James A. FitzPatrick
Nuclear Power Plant
P.O. Box 41
Lycoming, New York 13093
315 342-3840

New York Power Authority

William Fernandez II
Resident Manager

April 19, 1990
JAFP-90-0341

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

SUBJECT: DOCKET NO. 50-333
LICENSEE EVENT REPORT: 90-011-00
Shutdown Cooling System Isolation

Dear Sir:

This Licensee Event Report is submitted in accordance with 10 CFR 50.73(a)(2)(iv).

Questions concerning this report may be addressed to Mr. Hamilton Fish at (315) 349-6013.

Very truly yours,

WILLIAM FERNANDEZ

WF:HCF:lar

Enclosure

cc: USNRC, Region I
USNRC Resident Inspector
INPO Records Center
American Nuclear Insurers

Cert No P148799818

IE22
1/1

NRC Form 366 (9-83)

U.S. NUCLEAR REGULATORY COMMISSION
APPROVED OMB NO. 3150-0104
EXPIRES: 8/31/88

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
JAMES A. FITZPATRICK NUCLEAR POWER PLANT	0 5 0 0 0 3 3 3	1 OF 0 5

TITLE (4) Shutdown Cooling Isolation - Deficient Procedure Allowed Reverse Flow of Normal Starting Pressure Transient to Trip Isolation Pressure Switches

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 5 0 0 0
0 3	2 0	9 0	9 0	0 1 1	0 0	0 4	1 9	9 0		0 5 0 0 0

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

OPERATING MODE (9): N

POWER LEVEL (10): 0 0 0

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	
	AREA CODE	
Hamilton C. Fish	3 1 5	3 4 9 - 6 0 1 3

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 fifteen single-space typewritten lines) (16)

EIIS Codes are in []

At 9:20 A.M. on 3/20/90, while the reactor was in hot shutdown following a scram (LER-90-009), the reactor high pressure isolation logic tripped and isolated Residual Heat Removal (RHR) [BO] B Shutdown Cooling (SDC) system as it was being started. This logic is set to trip at less than or equal to 75 psig to protect and isolate the low pressure piping of the SDC from high reactor pressure. At the time of the trip the reactor pressure was 6 psig, which is less than one tenth of the trip setpoint. Investigation found that the trip logic pressure sensors were properly calibrated and set at approximately 60 psig equivalent reactor pressure and that the reactor pressure reading of 6 psig was accurate.

RHR pump D discharges into a common header with reactor water recirculation (RWR) [AD] pump B. Opening the SDC injection valve following RHR pump start initiated a hydraulic pressure transient which was transmitted in a reverse flow direction through RWR pump B to the isolation logic pressure switches located in the suction line from the reactor to the RWR pump. The pressure transient was sufficient to trip the pressure switches and isolate the system.

The operating procedure was revised to stop and isolate the reactor water recirculation pumps prior to starting the RHR pump. LER-90-002 reports an almost identical event.

NRC Form 366 (9-83)

NRC Form 366A
(9-83)

LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

U.S. NUCLEAR REGULATORY COMMISSION
APPROVED OMB NO 3150-0104
EXPIRES 8/31/85

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
JAMES A. FITZPATRICK NUCLEAR POWER PLANT	0 5 0 0 0 3 3 3	9 0	— 0 1 1	— 0 0	0 2 OF 0 5

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

EIIS Codes are in []

Description

On March 20, 1990 the reactor was in a hot shutdown condition following an automatic scram (LER-90-009) from full power the previous day (3/19/90) and was in the process of being brought to cold condition. To bring the reactor to a cold condition and to maintain the reactor water temperature below 212° F, the shutdown cooling mode (SDC) of the Residual Heat Removal (RHR) [BO] is started. During initiation of the system an automatic isolation occurred at 9:20 A.M. due to an apparently spurious high reactor pressure signal.

