CP&L

Carolina Power & Light Company

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December 14, 1990

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U.S. Nuclear Regulatory Commission
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BRUNSWICK STEAM ELECTRIC PLANT UNIT 2
DOCKET NO. 50-324
LICENSE NO. DPR-62
SUPPLEMENTAL LICENSEE EVENT REPORT 2-90-008

Gentlemen:

In accordance with Title 10 of the Code of Federal Regulations, the enclosed Supplemental Licensee Event Report is submitted. The original report fulfilled the requirement for a written report within thirty (30) days of a reportable occurrence and was submitted in accordance with the format set forth in NUREG-1022, September 1983.

Very truly yours,

J. L. Harness

J. L. Harness, General Manager
Brunswick Nuclear Project

TMJ/

Enclosure

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IE22
11

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NRC FORM 366 U.S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT (LER)

APPROVED OMB NO. 3150-0104
EXPIRES: 4/30/92
ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS INFORMATION COLLECTION REQUEST: 50.0 HRS. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE RECORDS AND REPORTS MANAGEMENT BRANCH (P-530), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555, AND TO THE PAPERWORK REDUCTION PROJECT (3150-0104), OFFICE OF MANAGEMENT AND BUDGET, WASHINGTON, DC 20503.

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Brunswick Steam Electric Plant Unit 2	05000324	01 OF 09

TITLE (4) SCRAM Resulting from Turbine Trip on High Level due to Blown Fuse in FW Logic

EVENT DATE (5)			LER NUMBER (6)					REPORT DATE (7)			OTHER FACILITIES INVOLVED (8)	
MONTH	DAY	YEAR	YEAR		SEQ. NO.		REV. NO.	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER
08	16	90	90	-	008	-	01	12	14	90		

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

OPERATING MODE (9)	1		20.402(b)		20.405(c)	X	50.73(a)(2)(iv)		73.71(b)
POWER LEVEL (10)	100		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 and Text)
			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
THERESA M. JONES, REGULATORY COMPLIANCE SPECIALIST	(919) 457-2039

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	SB	--FU	S156	Y						

SUPPLEMENTAL REPORT EXPECTED (14)				EXPECTED SUBMISSION DATE (15)	MONTH	DAY	YEAR
X	YES (If yes, complete EXPECTED SUBMISSION DATE)		NO		03	15	91

ABSTRACT (Limit to 1400 spaces, i.e. approximately fifteen single space typewritten lines) (16)

On August 16, 1990, Unit 2 reactor was at 100% power. The RPS, HPCI, RCIC, ADS, RHR/LPCI, CS, SBGT, SLC, DG and plant electrical system were operable in standby readiness. The reactor level control system was operating in automatic - three element control and at 185 inches. At 0942, the reactor automatically shutdown on a "TSV Fast Closure" RPS trip signal caused by a turbine trip on reactor high water level. During this event, the HPCI turbine stop valve cycled closed and then open, water intrusion into the HPCI oil was noted, the RCIC barometric condenser vacuum pump experienced an electrical fault, and a loss of the recirculation pumps resulted in temperature transients in the vessel. HPCI and RCIC operability were not affected, the recirculation pumps are now being powered from the start up auxiliary transformer to prevent their loss during future reactor trips. The cause of this event was failure of primary power fuse C32-F5, which supplied power to the steam flow inputs of the three element feedwater control logic. Loss of the steam flow inputs resulted in a maximum demand signal to the RFPs and a rapid increase in reactor level up to the high level turbine trip point which, in turn, caused a reactor scram on TSV position. Primary power fuse C32-F5 and its associated circuit were evaluated. The fuse has been replaced. The failed fuse was analyzed by the HE&EC with the initial conclusion that the fuse failed due to a short duration exposure to a current in excess of 20 amps. The HE&EC is continuing its investigation. A supplement to this report will be issued by March 15, 1991. Past similar events include LERs 2-88-018, 2-88-023 and 2-90-008. The safety significance of this event is minimal. The plant responded as designed.

