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FACIL:50-244 Robert Emmet Ginna Nuclear Plant, Unit 1, Rochester G 05000244

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SUBJECT: LER 90-011-00:on 900619,fire damper found missing during Surveillance Test PT-13.26 due to lack of installation.
W/9 ltr.

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July 19, 1990

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

Subject: LER 90-011, Fire damper found missing during surveillance test PT-13.26, due to not being installed, causes a condition prohibited by Technical Specifications.
R.E. Ginna Nuclear Power Plant
Docket No. 50-244

In accordance with 10CFR50.73, Licensee Event Report System, item (a)(2)(i)(B), which requires reporting of "Any Operation or Condition Prohibited by the Plants Technical Specifications", and plant Technical Specifications, section 3.14.6, which requires a 30 day special report, the attached licensee event report LER 90-011 is hereby submitted.

This event has in no way affected the publics health and safety.

Very truly yours

Robert C. Mecredy

Robert C. Mecredy
Division Manager
Nuclear Production

xc: U.S. Nuclear Regulatory Commission
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Ginna USNRC Senior Resident Inspector

P340957258

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NRC Form 366 (9-83)

U.S. NUCLEAR REGULATORY COMMISSION

APPROVED OMB NO. 3150-0104 EXPIRES: 8/31/88

LICENSEE EVENT REPORT (LER)

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TITLE (4) Fire Damper found missing during surveillance test PT-13.26, due to not being installed, causes a condition prohibited by Technical Spec.

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 6	1 9	9 0	9 0	0 1 1	0 0	0 7	1 9	9 0		0 5 0 0 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 9 8

20.402(b)	20.405(c)	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)	X OTHER (Specify in Abstract below and in Text, NRC Form 366A)
20.405(a)(1)(iii)	X 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	
Mark E. Cavanaugh Fire Protection Engineer	3 1 5	5 2 4 - 4 4 4 6

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)

On June 19, 1990 at approximately 1530 EDST with the reactor at approximately 98% full power, a Technical Specification Fire Damper was discovered missing.

The Technical Specification Fire Damper was missing due to not being originally installed.

The underlying cause of the missing fire damper was due to be inadequate design information.

Immediate corrective action was to establish an hourly fire watch to patrol the area.

Subsequent action will be to complete a visual inspection and perform a trip test of all Technical Specification and Appendix R Fire Dampers.

This report is also being submitted under the "Other" category because Technical Specifications also requires a special 30 day written report.

NRC Form 366 (9-83)

I INITIAL PLANT CONDITIONS

The plant was at approximately 98% steady state full power with no major activities in progress.

II DESCRIPTION OF EVENT

A. Dates and approximate times for major occurrences:

- June 19, 1990, 1530 EDST: Event date
- June 19, 1990, 1530 EDST: Discovery date and time
- June 19, 1990, 1600 EDST: Fire protection initiated touring the area once per hour in compliance with Technical Specifications

B. EVENT:

On June 19, 1990 at approximately 1530 EDST with the reactor at approximately 98% full power, Fire Protection and Safety along with HVAC personnel were performing PT-13.26 Testing of Fire Dampers. During the testing of fire damper RR-113-P which is the supply duct for the Mux room HVAC system, it was noted that there was no fire damper in the duct. There are fire dampers in the duct between the Mux room and the Relay room. This duct penetrates the Relay room south wall to a stairwell which leads to the Control room. This duct is identified as being a two (2) hour Technical Specification fire wall.

On June 19, 1990 at approximately 1600 EST, the Fire Protection section assigned a fire watch to perform hourly tours.

C. INOPERABLE STRUCTURES, COMPONENTS OR SYSTEMS THAT CONTRIBUTED TO THE EVENT:

- Missing fire damper RR-113-P

D. OTHER SYSTEMS OF SECONDARY FUNCTION AFFECTED:

None

E. METHOD OF DISCOVERY:

This event was discovered as a result of a pilot test program on fire damper drop testing.

F. OPERATOR ACTION:

The Control room Operators were notified by the Fire Protection Engineer at which time they performed the following:

- Initiated an A-52.4.1 <u>Control of Limiting Conditions for Operating Fire Suppression and Detection Equipment</u>
- Initiated an A-25.1 <u>Ginna Station Event Report</u>
- Notified Duty Engineer
- Notified Shift Technical Advisor (STA)
- Notified Nuclear Regulatory Commission Resident Inspector

G. SAFETY SYSTEM RESPONSES:

None

III <u>CAUSE OF EVENT</u>

A. Immediate cause:

The immediate cause was determined that a fire damper was never installed in a duct which penetrated the two hour fire wall which separated the Relay room from the Control room stairwell.

B. Root cause:

The root cause was determined to be inadequate design information.

IV ANALYSIS OF EVENT

A 1 1/2 hour rated fire damper was not installed in a two hour fire wall so Technical Specification 3.14.6 could not be met. This event is reportable in accordance with 10CFR50.73, Licensee Event Reporting SYstem, item (a) (2) (i) (B), which requires reporting of, "Any Operation or Condition Prohibited by the Plant Technical Specification", and (Other) plant Technical Specifications, section 3.14.6, which requires a 30 day special report.