The piping system and heat exchangers of the SDC are rated for low pressure (compared to normal reactor operating pressure) service. The low pressure RHR system must be protected from exposure to normal reactor operating pressure because the system takes suction from and discharges to the reactor water recirculation system (RWR) [AD] piping which is connected directly to the reactor vessel. This overpressure protection is accomplished by isolation signals which close suction isolation valves (10MOV-17 and -18) on the SDC system when reactor pressure exceeds a setpoint of less than or equal to 75 psig reactor dome pressure (Technical Specification Table 3.2-1) is sensed.

The operators reduced reactor pressure to approximately 6 psig as determined from the plant process computer. The operator also verified that the reactor high pressure protective isolation (75 psig) relays were reset. When sufficient RHR water purity was obtained, the B side RHR system was lined up for the SDC mode. RHR pump D was started. The Low Pressure Coolant Injection (LPCI) outboard injection valve 10MOV-27B was jogged open. At 9:20 A.M. annunciator 09-4-3-22, "Shutdown Cooling Suction Header Pressure High", activated, isolation valves 10MOV-17 and -18 closed, and RHR D pump tripped. The annunciator cleared immediately. The operators verified that a proper isolation had taken place and that the isolation relays 16A-K28 and 50 had reset. The keepfull system was restored to service on RHR B system.

The reactor pressure was then further reduced by continued venting to approximately atmospheric pressure. The SDC system was restarted at 10:42 A.M., 1 hour and 22 minutes after the isolation. There was no interruption in the removal of decay heat.

The pressure switches for the reactor high pressure protection isolation to the RHR system had been functionally tested on March 15, 1990, five days before this event, and found to be within calibration tolerances. No adjustments were made. Trip points were set at an equivalent (allowing for a 35 psig head correction factor) reactor pressure of approximately 60 psig which is 15 psi less than the Technical Specification limit.

NRC FORM 366A
(9-83)

Cause

Investigation of this event considered as possible causes the potential for the pressure sensing instruments to be out of calibration and the possibility of a pressure surge from initial pump operation and flow regulation causing a hydraulic pressure transient at the isolation logic pressure sensors. A check of the pressure switches for the isolation logic verified that they were in calibration five days before the event and they were left as found.

The reactor pressure indication of 6.3 psig was obtained by the operator from the computer system display. Investigation of the validity of this value found that it was a composed point which is calculated from the input from ten wide-range pressure transmitters. The computer system for this point is provided with signal checking and validation logic which would have provided indication to the operator for a condition where the validity of the pressure reading was questionable. Therefore, this reading is considered to be highly reliable.

Strip chart recordings of reactor pressure are available. However, they are recorded on slow speed of one inch per hour. The range for these charts is from 0 to 1,500 psig compressed to a vertical scale of approximately 4.25 inches where a 0.057 inch division equals 20 psig. On this scale, the resolution of a small pressure spike (from 6 psig to 60 psig lasting less than a few seconds) would be insufficient to determine whether a small local pressure transient had in fact taken place.

Following the previous identical occurrence of this event on January 20, 1990 (LER-90-002), a hydraulic pressure transient was considered to be the most likely cause of the isolation. Drawings show the pressure sensors located on the suction of loop A of the RWR system. However, the RHR SDC flow was being injected into the discharge loop B of the RWR system. Thus a hydraulic pressure transient would have had to be of sufficient magnitude to be transmitted through the reactor vessel volume to the opposite RWR loop.

Because this hypothesis could not be strongly supported if the pressure sensors were in the opposite loop from the injection loop, it was decided to investigate the accuracy of the drawings. During the fall 1989 outage, the pipe support inspection program had performed as-built verification walkdowns for small bore piping in the primary containment drywell. The as-found piping was compared to the stress isometric drawings (MSK). These drawings were verified to be accurate. When the MSK for the SDC pressure sensor piping was checked, it showed that the instrumentation was in fact installed on the RWR pump B reactor suction. Therefore, the system flow diagrams (FM) which showed the instrumentation to be located on the RWR A loop were not correct. Thus

the verified correct MSK drawings support the fact that the B RHR SDC pump D was injecting into the B RWR loop where the pressure sensors are in fact located and support the hypothesis that a start-up pressure transient of the SDC was transmitted through the RWR system and tripped the isolation pressure switches.