INITIAL CONDITIONS

On August 16, 1990, Unit 2 reactor was at 100% power. The RPS, HPCI, RCIC, ADS, RHR/LPCI, CS, SBGT, SLC, DG and plant electrical system were operable and in standby readiness. The reactor feedwater level control system was operating in automatic - three element control and level was being maintained at 185 inches (ie; normal operating level). Reactor pressure was 1002 psig.

EVENT DESCRIPTION

At 0942 on August 16, 1990, the Unit 2 reactor automatically shutdown on a "TSV Fast Closure" RPS trip signal. The TSV fast closure was caused by a turbine trip on reactor high water level. Preceding the reactor trip, the RO observed the four individual steam flow indications (2-C32-R603A through D) rapidly decrease to zero. A loss of the total steam flow indication on the steam flow/feed flow recorder (2-C32-FR-R607) was also noted. The RO responded to the resulting level transient by adjusting the feedwater master control level setpoint. When the transient continued, the RO placed the feedwater master control system to manual. At that time, the RO noted an erratic increase in reactor level which subsequently increased to the high water level turbine trip setpoint (208 inches). The turbine tripped off line and the reactor scrammed on TSV Fast Closure. After the scram, the RO performed automatic actions required by the EOPs and subsequent actions were directed by the SF utilizing the EOP flowchart Path-3, High Power SCRAM. The RFPs had also tripped on the high level. The corresponding loss of feed flow to the reactor resulted in level decreasing. At 166 inches a LL#1 signal was generated and PCIS groups 2 and 6 automatically isolated. Group 8 also received an isolation signal but the valves were already closed and automatic actuation did not occur, per design. At 0946, to increase reactor level, RCIC was manually initiated at 135 inches reactor level. At 0948, HPCI was manually initiated at 122 inches. At 0949, a second RO started the 2A RFP and began injecting feedwater into the vessel. With reactor level restored and pressure stabilized, the EOP was exited and GP-05, Unit Shutdown, was entered at 0955.

EVENT INVESTIGATION

SEQUENCE OF EVENTS/INVESTIGATION

09:41:56 A primary power fuse (C32-F5) fails open in the Feedwater Level Control System (FWLCS), panel (H12-P612). Power is lost to modules C32-4405A/D, K616, K620, K628 and K650. Power loss to C32-4405A/D causes an immediate loss of the four steam flow signals.

Loss of primary power to power supply C32-K620 causes loss of indication of Turbine Steam (First stage pressure) and narrow range reactor pressure. These signals drop to zero about 2 seconds after the loss of power. This delay is consistent with capacitors in the power supply.

Loss of power to trip module C32-K628 will supply a partial permissive for the 45% speed reactor recirculation pump runback circuit, however due to later events, this runback does not play a part in the event.

The most serious problem is created by loss of power to modules K616 and K650. Because FWLCS is operating in "3 Element", the power loss causes the level feedback signal going to the master feedwater controller (R600) to drop to zero. The controller "sees" this as a low reactor level and increases the speed demand signal to both the "A" and "B" feedpump turbines. The pumps speedup, level in the vessel begins to climb.

Note: It is possible the output of module C32-K650 would vary erratically during the time DC power inside the module is decaying away. This in turn could cause the deviation meter on the master controller to swing and give the appearance of erratic level swings.

The cause of the fuse failure has not been determined. The circuit was investigated by I&C personnel and no abnormal loading was found. The fuse was replaced. The original fuse is an Appendix R fuse which places its installation at three to five years ago but a definite date has not been determined. The failed fuse was sent to the Harris Energy and Environmental Center (HE&EC) for failure analysis. As a precautionary measure, the unit was returned to operation in single element control to prevent a loss of feedwater control should the fuse fail again. The fuse performance was reviewed by the Plant Nuclear Safety Committee on September 14, 1990, and the unit was allowed to transfer to three element control.

On October 12, 1990, at 1401, another Unit 2 reactor cram on TSV Fast Closure occurred when a second Gould Shawmut fuse blew in the feedwater control circuitry. This event was reported in LER 2-90-016. The involved fuse was also sent to the HE&EC center for analysis and an event recorder is currently monitoring the FW power supply circuitry. In addition, Engineering Evaluation Report (EER) 90-0262 was written to allow temporary replacement of the Gould Shawmut fuses in the Unit 2 feedwater control circuitry with Bussman Min fuses.

