reliability and safety concerns with similar valves on Unit 2 at this site. Failure to identify the error in the final EFW design change was the result of inadequate design review in accordance with 10CFR50.59. The discrepancy in design caused the unit to be in an unanalyzed condition and operation could have occurred outside the established design basis for the plant. While the unit was in cold shutdown, the check valves were installed. Management review of this event is continuing and additional information will be included in a scheduled supplemental report. The unit was returned to power operation upon completion of the maintenance activities and modification of the EFW system to include the check valves. This occurred on 1/31/86 and testing of the EFW system was completed on 2/2/86. A similar design change deficiency was reported in LER 50-313/85-001.

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NRC Form 366A
(9-83)

Form 1062.01B
U.S. Nuclear Regulatory Commission
Approved OMB No. 3150-0104
Expires: 8/31/85

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

I. Description of Event

A. Unit Status

The unit was operating at 89% power, with reactor coolant average temperature 579°F and reactor coolant system pressure 2155 psia. The unit was power limited to 89% due to high level in the 'A' Once Through Steam Generator (OTSG).

B. Component Identifier

Emergency Feedwater (EFW) System: System Identifier = JB/JE.

This event involved a failure of an administrative process for the review of station design changes of the EFW system per the requirements of 10CFR50.59.

C. Sequence of Events

During a Nuclear Regulatory Commission (NRC) Safety System Functional Inspection (SSFI) on 1/14/86, at 1400 hours, it was communicated to the Arkansas Nuclear One (ANO) plant staff that a possible design deficiency in the EFW system had been discovered by the NRC inspection team. An engineering evaluation was initiated by ANO plant staff and Arkansas Power and Light (AP&L) personnel, and on 1/15/86 at 1645 hours the engineering evaluation concluded that for a very specific sequence of events, the EFW system would not meet single failure criteria. Based on the results of this evaluation, the NRC was immediately notified of the findings on 1/15/86 at 1734 hours via the Emergency Notification System in accordance with the provisions of 10CFR50.72 (b)(1)(ii). During the time of this evaluation, the unit was proceeding to cold shutdown for an unrelated maintenance activity. The unit was shut down 1/15/86. The unit was subsequently returned to power operations after completing the necessary maintenance activities and corrective modifications to the EFW system on 1/21/86. Testing of the modified EFW system was completed on 2/2/86.

II. Event Cause

A. Event Analysis

Arkansas Power and Light performed a review of the emergency feedwater system (EFW) in 1980 per the provision of NUREG 0737 II.E.1.1 and II.E.1.2 to improve the reliability of the EFW system and upgrade the system where necessary to ensure safety grade automatic initiation and flow indication. It was determined that upgrades to the installed EFW system would be required based on this review.

Part of those design changes proposed included modifications to the steam admission piping and actuation systems from each OTSG to the steam driven emergency feedwater pump (P-7A). Automatically actuated DC powered steam admission valves were installed in the steam supply line to P-7A. Previously installed AC powered steam admission valves in the steam supply lines from each OTSG were modified to receive closure signals from the Emergency Feedwater Initiation and Control system (EFIC) logic network for OTSG isolation if decreasing pressure is sensed for the respective OTSG. With the addition of the new DC powered steam admission valves which would be maintained normally closed, the AC powered steam admission valves were changed from the pre-design configuration of normally closed to normally open, thereby now requiring an isolation actuation function.

The OTSG steam supply line isolation valves receive AC power from the emergency safety features electrical busses. To ensure redundancy and channel separation, one valve is supplied with red channel AC actuation power and the other valve with green channel AC actuation power. The proposed design originally also included a check valve downstream of each motor operated isolation valve in the steam supply lines (see attached figure). These features would have ensured that on a Main Steam Line Break (MSLB) event upstream of the Main Steam Block Valves (MSBVs), there would be no cross-connecting of the unaffected OTSG to the break, and an intact steam supply via the unaffected OTSG would remain available to P-7A.

