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ILLINOIS POWER
An Illinova Company

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June 9, 1998

Docket No. 50-461

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

Subject: Clinton Power Station - Unit 1
Licensee Event Report No. 98-012-00

Dear Madam or Sir:

Enclosed is Licensee Event Report (LER) No. 98-012-00: High Differential Pressure Required to Close Excess Flow Check Valves Results in Valves Being Outside the Design Basis of the Plant. This report is being submitted in accordance with the requirements of 10 CFR 50.73.

Sincerely yours,

Walter G. MacFarland

Walter G. MacFarland, IV
Senior Vice President and
Chief Nuclear Officer

RAF/mlh

Enclosure

cc: NRC Clinton Licensing Project Manager
NRC Resident Office, V-690
Regional Administrator, Region III, USNRC
Illinois Department of Nuclear Safety

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NRC FORM 366 (4-95) U.S. NUCLEAR REGULATORY COMMISSION

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)
Failure to Test Valves 1SX013D/E/F in Accordance with the In-Service Testing Program 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 NAME	DOCKET NUMBER
									None	05000
06	30	87	98	016	00	06	09	98	None	05000

OPERATING MODE (9)	1	THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more) (11)							
			20.2201(b)		20.2203(a)(2)(v)	X	50.73(a)(2)(i)		50.73(a)(2)(viii)
POWER LEVEL (10)	050		20.2203(a)(1)		20.2203(a)(3)(i)		50.73(a)(2)(ii)		50.73(a)(2)(x)
			20.2203(a)(2)(i)		20.2203(a)(3)(ii)		50.73(a)(2)(iii)		73.71
			20.2203(a)(2)(ii)		20.2203(a)(4)		50.73(a)(2)(iv)		OTHER
			20.2203(a)(2)(iii)		50.36(c)(1)		50.73(a)(2)(v)		Specify in Abstract below or in NRC Form 366A
			20.2203(a)(2)(iv)		50.36(c)(2)		50.73(a)(2)(vii)		

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (Include Area Code)
T. A. Danley, Nuclear Station Engineer	(217) 935-8881, Extension 3959

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

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)

During a System Design and Functional Validation (SDFV) review of the Shutdown Service Water System (SX), a condition was identified where valves 1SX013D, 1SX013E, and 1SX013F, Division 1, 2, and 3 SX Pump Discharge strainer motor operated flush valves respectively, were not included in the In-Service Testing (IST) Program. Further evaluation determined that these valves were required to be included in the IST Program. The cause of this event was personnel error due to a lack of design basis understanding. This lack of design basis understanding resulted in inappropriately downgrading the safety classification of valves 1SX013D / E / F from an active to a passive safety function and removing the valves from the IST Program. The corrective actions for this event are: testing the safety functions of valves 1SX013D / E / F and adding the valves to the IST, Generic Letter (GL) 89-10, and GL 96-05 Programs; revising the Updated Safety Analysis Report (USAR) Table 3.9-5 to include these valves; and enhancing the Engineering Department's use and understanding of the plant's design and licensing bases.

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DESCRIPTION OF EVENT

On May 11, 1998, it was determined that the plant had been in a condition outside the design basis relating to Regulatory Guide 1.11, "Instrument Lines Penetrating Primary Reactor Containment," and, did not meet Improved Technical Specification 3.6.1.3, "Primary Containment Isolation Valves (PCIVs)" (formerly Technical Specification 3.6.4) while in Modes 1 (Power Operation), 2 (Startup), and 3 (Hot Shutdown) since receipt of the Operating License on September 29, 1986. Upon receipt of the Operating License the plant was in Mode 5 (Refueling) with reactor coolant temperature [RCT]at ambient and reactor pressure at atmospheric. This condition resulted from Primary Containment Isolation Excess Flow Check Valves [ISV] 1CM051 and 1CM053 being equipped with poppet springs requiring a closing differential pressure greater than maximum peak drywell pressure expected during a design basis Loss of Coolant Accident.

On October 29, 1996, as a result of surveillance testing, it was identified that instrument line excess flow check valves 1E51-F337A, 1CM003A, 1CM003B, 1SM009, and 1SM010, connected to the containment atmosphere, failed to close at a differential pressure less than the expected peak accident pressure for the containment. It was also identified during the testing that instrument line excess flow check valve 1CM051, connected to the drywell atmosphere, failed to close at a differential pressure less than the expected peak accident pressure for the drywell. The recorded values did, however, meet the acceptance criteria specified in the Operational Requirements Manual (ORM) and CPS 9864.01, "Excess Flow Check Valve Operability Test." Condition Report 1-96-10-460 was written to document and investigate this issue and was added to existing Mode 1, 2, and 3 restraints.

Excess flow check valves are leakage limiting devices installed in various sensing lines that penetrate primary containment. The valves are designed to remain open under no flow conditions and to restrict flow in the event of a break in the instrument line downstream of the valve under accident conditions. The valve has two major operating parts, a poppet assembly consisting of a flow restrictor (poppet) and a spring; and a position switch [33]. The poppet assembly is spring loaded in the open position. As flow increases through the valve, the induced drag forces on the poppet compresses the spring until the assembly is fully seated. The magnitude of the flow needed to close the valve is based primarily on the size of the spring; the stiffer the spring, the greater the required flow. The position switch is a magnetic switch that senses the position of the poppet assembly to indicate, in the Main Control Room, if the valve is open or closed.