The draft report from the HE&EC concludes that the Gould Shawmut A25Z2 fuse was found to not exhibit any features which would be interpreted as defective or possibly responsible for the failure of this fuse. Based on limited testing and the observed features, it is thought that the submitted fuse failed due to a short duration exposure to a current in excess of 20 amps. The draft report recommended that additional testing of Gould Shawmut A25Z2 fuses be conducted to more accurately determine the failure conditions for the submitted fuse. This testing is in progress.

09:42:02 Reactor high level alarm is received (192").

This response is consistent with three element control logic. The zero indication of steam flow resulted in the RFPs increasing to maximum flow causing a rapid increase in water level.

09:42:05 Increased feedwater flow causes condensate pressures to drop. The "C" condensate booster pump auto starts, per design.

09:42:06 The increased feedwater flow causes a reactor power transient due to positive reactivity addition. APRMs exceed the 108% rod block alarm setpoint. Peak power is 110%.

The increased feed flow resulted in cooler water being injected to the vessel. The noted increase in power is consistent with an injection of cooler water.

09:42:11 Reactor high level trip setpoint (208") is reached, a trip signal is sent to the main turbine and both feedwater pumps, per design to protect the turbines from water impingement.

Closure of the Turbine Stop Valves (TSV) generates the reactor SCRAM signal and rod insertion begins. This is an anticipatory SCRAM designed to minimize/prevent a pressure spike upon closure of the stop valves. All four Reactor Protection System (RPS) channels also receive a trip signal from the Turbine Control Valve Fast Closure (TCVFC) circuits.

The turbine trip causes a generator trip which removes power to the unit auxiliary transformer (UAT). Loss of power to the UAT removes the voltage from the "2B" 4160 volt bus which supplies power to the reactor recirculation (RR) pumps. Power is lost to the RR M/G set and the RR pumps begin to coast to a stop.

09:42:22 Because feed to the vessel was secured when the feedpumps tripped, level decreased to Low Level 1 (166"). The four RPS channels detected the low level and generated a trip signal. No rod movement occurred because the rods were already inserted. PCIS group 2, 6 and 8 isolation commands were generated, per design. Valves in groups 2 and 6 closed, per design; group 8 valves were already closed and therefore did not change position.

09:42 AOG is bypassed by high flow. High SJAE discharge pressure noted.

After the RFPs tripped, condensate flow to the SJAE intercondenser and aftercooler was virtually eliminated. This prevented condensing of steam and caused the steam to be ejected into the ejector discharge, resulting in the observed increase in flow. The increased flow and moisture from the steam resulted in the observed high SJAE discharge pressure. This was remedied by opening the SJAE minimum flow line which allowed the steam to condense.

09:42:27 In accordance with plant procedures (EOP-1, Flow Path 3), the RO transfers the mode switch from "RUN" to "SHUTDOWN". This action creates the expected manual SCRAM signal in both RPS divisions.

Removing the mode switch from "RUN" also enables the 40% steam flow isolation instrumentation. When steam flow indication is available, the RO verifies that total steam flow is less than 40% prior to removing the mode switch from run. In this case, (ie, without steam flow indication) the operator waited an appropriate amount of time (determined by past experience) prior to moving the mode switch and successfully prevented a main steam line (Group 1) isolation.

A half Group 1 isolation command is generated on channel "A2". Individual flow signals are not recorded, therefore one cannot determine which flow instrument created the isolation command.

09:42:40/46 As the Scram Discharge Volume fills up, the high level rod block as well as the high-high levels for all four RPS sub-channels are generated, per design.

09:46:41 RCIC is started manually and begins to inject. Level decrease is slowed to about 2" per minute. Level is about 125".

During operation of the RCIC system the barometric condenser vacuum pump tripped due to an electrical fault.

09:49:17 HPCI is started manually and begins to inject, level starts to increase at about 1/2" per second. SBGT is also started to support HPCI operation.

