ARKANSAS POWER & LIGHT COMPANY

June 9, 1989

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SUBJECT: Arkansas Nuclear One - Unit 1
Docket No. 50-313
License No. DPR-51
Licensee Event Report No. 50-313/88-026-00

Gentlemen:

In accordance with 10CFR50.73(a)(2)(i)(B), (a)(2)(ii), and (a)(2)(v), enclosed is the subject report concerning the Decay Heat Removal/Low Pressure Injection System and the Core Flood System satisfaction of seismic design criteria.

Very truly yours,

E. C. Ewing
General Manager,
Plant Support

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Form 1062.01A

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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) 0 5 0 0 0 3 1 | PAGE (3) 1 OF 0 4

TITLE (4) Decay Heat Removal/Low Pressure Injection System and Core Flood System Satisfaction of Seismic Design Criteria

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 7	0 6	8 8	8 8	0 2	0 0	0 6	0 9	8 9		0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9): N

POWER LEVEL (10): 0 8 5

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)	X 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)	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)
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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)

__ 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)

In performing design reviews in conjunction with a design change associated with the MOVATs project and the reanalysis resulting from the HPI backflow event, certain discrepancies were identified in the design analyses and assumptions related to the Decay Heat Removal (DHR)/Low pressure Injection (LPI) and Core Flood (CF) systems. No specific analyses indicated an inability of these systems to perform their intended functions. AP&L conservatively assumed that these conditions together could have impacted the operability of these systems. These conditions involved spring can travel in the DHR suction piping and an installed hanger between the Reactor Coolant System (RCS) hot leg and the containment penetration. Discrepancies were also found in temperatures used for DHR/LPI system analyses and the actual versus analyzed weights of the two Core Flood (CF) tank outlet valves and two DHR/LPI check valves inside containment. Modifications (primarily involving piping supports) to address these discrepancies have been completed. Reviews of other systems have been conducted to verify the accuracy of qualifying analyses.

A. Plant Status

This event involves discrepancies discovered in the qualifying analyses for the Decay Heat Removal/Low Pressure Injection (DHR/LPI) [BP] System at Arkansas Nuclear One, Unit One (ANO-1). These discrepancies were discovered over a period of time, beginning July 6, 1988, when ANO-1 was operating at 85 percent of full power, with subsequent discoveries occurring in February and March 1989 while the unit was in cold shutdown. Also, a piping hanger discrepancy was found in September 1988 during refueling outage 1R8.

B. Event Description

In July 1988, piping calculations were being reviewed in support of a Design Change Package (DCP) for the motor-operated valve actuator tests (MOVATs) project. This review revealed a discrepancy in the originally analyzed temperature for the DHR suction piping inside containment. The affected section of piping was the DHR suction piping between the Reactor Coolant System (RCS) [AB] loop 'A' hot leg and the first DHR isolation valve, CV-1050 [BP-ISV]. The piping was found to be analyzed to a temperature of 518 degrees Fahrenheit; the maximum temperature to which the piping could be exposed was postulated to be 643 degrees. This particular condition was subsequently determined not to affect operability of the piping. Additional discrepancies were found involving spring can travel designs for seismic displacements. A spring can hanger for the suction piping was also found not to be installed as designed. A visual inspection of the hanger's as-found condition did not reveal any damage, and the hanger appeared to be withstanding its design load. Subsequent analysis also revealed discrepancies in analyzed valve weights for two valves (CV-1410 and CV-1050) in the DHR system. Reanalyses using corrected weights (and incorporating a recent support modification to address the DHR discrepancy discussed above) showed the system to remain within code.

In January 1989, a plant trip occurred followed by RCS backleakage into the High Pressure Injection (HPI) System through a failed check valve. (The reactor trip is discussed in Licensee Event Report 50-313/89-002-00; the HPI backleakage event in Licensee Event Report 50-313/89-004-00.) As part of the followup to the backleakage event, other safety-related systems were being reviewed for any potential exposures to temperatures for which the piping was not qualified. In February 1989, a review of piping analyses for the inlet and outlet piping of the DHR/LPI pumps revealed that the pump inlet piping was qualified for 280 degrees, but the pump outlet piping was qualified only for 250 degrees. Additionally, the DHR/LPI injection lines inside containment were found to have been analyzed only to 208 degrees. As a result of reanalyses, one piping support (outside containment) required modifications, based on an assumed design temperature of 300 degrees, consistent with existing operating criteria.

That reanalysis also revealed discrepancies in the as-analyzed weights of the Core Flood Tank (CFT) outlet valves [ISV] (CV-2415 and CV-2419) and two adjacent DHR/LPI check valves [BP-V] (DH-13A and DH-13B). The valve weights were less than the actual valve weights. (The DHR/LPI check valves are located just upstream of the DHR/LPI connection with the CFT outlet line.) Although the piping itself was determined to meet code allowables in the new thermal reanalysis, five piping supports inside containment required modifications due to high thermal loads to ensure that design acceptance limits for the piping supports were met.

