IES
UTILITIES INC.

August 4, 1994
NG-94-2848

Mr. John B. Martin
Regional Administrator
Region III
U. S. Nuclear Regulatory Commission
801 Warrenville Road
Lisle, IL 60532

Subject: Duane Arnold Energy Center
Docket No: 50-331
Op. License DPR-49
Licensee Event Report #94-010

Gentlemen:

In accordance with 10 CFR 50.73 please find attached a copy of the subject Licensee Event Report.

The following new commitments are made in this letter:

Complete a review to evaluate the adequacy of the technical review process, and any recommendations made in this process, involving General Electric Technical Information Letters by August 31, 1994.

Very truly yours,

David L. Wilson

David L. Wilson
Plant Superintendent - Nuclear

DLW/RJM/eah

cc: Director of Nuclear Reactor Regulation
Document Control Desk
U.S. Nuclear Regulatory Commission
Mail Station P1-137
Washington, D. C. 20555

NRC Resident Inspector - DAEC

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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)

(See reverse for required number of digits/characters for each block)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
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TITLE (4): Manual Scram Due to Loss of Electro-Hydraulic Control(EHC) Oil

EVENT DATE (5)			LER NUMBER (6)			REPORT NUMBER (7)			OTHER FACILITIES INVOLVED (8)	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER
										05000
07	10	94	94	-- 010 --	00	08	04	94	FACILITY NAME	DOCKET NUMBER 05000

OPERATING MODE (9)	1
POWER LEVEL (10)	68

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

	20.402(b)		20.405(c)	X	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)		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 (Include Area Code)
Robert J. Murrell, Licensing Specialist	(319) 851-7900

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
X	JJ	TBG	G084	Yes					

SUPPLEMENTAL 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 15 single-spaced typewritten lines) (16)

On Sunday, July 10, 1994, at 2201 hours, with the plant operating at 68% power, a manual scram was inserted in response to an impending loss of Electro-Hydraulic Control (EHC) reservoir level. The loss of EHC reservoir level was caused by a 210 degree crack in the 1" EHC tubing supplying the number 3 main turbine control valve (CV-3).

All control rods fully inserted and vessel level dropped below the low level setpoint causing groups 2-5 primary containment isolations. Vessel level was restored and returned to normal. Reactor pressure was controlled by the turbine bypass valves and main steam line drains. There were no emergency core cooling system actuations and no safety relief valve openings.

Corrective actions include repair of the ruptured tubing and installation of accumulators on the CVs to prevent reoccurrence of this event. The reactor was restarted on July 16, 1994.

I. DESCRIPTION OF EVENTS:

On Sunday, July 10, 1994, the plant was operating at 100% power. The B Control Building Chiller was in day 6 of a 30 day administrative Limiting Condition for Operation (LCO) and the B Standby Gas Treatment system was in day 1 of a 7 day LCO for an inoperable heater.

At 2150 hours, an Electro-Hydraulic Control (EHC) reservoir low level alarm was received in the Control Room. The Auxiliary Operator was dispatched to investigate the alarm. At 2154 hours, the Auxiliary Operator reported that tank level was at the low level alarm setpoint (-4") and decreasing rapidly. When tank level reached -6" at 2155 hours, the Control Room operators commenced a rapid power reduction in accordance with Integrated Plant Operating Procedure 4 and the Second Assistant was dispatched to the Heater Bay to investigate the cause of the apparent EHC leak. At 2201 hours, the Second Assistant reported a large leak in the Heater Bay and a manual scram was then inserted.

Following the scram, expected core void collapse caused indicated vessel level to drop below the 170" low level setpoint to a minimum of 146.5". All required primary containment isolation system (PCIS) isolations, Groups 2-5, occurred when initiated by the 170" low level indication. Level was returned to normal and maintained with use of the B feedwater pump.

At 2204 hours, the scram was reset and torus cooling was placed in service. At 2211 all PCIS isolations were reset and the reactor water cleanup system was returned to service.

There were no emergency core cooling system (ECCS) actuations and no safety relief valve (SRV) openings during this event. Reactor pressure peaked at 1006 psig and was controlled by the turbine bypass valves and manual operation of the main steam line drains. The reactor was restarted on July 16, 1994, after corrective actions were completed.

II. CAUSE OF THE EVENT:

The cause of the event was a 210 degree crack in the 1 inch fluid actuating supply (FAS) line to CV-3. General Electric Technical Information Letter (GE TIL) 1123-3, "Hydraulic Leaks in BWR Steam Turbine EHC Systems", dated November 24, 1992, identifies this tubing as being susceptible to fatigue failures induced by large pressure pulses in the EHC fluid. These pressure pulses are thought to be induced by electronic noise in the EHC circuitry. An initial review which was performed shortly after receipt of the TIL, had determined that the plant was not susceptible to the failures described within the TIL. The pressure pulses also led to the failure of several EHC tubing supports which caused significant EHC tubing vibration. It is unknown at this time if the cause of the tube crack is from the internal pressure pulses or the external stresses created by the excessive vibrations.

III. ANALYSIS OF THE EVENT

This event had no adverse effect on the safe operation of the plant. The magnitude of the transient was minimized by the response of the Control Room operators who identified the EHC leak and prepared the plant for the impending scram. Prior to inserting the manual scram, power was lowered to 68% by reducing reactor recirculation flow. After inserting the manual scram, pressure control was maintained with the bypass valves and later with the steam line drains when the EHC system was secured. Also following the insertion of the manual scram, reactor water level was recovered from the core void collapse and maintained between 158" and 204". Group isolations 2-5 were received as expected with no major equipment problems. The EHC leak was stopped by securing pumps in the EHC system.

Throughout this event, reactor vessel level and pressure were maintained within safe operating limits and the core thermal power limit was not exceeded. The plant is analyzed for this type of scram. All engineering safety features (ESFs) functioned as designed. There were no ECCS actuations or SRV openings.

IV. CORRECTIVE ACTIONS:

The damaged section of EHC tubing was replaced and the EHC fluid from the failure of the tube was cleaned up. The three similar sections of tubing were inspected with no abnormalities identified. Based on guidance provided within GE TIL 1123-3, hydraulic accumulators have been installed in the supply lines to the turbine control valves. The accumulators absorb the shock from the pressure pulses induced by electronic noise in the EHC circuitry. Other corrective actions include:

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

1. Instrumentation has been temporarily installed to aid in the monitoring and detection of EHC system vibration and pressure fluctuations.

2. A review is underway to evaluate the adequacy of the technical review process, and any recommendations made in this process, involving GE TILs. This review is scheduled to be completed no later than August 31, 1994.

3. The failed section of tubing will be analyzed for failure mode.

V. ADDITIONAL INFORMATION:

A. Several other EHC related scrams have occurred throughout plant history, however, the causes for those events were different than this event.

B. Applicable EIIS System Codes:

1. Electro-Hydraulic Controls - JI, JJ.
2. Reactor Recirculation System - AD
3. High Pressure Coolant Injection System - BJ
4. Reactor Core Isolation System - BN
5. Turbine Steam Bypass Control System - JI