Note: A detailed review of the "ERFIS" traces revealed the turbine stop valve opened, closed 8.5 seconds later, stayed closed 11.5 seconds and then re-opened during the HPCI startup. During this time HPCI turbine speed coasted down from approximately 3300 rpm to 1000 rpm and then ramped to rated speed. A review of the ERFIS data, system logic and operating characteristics indicated that the closure was not due to a trip signal, a mechanical overspeed or a balance chamber adjustment problem. It has been determined that the most likely cause was a momentary perturbation of the HPCI oil system pressure downstream of the oil pressure control valve during HPCI start up. This is under investigation.

In addition, a sample of the HPCI oil, taken on 8-17-90, was found to contain approximately 12% water. A previous sample, taken on 7-25-90, contained only 0.25% water. Water intrusion has been a problem in the past and is monitored. The monitoring effort has determined that a significant portion of the problem was corrected by the replacement of the steam supply valves. Since this replacement, significant water intrusion has been identified on two occasions. After this most recent intrusion, troubleshooting was performed which eliminated the possibility of a gross lube oil cooler failure. In addition, Unit 2 HPCI has been operated five times since

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	and the in leakage resulted in less than 1% water. The exact cause of the in leakage has not been determined but an investigation is continuing. Currently this is not an operability concern.
09:49:40	Computer printout gives the appearance that "B2" channel of SDV high high level is alternating between a set and reset state. (Follow-up investigation verified proper operation of both the float and thermal level sensors).
09:51:15	Rx Feedpump "A" begins to feed vessel.
09:51:20/23	Reactor level is above the Level 1 setpoint (166") and continues to increase.
09:51:42	Rx Feedpump "A" speed is decreased to stop vessel feed.
09:51:48	High reactor level alarm is reached (192").
09:51:50	HPCI is manually tripped.
09:53:42	RCIC is manually secured.
09:53:47	High level turbine trip is reached (208").
09:54	RHR loop "A" is placed in torus cooling mode. SDV vent and drain valves verified closed.
09:56	RO bypasses the SDV high-high level and resets the SCRAM.
09:57	Control rods verified full in.
09:58	RHR loop "B" is placed in torus cooling mode.
10:07	SDV "B1" resets.
10:09	SDV "B2" resets.
10:42	Buss "2B" is energized from "SAT". The RR pumps cannot be restarted because the temperature limits for restart are not satisfied. Primary cause is the partial plugging of the bottom head drain.
10:43	SDV "A2" resets.
11:03	Both HPCI and RCIC are restored to "Standby".
11:35	SDV "A1" resets.
11:49	SDV high level keylock bypass is removed.
11:57	Secured the SBGT trains.
12:05	Piping walkdowns on HPCI, RCIC, RHR and the SDV completed. No abnormalities noted.

Temperature Transient Occurrence (Reference Sequence of Events 09:42:11)

On August 16 and 17, 1990, three separate transients that exceeded a 100 degree fahrenheit (F) change in reactor pressure vessel (RPV) temperature in one hour occurred. The first transient involved a cool down and the other two involved a heat up. Contributing to the first transient was a partially plugged bottom head drain line and the tripping of the recirculation pumps which allowed for stratification of the reactor coolant.

Transient Number	Temperature Deg. F Initial	Final	Pressure (PSIG) Initial	Final
1	420	285	825	650
2	110	285	0	56
3	120	250	0	30

As required by Technical Specification 3.4.6.1, General Electric (GE) was requested to perform an evaluation on the fracture toughness properties. GE Service Information Letter (SIL) 430 provides the following guidance to determine adherence to the 100 deg. F/hr requirement: use steam dome temperature (based on pressure) whenever reactor temperature in >212 deg. F and the recirculation suction temperature whenever reactor temperature is <212 deg. F. GE concluded that the 100 degree F/hr requirement limit was not exceeded.