In April 1989, an operability assessment of the DHR/LPI system was completed which considered the temperature analyses discrepancies for the DHR/LPI piping and the valve weight discrepancies. Although no specific analysis indicated the system would not have performed its functions, based on the fact that support modifications were required to meet operability limits, the DHR/LPI system was conservatively considered to have been inoperable prior to completion of the support modifications.

C. Safety Significance

These conditions are considered to have been potentially significant because seismic analyses did not address the operability of the DHR/LPI system and the Core Flood System given the worst possible scenario of failures consistent with the conditions noted above. This worst-case scenario could also include a loss of reactor coolant from the ruptured decay heat suction line, which connects to the RCS loop 'A' hot leg. (Even in this extreme case, the High Pressure Injection System would be available to provide emergency core cooling, and long-term heat removal could be accomplished by reflux boiling using the loop 'B' steam generator (reactor coolant steam being condensed in the steam generator tubes). Decay Heat piping outside containment would still have been isolated from the RCS and any loss of coolant due to a pipe rupture would have been contained in the reactor building.)

Note that the assumption of total loss of the DHR/LPI and CF systems are hypothetical results involving worst-case scenarios. System redundancies coupled with conservative code allowables are such that the probability of a total loss of the DHR/LPI and CF systems is very slim. Also, stress in excess of code allowables does not necessarily imply a postulated system failure (some "operability" determinations were based only on such conditions); code limits are conservative enough that the worst probable result could have been yielding of pipe or supports instead of a complete rupture of the piping. However, the existing conditions increased the potential for a loss of reactor coolant and degradation of the DHR/LPI and CF systems as a result of a seismic event.

D. Root Cause

These conditions appear to have been oversights by the various design groups involved in these systems' designs. Also, a subsequent design change did not appropriately consider certain temperatures in the DHR/LPI piping because of an apparent oversight between the system design and the piping design.

E. Reportability

Based on conservative assumptions regarding operability, the potential loss of the DHR/LPI and CF systems during a seismic event has been considered to be a condition outside the design basis of the plant and is therefore reported under 10CFR50.73(a)(2)(ii). As this assumed loss would have affected both trains of DHR/LPI and both CF tanks, this is considered an event or condition that alone could have prevented the fulfillment of the safety function of systems that are needed to remove residual heat or mitigate the consequences of an accident; this event is therefore reported under 10CFR50.73(a)(2)(v). Also, again assuming inoperability of these systems, operation with technically inoperable DHR/LPI and CF systems in modes when these systems were required by Technical Specifications to be operable is reported under 10CFR50.73(a)(2)(i)(B).

The inoperability of the DHR suction line was reported to the NRC Operations Center on April 26, 1989, at 2005 hours, in accordance with 10CFR50.72(b)(2)(i). The inoperability of the DHR/LPI and CF systems due to the valve weight discrepancies was reported to the NRC Operations Center on May 24, 1989, at 1500 hours, also in accordance with 10CFR50.72(b)(2)(i).

F. Corrective Actions

A Design Change Package (DCP) installed in October 1988 during refueling outage 1R8 replaced affected DHR suction line supports and modified existing supports to bring the section of the DHR system within code allowables for the seismic loadings. This DCP also included repair of the incorrectly installed hanger.

In March 1989, while the unit was in cold shutdown, modifications were completed on five pipe supports inside containment for the DHR/LPI and CF injection piping affected by valve weight discrepancies. Also, one support outside of containment was modified to help ensure the DHR/LPI system was qualified to its proper design temperature. Qualifying calculations were prepared to demonstrate DHR system operability to 300 degrees for piping upstream of the decay heat coolers and for 280 degrees downstream of the coolers. Piping used by LPI has been qualified for a temperature range from 40 degrees (lowest Borated Water Storage Tank Temperature allowed by Technical Specifications) to 280 degrees (highest postulated process temperature in an accident scenario).

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

As Operations procedures allow the decay heat coolers to be bypassed, applicable procedures have been changed to require RCS temperature to be less than 280 degrees before placing the DHR system in service, thus ensuring that design temperatures of piping downstream of the coolers are not exceeded.

To address potential generic implications, design Engineering has currently been reviewing other selected systems (e.g., Makeup and Purification/High Pressure Injection) to identify any piping which may not have been originally qualified to the proper design temperatures and pressures (as well as equipment weights).

Field walkdowns performed in support of the ASME Section XI In-Service Inspection (ISI) program and the Isometric Update Project will also be conducted that should identify any other existing discrepancies between design basis documents and actual field conditions.

G. Additional Information

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