During the design change review, concerns were raised regarding the reliability of the proposed check valves. The proposed valve arrangement and check valve type were similar to that utilized in the ANO-2 emergency feedwater system original design. ANO-2 had experienced several reliability problems with the check valves (reference LER 50-368/82-031-00 and 82-031-01) and the AP&L design staff determined it prudent to exclude these check valves from the final design, thus eliminating a potential source of future operational and maintenance problems and safety concerns.

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

During the design review conducted by the NRC SSFI team at ANO the week of 1/31/86, the inspection team determined the installed design failed to meet single failure criteria for a specific postulated design based event. The postulated scenario was as follows:

Given a MSLB on the 'A' OTSG upstream of the MSBVs and a concurrent loss of red channel AC power,

(1) the motor driven EFW pump (P-7B) would be unavailable since it is powered from the red channel AC electrical bus,

(2) the 'A' OTSG automatically actuated isolation valve (CV-2667) would fail to close since it is also powered from the red channel AC electrical bus,

(3) the 'A' and 'B' OTSGs would remain cross-connected through the steam supply lines to P-7A, and

(4) due to steam cross-feed to the break, sufficient steam flow would probably not be available to supply P-7A from the intact 'B' OTSG.

B. Root Cause

The root cause of this event was an inadequate review of the EFW design change under the requirements set forth in 10CFR50.59 and represents a breakdown in the administrative controls established over design changes to the facility.

C. Basis for Reportability

This report is being submitted to meet the requirements of 10CFR50.73 (a)(2)(ii) since the design deficiency, as noted, represented a safety system configuration where the plant was in an unanalyzed condition and operation could have occurred under the postulated scenario outside the design basis. It is the opinion of the plant staff that operator action could be relied upon to mitigate the consequences of such an event, and that the scenario proposed had a very limited probability of occurrence. Therefore, this event represented a minimal potential for degradation to the health and safety of the general public.

III. Corrective Actions

A. Immediate

A plant shutdown for an unrelated maintenance activity was underway when the 10CFR50.72 notification was made for this event. The plant was placed in cold shutdown which negated the requirements for EFW system operability.

B. Subsequent

An engineering evaluation of this event conducted by AP&L personnel indicated that the original concerns with the reliability of the check valves that were omitted from the proposed EFW modification may be mitigated by establishing routine preventive maintenance schedules. Therefore, it was determined to install the check valves in the Unit 1 steam supplies to P-7A during the recent maintenance outage, and initiate an augmented preventive maintenance program.

C. Future

Plans for future action are currently pending AP&L management review. The facility design change process and the 10CFR50.59 review process are under review at this time. Upon completion of AP&L management review of this occurrence, a supplemental report will be submitted.

IV. Additional Information

A. Supplemental

An analysis was subsequently performed to determine the adequacy of available steam supply to P-7A following the postulated event of MSLB with concurrent loss of red channel AC power. Although the event would have allowed the unaffected OTSG to lose steam through the cross-connection, the calculations showed the resulting steam supply to P-7A would have been sufficient to provide emergency feedwater to the unaffected OTSG for decay heat removal. Sufficient feedwater for decay heat removal could have been provided for a period in excess of 48 hours.

B. Similar Events

A similar report that detailed deficiencies in the facility design change process was submitted as LER 50-313/85-001. A supplemental report on the noted LER will also be pending AP&L management review and actions taken in regard to the event outlined in this report.

NRC Form 366A
(9-83)