Clinton Power Station uses excess flow check valves manufactured by Dragon Valves, Inc., under the following part numbers: 14455-1 (Type 1), 14455-3 (Type 3), 14455-5 (Type 5), 14455-7 (Type 7), and 14455-9 (Type 9). Types 1, 5, and 9 are used in low pressure air applications, Type 3 in low pressure water applications, and Type 7 in high pressure water applications. All five valve types share the same valve body design. Types 1, 5 and 9 valves use the same poppet but differ in the stiffness of the installed spring; light, moderate, and heavy, respectively. Types 3 and 7 valves use a poppet that is slightly different in length than that used in valve Types 1, 5, and 9. The spring used in the Type 3 valve is considered light, and the Type 7 uses the same spring as the Type 5. Valves 1E51-F337A, 1CM003A, 1CM003B, 1SM009, and 1SM010 are Type 1 and 1CM051 is a Type 9.

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

The failure mode investigation determined that the most probable failure mode for the five Type 1 valves was inflated values resulting from errors introduced by the test method used in measuring the differential pressure. The most likely failure mode for 1CM051 was the use of a spring that was stiffer than required. Discussions conducted with Dragon Valves revealed that they did not have test data or engineering analysis that would establish the design closing differential pressure used to size the poppet assembly. As a result Clinton Power Station conducted testing to determine closing characteristics of the Type 1, 3, 5, and 9 valve springs. The results of this testing are as follows

Valve Type	Maximum Flow	Average Closing Differential Pressure
Type 1	2.0 SCFM	3.8 psig
Type 5(7)	2.7 SCFM	7.1 psig
Type 9	5.4 SCFM	25 psig
Type 3	.4 gpm (1.7 SCFM)	3.5 psig

Based upon this testing it was determined that valves 1E51-F337A, 1CM003A, 1CM003B, 1SM009, and 1SM010, would have closed below the expected peak accident pressure for the containment as required, but the differential pressure required to close the two Type 9 valves in the plant, 1CM051 and 1CM053, exceeded the expected peak drywell pressure.

No automatic or manually initiated safety system response was necessary to place the plant in a safe and stable condition. This event was not directly affected by other inoperable equipment or components.

CAUSE OF EVENT

The cause of this event has been attributed to personnel error during the development of the design/purchase specifications for the excess flow check valves. Instructions were not included in the specifications on how to apply the minimum/maximum differential pressure and allowable leakage information provided on the valve data sheets. The specification should have clearly stated that the valves were to close by the minimum differential pressure specified and that the allowable leakage should not be exceeded when subjected to the maximum differential pressure specified. Because this information was not included, Dragon Valves supplied excess flow check valves with springs that were designed to close between the minimum and maximum pressures stated on the valve data sheet. As a result, valves 1CM051 and 1CM053 were provided with springs that were too stiff for the application and incorrect acceptance criteria was incorporated into the Technical Specifications and CPS 9864.01, "Excess Flow Check Valve Operability Test."

CORRECTIVE ACTIONS

Engineering Change Notices were developed and implemented to replace Type 5 and Type 9 Excess Flow Check Valve springs with Type 1 springs. This will provide additional assurance that the excess flow check valves will close below expected peak pressures as required. This action addresses cause of the event. Because of the age of this event no action is being taken to specifically address the issue of the personnel error.

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

An improved test methodology, based upon the test methods used to determine the valve closing characteristics, was developed and incorporated into CPS 9864.01, "Excess Flow Check Valve Operability Test."

The acceptance criteria in the ORM and CPS 9864.01, "Excess Flow Check Valve Operability Test." were revised.

Modified Type 5 and Type 9 excess flow check valves and valves 1E51-F3337A, 1CM003A, 1CM003B, 1SM009, and 1SM010 were tested/re-tested with satisfactory results.

ANALYSIS OF EVENT

This event is reportable under 10CFR50.73(a)(2)(i)(B) because Clinton Power Station was not in compliance with Improved Technical Specification 3.6.1.3 (formerly Technical Specification 3.6.4) while in Modes 1, 2, and 3, since receipt of the Operating License on September 29, 1986, because excess flow check valves 1CM051 and 1CM053 were not OPERABLE, in that the differential pressure required to close these valves was greater than the expected peak drywell pressure.

This event is also reportable under 10CFR50.73(a)(2)(ii)(B) because the differential pressure required to close excess flow check valves 1CM051 and 1CM053 was more than the expected peak drywell pressure. Therefore the requirement of Regulatory Guide 1.11, "Instrument Lines Penetrating Primary Reactor Containment," for the valves to "close or be closed if the instrument line integrity outside containment is lost during normal operation or accident conditions" was not met.

This condition is of minor nuclear safety significance because the likelihood of the instrument line integrity being lost is small since the lines and instruments are seismically restrained and even during accident conditions these lines are subject to very low pressures. Additionally these lines can be manually isolated.

ADDITIONAL INFORMATION

This event was evaluated as not reportable in 1996 because the measured values did not exceed the acceptance criteria specified in the Technical Specifications and the requirements of Regulatory Guide 1.11 were misinterpreted

Clinton Power Station has not reported any similar events in recent history.

For further information regarding this event, contact S. J. Kowalski, Plant Engineering, at (217) 935- 8881, extension 3902.