An assessment was performed considering the safety of this event with the following results and conclusions:

There were no operational or safety consequences of implications attributed to the missing fire damper in the fire barrier wall because:

- The control complex is made up of fire zones AHR, RR and CR. Fire zone AHR was not affected by the missing fire damper. Fire zone RR and CR fire induced effects, safe shutdown methods and proposed compliance methods more similar and are described as follows:

  Reactor makeup capability - a fire in the area may damage control circuitry to 480V AC buses 14 and 16 providing a loss of charging pumps and safety injection pumps. The alternative shutdown capability is provided by the charging pump 1A control circuitry modification. Local transfer, circuit isolation and control of the pump at the charging pump room will be accomplished. Manual valve V-358 providing RWST suction to the charging pump must be manually opened in charging pump room.

  Reactor reactivity control - will be ensured by maintenance of makeup capability as described above.

  Decay heat removal - A fire in this area has the potential for damaging power feeds to both motor-driven AFW pumps, both dc trains A and B 480V ac battery charger feeds, and control circuitry for the standby, turbine-driven and motor-driven AFW pumps. The alternative shutdown capability is provided by operating the turbine-driven AFW pump in the Intermediate Building

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LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

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

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at the pump. The operator will manually isolate dc power to the turbine steam admission valves V-3504A and V-3505A at main dc distribution panels 1A and 1B in fire area BR1A and fire area BR1B. Manual operation of one of the valves will then be accomplished to start the turbine driver. DC power is required for operation of the turbine-driven AFW lube oil pump. Power to this pump is supplied from dc train B through the Turbine building dc distribution panel. A transfer switch will is located in the Intermediate building to isolate control circuits to CT and to allow for local control of the dc lube oil pump.

Manual operation of discharge valves V-3996 and C-4297 (or V-4298) will then be accomplished to permit AFW flow to the appropriate steam generator. As before, power will be isolated or instrument air isolated and bled off before manual operation is attempted.

Long term decay heat removal will be accomplished by use of the RHR system. RHR pump control circuits may be damaged by a fire in the area. Adequate time exists to either repair damaged control circuits or to locally control RHR pump breakers. This analysis assumes that repairs will not be made and that the breakers will be operated locally.

Reactor Pressure Control - pressure control is ensured by automatic operation of pressurizer safety valves and by maintenance of makeup capability as described above.

Support systems - Service water and CCW pump control may be affected by a fire in these areas. Emergency diesel generator control may be affected by a fire in theses areas. However, alternative shutdown for EDG 1A will allow for complete isolation of all necessary control circuits from the Control and Relay room. EDG cooling can be accomplished by manually operating the service water pump breakers in the screenhouse or via the underground yard fire water system. A local EDG connection exists that allows for a local water supply hook up in case service water is not available. CCW pump control is necessary for cold shutdown. Adequate time exists to allow for local control of CCW pump breakers.

Process Monitoring - A fire in this area could damage all required primary and secondary plant instrumentation circuits. Alternative shutdown modifications have been made to ensure that required process variables are always provided to the plant operators. An instrument panel (IBELIP) has been installed in the Intermediate building north area on elevation 253 ft near the turbine-driven AFW pump to provide the following indications:

(1) Primary temperature - RCS loop a hot and cold leg (TI-409A-2, B-2)

(2) Steam generator 1A wide-range level (LI-460A),

(3) Steam generator 1A pressure indication (PI-469A), and

(4) Turbine-driven AFW flow indication (FI-2015A).

With this instrumentation, along with operation of the turbine-driven AFW pump, the operator will be able to adequately control the decay heat function after a fire. Power to this instrument panel is supplied from a breaker in the Turbine building dc distribution panel feeding an invertor.

A second panel (ABELIP) has been installed in the Charging pump room to provide the required primary indications for control of the primary makeup function. This panel will provide the following primary indications:

(1) Primary pressure indication (PI-420B), and

(2) Pressurizer level indication (LI-428A).

Power to this panel is supplied by an invertor supplied from a breaker at the Auxiliary building dc distribution panel 1A2.

Since the train A battery charger power feed may be damaged, the train A battery may become depleted eight hours or more after the fire event. An intertie exists between the TSC battery charger (supplied from the TSC

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diesel generator) and main fuse cabinet 1A (train A dc power). Use of this existing intertie will provide for long-term operation of process monitoring instrumentation located at the ABELIP.

Both redundant trains of source range neutron monitoring may be damaged by a fire in this area. A spare monitor drawer is located on site to provide the capability to bypass fire damaged circuits. A procedure is used to connect the existing neutron monitor inside the containment to the new drawer. A local power supply will be used to power the new drawer.

Based on the above and RG&E Ginna Station Appendix R Alternative Shutdown Report, it can be concluded that the employee's and the public's health and safety was assured at all times.

V CORRECTIVE ACTIONS

A. Actions taken to return inoperable components to operable status:

- An hourly fire watch was established to patrol the area once per hour

- An evaluation will be conducted to determine a fire damper rating and size

- A new fire damper will be installed in the duct to bring the fire barrier in compliance with a rated barrier

B. ACTIONS TAKEN OR PLANNED TO PREVENT RECURRENCE:

- Complete a visual inspection and perform a trip test of all Technical Specification and Appendix R fire dampers

VI ADDITIONAL INFORMATION

A. Failed components

None

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

B. Preview LER's on similar events

A similar LER event historical search was conducted with the following results: no documentation of similar LER events with the same root cause at Ginna Station could be identified

C. Special Comments

None