The potential for reverse flow exists when pumps operate in parallel in a single system and discharge into a common pressurized plenum. Any mismatch between the discharge heads of the pumps will induce a hydrodynamic disturbance. The potential for this phenomenon to occur exists at operating GE/BWRs equipped with variable speed reactor water recirculation pumps. This is the situation which occurred when the residual heat removal (RHR) pump discharge was connected into the same manifold as the discharge from the recirculation pump. When the recirculation pump discharge head was less than the pressure maintained in the manifold by the RHR system, a reverse flow developed through the recirculation pump.

Therefore, the immediate cause of this event is attributed to a hydraulic pressure transient resulting from initial flow from the SDC system initiation being transmitted in a reverse direction through the RWR pump B to its suction line from the reactor.

This problem had not been previously documented as an LER prior to this year for two reasons. First, the A side of the RHR SDC system has been used almost exclusively when SDC was required. This was done because loop A contains two motor operated valves to flush the system prior to use and to control reactor vessel water level by bleeding control rod drive [AA] in-leakage to the radwaste system. In contrast, the B side requires use of a manual valve located in a remote section of the plant to accomplish their function. Understandably the A side was used preferentially for SDC. Because the A side SDC injected into the discharge of the RWR pump A, the initial pressure transient was hydraulically isolated by the volume of the reactor vessel water from the isolation pressure sensors located in the suction of RWR pump B. Thus starting the SDC system on the A side did not result in an isolation due to the initial pressure surge.

Secondly, the isolation of the SDC due to a spurious signal was not a reportable event under the rules in effect during the first ten years of plant operation. Thus, even if the B side of SDC was used and initially isolated, it was not a reportable event.

The root cause of the event was a deficient operating procedure Cause Code [D]. The procedure did not direct stopping the RWR pumps and closing of pump discharge valves which would have isolated the pressure sensors from the normal and expected pressure transient resulting from pump start-up.

Analysis

Isolation of the shutdown cooling system by the reactor high pressure isolation logic is reportable under the provisions of 10 CFR 50.73(a)(2)(iv) as an activation of an engineered safety feature [JE]. There were no system or equipment failures. The shutdown cooling system isolation was performed in accordance with design. Although the isolation reset immediately, it was decided to delay attempted restart of the SDC until the reactor pressure had dropped to a lower value. The SDC was restarted successfully at 10:42 A.M., 1 hour and 22 minutes after the isolation. The system could have been restarted before that time if it had been required. There was no interruption in the removal of decay heat from the reactor.

Corrective Action

1. Temporary changes were made to Operating Procedure OP-13, "Residual Heat Removal System", under the section for "Shutdown Cooling Configurations" to require stopping of the reactor recirculation pumps [AD] and closing of the pump discharge valves prior to starting the RHR pumps for shutdown cooling. This change is in accordance with recommendations of NSSS supplier and effectively isolates the pressure sensor from the initial pressure surge. In addition, a caution was added to note that spurious high suction pressure isolations may occur when starting the B side RHR system.

   The effectiveness of this change was verified during a planned plant shutdown on March 31, 1990. SDC was initiated using the revised procedure with sensitive pressure monitoring and recording instruments attached to the system. No pressure transients were observed and the system did not isolate.

2. The physical location of the instrumentation for system isolation was checked on verified MSK drawings which had recently been walked down. The system flow diagram (FM) plant drawings will be corrected.

Additional Information

LER-90-002 describes an almost identical isolation of the shutdown cooling mode of the RHR system due to a spurious high pressure signal.