Possible Pin Hole Fuel Leak

In accordance with established procedures, weekly samples of the reactor coolant and off gas are analyzed for the fission products of iodine, krypton and xenon isotopes. In addition, sampling is also carried out whenever a power change of 15% or greater in one hour occurs. Sampling after the 8-16-90 SCRAM indicated a factor of ten increase in the calculated iodine dose equivalent (ie, from 4.413 E-4 to 2.984 E-3 micro curies/milliliter). Subsequent samples taken on 9-6-90 and 9-10-90 have returned to nearly the same iodine dose equivalent that existed on 8-9-90 (ie, 1.881 E-4 and 6.183 E-4 respectively). The graphic representation of the isotopic analysis indicates that a very small leak may exist because the increase has occurred in the xenon and krypton gases but the iodine has essentially remained the same. In addition, subsequent sampling shows that the total micro curie release rate has essentially not changed. If a leak exists it will manifest itself by an increase in release rate after a rod shuffle. Monitoring of the release rate will continue with Chemistry personnel working closely with Nuclear Engineering to determine the location of the leak, if it exists.

Indications of a single failure, small pin hole leak, possibly a very small end well failure, were detected after a rod shuffle on October 11, 1990. Chemistry and Nuclear Engineering personnel are continuing to work together to determine its location and expect to be able to make a determination after March 1991 when core characteristics will be more favorable for finding such a small leak.

EVENT CAUSE

The cause of this event was failure of primary power fuse C32-F5, which supplied power to the steam flow inputs of the three element feedwater control logic. Loss of the steam flow inputs resulted in a maximum demand signal to the RFPs and a rapid increase in reactor level up to the high level turbine trip point which, in turn, caused a reactor scram on TSV position.

CORRECTIVE ACTIONS

Primary power fuse C32-F5 and its associated circuit were evaluated. No cause for the fuse failure was determined. The fuse was replaced and the failed fuse was analyzed by the HE&EC. The draft HE&EC report indicates that the fuse failed due to an actual short duration exposure to a current in excess of 20 amps. HE&EC is continuing its investigation. The Gould Shawmut fuses in the feedwater control circuitry have been replaced temporarily by Bussman Min fuses in accordance with EER 90-0262.

The HPCI turbine stop valve cycling problem and the water intrusion into the HPCI oil are being investigated.

The cause of the RCIC barometric condenser vacuum pump electrical fault has not been determined. The pump has operated satisfactorily since this event.

The GE information provided in the evaluation of the Unit 2 cool down and heat up transients will be reviewed against current practices and procedures.

The reactor recirculation pumps will be powered from the SAT to minimize temperature transients due to loss of the UAT with the reactor vessel bottom drain clogged (a more highly probable event than the loss of the SAT).

Actions pertaining to the bottom head drain clogging have not been determined.

Monitoring of the reactor coolant and off gas release rates will continue in accordance with currently established procedures.

A supplement to this report will be issued by March 15,1991, which will provide the results of the ongoing fuse analysis, investigations and determinations.

EVENT ASSESSMENT

The safety significance of this event is minimal. The plant responded as designed with the exception of the RCIC barometric condenser vacuum pump which is not required for the operation of RCIC.

Past similar events include LERs 2-88-018 and 1-88-023 and a subsequent event was reported in LER 2-90-016.

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SYSTEM/EQUIPMENT ABBREVIATIONS & EIIS CODES

ADS	Automatic Depressurization System	EIIS not found
CS	Core Spray	BM
DG	Diesel Generator	EK
FWLCS	Feedwater Level Control System	JK
HPCI	High Pressure Coolant Injection	BJ
PCIS	Primary Containment Isolation System	JM
RCIC	Reactor Core Isolation System	BN
RHR/LPCI	Residual Heat Removal/Low Pressure Coolant Injection	BO
RFP	Reactor Feed Pump	SJ/P
RPS	Reactor Protection System	JC
SBGT	Standby Gas Treatment System	BH
SLC	Standby Liquid Control	BR
TSV	Turbine Stop Valve	TA/ISV

COMPONENT EIIS CODES

C32-R603 A-D	SB/FI
C32-FR-R607	SB/FR
C32-F5	SB/FU

ABBREVIATIONS

EOP	Emergency Operating Procedures
GP	General Plant Operating Procedure
LL	Low Level
RO	Reactor Operator
SF	Shift Foreman