ARKANSAS POWER & LIGHT COMPANY

June 16, 1989

1CANØ68911

U. S. Nuclear Regulatory Commission
Document Control Desk
Mail Station P1-137
Washington, D. C. 20555

SUBJECT: Arkansas Nuclear One - Unit 1
Docket No. 50-313
License No. DPR-51
Licensee Event Report No. 50-313/89-022-00

Gentlemen:

In accordance with 10CFR50.73(a)(2)(i)(B), attached is the subject report concerning a design error which resulted in an inoperable emergency feedwater system piping support due to the potential for overload during a seismic event.

Very truly yours,

E. C. Ewing
General Manager,
Plant Support

ECE:JDJ:sgw
attachment

cc w/att: Regional Administrator
Region IV
U. S. Nuclear Regulatory Commission
611 Ryan Plaza Drive, Suite 1000
Arlington, TX 76011

INPO Records Center
1500 Circle 75 Parkway
Atlanta, GA 30339-3064

IE22
1/1

Form 1062.01A

NRC Form 366
(9-83)

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

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 3

TITLE (4) Design Error Results in an Inoperable Emergency Feedwater System Piping Support due to Potential for Overload during Seismic Event

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 3	1 0	8 9	8 9	-- 0 2	-- 0 0	0 6	1 6	8 9		0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9): N

POWER LEVEL (10): 0 0 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)	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)
Julie D. Jacks, Nuclear Safety and Licensing Specialist	5 0 1 9 6 4 - 3 1 1 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
B	B A	S P T		N					

SUPPLEMENT 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)

As a followup to an event of 01/20/89 (reference LER 50-313/89-004-00) when High Pressure Injection system piping was exposed to temperatures higher than design temperatures, other safety-related systems were being reviewed to compare design temperatures to possible operating temperatures. As part of this review, in March 1989 a rigid support on the Emergency Feedwater (EFW) system piping to 'B' Once-Through Steam Generator (OTSG) was identified as being potentially insufficient for the load which might be experienced during a seismic event. Based on an engineering evaluation completed in May 1989, the piping was conservatively judged to be inoperable, although no specific analysis was performed which demonstrated that the piping could fail if the support failed. This judgement was based on the load the support could be required to bear and on the fact that failure of this support would leave an unsupported piping span of approximately 38 feet. The affected piping is located inside the Reactor Building. The affected support was replaced during the maintenance outage in which the condition was identified. The cause of the deficient support was apparently the support designer's failure to use appropriate design loads during the initial design phase. Current design modification procedures should prevent recurrence of this type of event.

A. Plant Status

On March 10, 1989, when this condition was discovered, Arkansas Nuclear One, Unit One (ANO-1) was at Cold Shutdown. Reactor Coolant System (RCS) [AB] temperature was 107 degrees Fahrenheit and RCS pressure was atmospheric pressure.

B. Event Description

On January 20, 1989, an event occurred at ANO-1 in which RCS backleakage through a failed-open check valve resulted in the design temperature of the High Pressure Injection System piping being exceeded. (The event was reported in Licensee Event Report 50-313/89-004-00.) As a followup to this event, other safety-related systems were reviewed to determine if the potential existed under normal or emergency conditions for the piping to be exposed to temperatures higher than the design temperatures. A review of existing analyses for the Emergency Feedwater System (EFW) [BA] piping inside containment to Once-Through Steam Generator (OTSG) [SG] E24B showed that the piping was analyzed for temperatures of 70 to 90 degrees, but the piping could actually be exposed to temperatures of 40 to 148 degrees. In March 1989, a new thermal analysis was prepared for the affected EFW piping to ensure the piping and supports were qualified for this temperature range. Although the operability of the piping was subsequently determined not to have been affected by the change in the range of design temperatures, during the course of the analysis a rigid support (EFW-9) [BA-SPT] was identified as being potentially insufficient to support the piping given the loads expected as the result of a postulated seismic event. An engineering evaluation completed in May 1989 concluded that the as-found condition of EFW-9 would not be within pipe support operability criteria.

The seismic load on support EFW-9 was predicted to be approximately 1800 pounds. Failure of this support during a seismic event would have resulted in an unsupported EFW piping span of approximately 38 feet between rigid supports. Although a detailed analysis of the EFW piping assuming a failure of EFW-9 was not performed, the EFW piping to the 'B' OTSG was judged to have not been within piping code operability criteria based on the 1800 pound design load for EFW-9 and the distance between the supports adjacent to EFW-9.

Evaluations were performed on inside containment EFW piping, pipe supports, nozzles and containment penetrations included in the new thermal stress models for the EFW system supply to 'B' OTSG as well as 'A' OTSG. No other design deficiencies were found which impacted system operability.

C. Safety Significance

A failure of this EFW line during a seismic event could result in a loss of EFW flow to the 'B' OTSG. However, EFW to the 'A' OTSG would still be available, and the Emergency Feedwater Initiation and Control (EFIC) System would isolate 'B' OTSG when its pressure dropped below 600 psig. This event would be comparable to other design basis scenarios involving a loss of an OTSG, such as a main steam line break.

D. Root Cause

The root cause of this event was apparently the failure of the support designer to use the correct load when the support was designed during the initial construction of the plant. This assumption is based on the size of the load as determined in the recent stress analysis (1800 pounds) and the relatively small support structure for EFW-9 that was installed. The design load actually used for the original support qualification could not be determined as the original design support calculation for EFW-9 could not be located. The reason for this discrepancy could not be determined.

E. Basis for Reportability

Technical Specification 3.4.1.4 states that the reactor shall not be heated above 280 degrees unless both EFW pumps and their flow paths are operable. Since the system design of the EFW flow path to 'B' OTSG was not sufficient to ensure system operability following a seismic event, this flow path has been technically inoperable during prior plant startups and operation. Therefore, this event is reported in accordance with 10CFR50.73(a)(2)(i)(B) as operation in a condition prohibited by Technical Specifications.

The engineering determination of the inoperability of the piping in its as-found condition was completed May 11, 1989. However, this information was not received by Licensing until insufficient time remained to prepare the Licensee Event Report by June 10, 1989. This was discussed with the NRC Region IV staff and a submittal date of June 16, 1989, was agreed upon.

F. Corrective Actions

A Design Change Package completed March 21, 1989, installed a new support to replace the existing EFW-9 support. This work was completed prior to heatup from the maintenance outage during which the condition was discovered. Current design modifications procedures should prevent a recurrence of this type of event.

G. Additional Information

A similar event in which a design error resulted in the installation of an undersized snubber on the RCS pressurizer surge line is discussed in Licensee Event Report 50-313/88-015-01.

During refueling outage 1R7, a review of the seismic calculations for the EFW system revealed that three seismic supports were not installed during initial plant construction as required. This event was documented in LER 50-313/87-001-00.

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