Form 1062.01B
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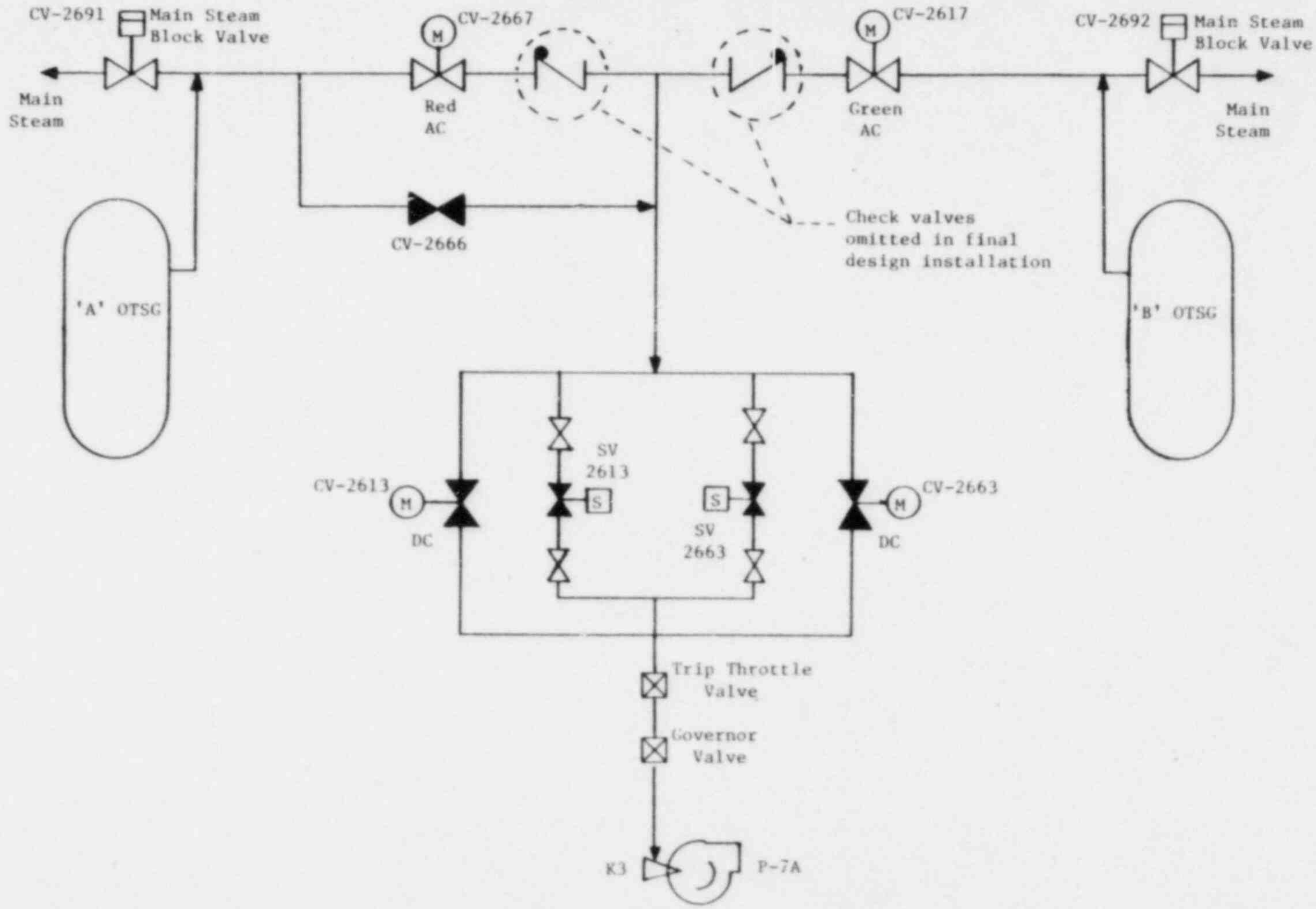
TEXT (If more space is required, use additional NRC Form 366A's) (17)

MAIN STEAM SUPPLY to TURBINE DRIVEN
EMERGENCY FEEDWATER PUMP P-7A



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February 17, 1986

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U. S. Nuclear Regulatory Commission
Document Control Desk
Washington, D.C. 20555

Subject: Arkansas Nuclear One - Unit 1
Docket No. 50-313
License No. DPR-51
Licensee Event Report
No. 86-002-00

Gentlemen:

In accordance with 10CFR50.73(a)(2)(ii), enclosed is the subject report concerning an inadequate 10CFR50.59 Design Change Review resulting in a design deficiency in the Emergency Feedwater System.

Very truly yours,

for J. Ted Enos
Manager, Licensing

JTE:RJS:lw

Enclosure

cc: Mr. James M. Taylor, Director
Office of Inspection and Enforcement
U. S. Nuclear Regulatory Commission
Washington, DC 20555

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