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)	DOCKET NUMBER (2)	PAGE (3)
McGuire Nuclear Station, Unit 2	0 5 0 0 0 3 7 0	1 OF 0 4

TITLE (4)
Lo-Lo Steam Generator Level Resulted in a Reactor Trip

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 8	2 1	8 4	8 4	— 0 2 0 —	0 0	0 9	2 0	8 4		0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9): 3

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)	[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	
Phillip B. Nardoci, Licensing Engineer	7 0 4	3 7 3 - 7 4 3 2

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	S B	V	C 7 1 0	Y					
X	E C	G E N	S 2 5 0	Y					

SUPPLEMENTAL 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)

On August 21, 1984, the unit two reactor protection system was actuated when the water level in steam generator 2C reached the lo-lo level trip setpoint. The primary reason for the reactor trip was the opening of safety relief valve 2SV-9 below its setpoint. Contributing factors were the loss of supply steam for the main feedwater pump turbines and the surge of cool auxiliary feedwater. This loss of supply steam was a result of a unit one reactor trip (Ref. LER 369/84-24). Unit two was in Mode 3 with A and B shutdown bank control rods withdrawn at the time of this event.

This event is attributed to Component Malfuncation due to the safety relief valve, 2SV-9, opening before reaching its open setpoint range. An Unusual Service Condition also contributed to the event due to the loss of supply steam because of the unit one reactor trip.

The safety relief valve's setting was adjusted to its proper value, and the valve retested with satisfactory results. Plant systems responded as designed for a reactor trip transient with no abnormal conditions. The opening of valve 2SV-9 posed no serious safety concern since it would not have lowered steam pressure below its design condition. The valve responded correctly, except that its setpoint was too low. Health and safety of the public were unaffected.

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On August 21, 1984, at 2225 hours, the unit two reactor protection system (EIIS:JC) was actuated when the water level in steam generator (EIIS:GEN) 2C reached the lo-lo level trip setpoint. Unit two was in Mode 3 in the process of start-up, but had stopped to correct a problem in the control rod drive system (EIIS:AA) when the reactor tripped. Shutdown control rod banks (EIIS:ROD) A and B were withdrawn when the unit tripped. The unit two reactor trip was the result of several separate incidents. The primary reason for the trip was the opening of relief valve (EIIS:V) 2SV-9 (main steam 2C safety No. 2) at 1140 psig, 50 psig below its setpoint of 1190 psig. Contributing factors were the loss of supply steam (EIIS:SB) for the main feedwater pump turbines and the surge of cool auxiliary feedwater causing a shrinkage of water level in steam generator 2C. This loss of supply steam was a result of a unit one reactor trip (at 2148) caused by a loss of offsite power (Ref. LER 369/84-24).

Unit one had been supplying auxiliary steam (EIIS:SA) to unit two. The unit two control operators realized immediately after the unit one trip that the supply of steam needed to operate the main feedwater pump turbines (EIIS:TRB) was limited. Operators initiated start-up of the auxiliary steam boiler, although it would take approximately one hour before an ample quantity of steam could be supplied. At the time the unit was steady and the control operators knew that they could start the motor driven auxiliary feedwater pumps (EIIS:P) if the level started to decrease in the steam generators.

After the unit one reactor trip, the unit two control operators discovered that the main steam isolation valves (EIIS:V) 2SM-1, 2SM-3, 2SM-5, and 2SM-7 had closed as a result of power interruptions caused by the unit one trip (the solenoid valves that hold the valves open had been deenergized). It is theorized that the fuse (EIIS:BRK) on inverter (EIIS:GEN) KXA had blown when the static transfer switch (EIIS:XIS) had moved to the alternate position and then back to the normal position. The movement of the switch to the alternate position saw a low voltage on auxiliary control power bus (240/120 VAC) (EIIS:CON) MKA (alternate power was being supplied by Unit 1) and the voltage started to drop on bus KXA. With this voltage drop, the KXA residual voltage and normal incoming voltage became up to seven degrees out-of-phase. Therefore, when the static switch returned to the normal position, the fuse blew due to this out-of-phase overcurrent condition. There is no apparent reason why the inverter swapped since its normal incoming power should have been normal. With the normal incoming power supply blocked, the static switch moved to the alternate power source (MKA) to maintain the bus. The power was coming from the 600 volt shared motor control center (SMXS) which had just gone through a dead bus transfer due to the unit one loss of offsite power. The shared motor control center was being supplied by the unit two load center 2SLXF instead of the unit one load center 1SLXF.

At 2152, the Control Room received a low main feedwater pump discharge pressure alarm. After noticing that main steam isolation valves had closed, the operators took manual control of the main steam power operated relief valves for pressure control and started to equalize pressure across the isolation valves by opening the main steam isolation bypass valves. The isolation valves were opened later with no apparent problems.

When the pressure in steam generator 2C reached 1142 psig, relief valve 2SV-9 opened (at 2209). The digital computer point alarmed (EIIS:ANN) and printed on the alarm typer; however, this point is for information only and does not appear on the alarm

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

video screen and may go undetected by the operator. The control operators were unaware that the valve had opened. The valve reseated four minutes after it had opened. During this time interval, steam pressure in steam generator 2C decreased approximately 40 psig and water level decreased approximately five percent.

Due to the loss of an adequate steam supply to operator the main feedwater pumps, the control operators were prepared to start the motor driven auxiliary feedwater pumps to maintain level if the level started to decrease. At ~2212 they throttled closed the auxiliary feedwater pump discharge control valves 2CA-40, 2CA-44, 2CA-56, and 2CA-60 to prevent shocking the steam generators when the auxiliary feedwater pumps were started and affecting the level too rapidly. The valves were also throttled closed so that the level in the other steam generators would not be affected. When they noticed the level was decreasing in steam generator 2C, the operators started the auxiliary feedwater pumps (at 2215).

Relief valve 2SV-9 reopened at 2216 and remained open for approximately ten minutes. This resulted in a pressure drop of approximately 30 psig and a level decrease of approximately fifteen percent. The open relief valve caused rapid cooling in steam generator 2C which resulted in a shrinkage of feedwater in the steam generator. The operators were still unaware that the relief valve was open, and they did not learn it until the alarm typewriter printout and transient monitor (EIIS:IG) data were reviewed. The discharge control valves were throttled closed too much and the level continued to drop. At ~2222 the control operators started increasing flow to steam generator 2C by slowly opening discharge valve 2CA-44, allowing a flow of approximately 20 gpm. The relief valve reseated nine seconds before the reactor tripped. The level continued to drop, and the operators responded by opening 2CA-44 more, increasing feedwater flow. This caused rapid cooling and shrinkage of the water level to the lo-lo level setpoint, generating a reactor trip signal at 2225.

Investigation found relief valve 2SV-9 to have a relief setting of 1142 psig instead of 1190 psig. The relief valve setting was adjusted to 1185 psig, and the valve was retested with satisfactory results. Review of all the relief valves on unit two found that six of the twenty relief valves have had either their relief setpoints checked or adjusted. None of these six valves have had this work performed more than once. The plant systems responded as designed for a reactor trip transient with no abnormal conditions. The reactor remained subcritical at all times. The reactor trip reinserted shutdown banks A and B. Pressurizer pressure remained within 10 psig of its reference value (2235 psig) at all times. The PORV's and Pressurizer Code Safety Valves were not challenged. Maximum Reactor Coolant (EIIS:AB) Average Temperature was 563°F, and the minimum was 560°F. This is slightly above the expected no-load value of 55.7°, and was a result of the slightly higher than normal steam pressures. Pressurizer level remained on scale at all times, and was within ~7% of the programmed value for the average coolant temperature. Minimum level was 30%. Letdown was not isolated.

Peak steam generator pressure was 1141 psig. Although 2SV-9 opened twice at 1142 psig prior to the trip, it did not reopen after the trip. No other safety valves opened post-trip. The steam generator PORV's were taken into manual and closed to prevent steam generator inventory losses. Steam pressure stabilized at ~1120 psig.

NRC Form 366A
(9-83)

U.S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

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

Following the reactor trip the auxiliary feedwater (EIIS:BA) valves to all 4 steam generators opened to their travel stops on the auxiliary feedwater automatic start signal. (Both motor driven pumps had been started prior to the reactor trip). Main feedwater (EIIS:SJ) was isolated on reactor trip with coincident low T_{ave} shortly after the trip. The minimum level on steam generator C was ^9% narrow range, which occurred just after the reactor trip. The auxiliary feedwater valves were taken in manual about one minute after the trip, and flow to steam generators A,B, and D was throttled as their levels were within 2% of the no-load target value of 38%. Flow to steam generator C was adjusted to 140 gpm and was maintained there until level reached 37% narrow range. The flow was reduced at that time.

Offsite power to McGuire Unit 2 was maintained at all times. Safety Injection (EIIS:BG) was not actuated. The Main Steam Isolation valves closed as designed on loss of power.

If this event had occurred at power, it is likely that a reactor trip would have occurred. Closure of the Main Steam Isolation Valves at power would have caused a reactor trip on low steam generator level.

The opening of 2SV-9 posed no serious safety concern since it would not have lowered steam pressure below its design condition. This valve is one of five relief valves that have different opening setpoints. There is one valve with a setpoint lower than 2SV-9 which is 1150 psig. The valve 2SV-9 would have opened in a design event earlier than would have been anticipated, but it would not have complicated the event. The valve responded correctly, except that its setpoint was too low. The health and safety of the public were not affected by this incident.

DUKE POWER COMPANY
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CHARLOTTE, N.C. 28242

HAL B. TUCKER
VICE PRESIDENT
NUCLEAR PRODUCTION

TELEPHONE
(704) 373-4531

September 20, 1984

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

Subject: McGuire Nuclear Station, Unit 2
Docket No. 50-370
LER 370/84-20

Gentlemen:

Pursuant to 10 CFR 50.73 Sections (a)(1) and (d), attached is Licensee Event Report 370/84-20 concerning a reactor trip resulting from Lo-Lo steam generator level which is submitted in accordance with §50.73 (a)(2)(iv). Initial notification of this event was made (pursuant to §50.72 Section (b)(2)(ii)) with the NRC Operations Center via the ENS on August 22, 1984. This event was considered to be of no significance with respect to the health and safety of the public.

Very truly yours,

H.B. Tucker / PBN

Hal B. Tucker

PBN:mjf

Attachment

cc: Mr. James P. O'Reilly, Regional Administrator
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cc: Mr. W. T. Orders
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McGuire Nuclear Station

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