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ACCESSION NBR:9507120033 DOC.DATE: 95/07/07 NOTARIZED: NO DOCKET #
FACIL:50-244 Robert Emmet Ginna Nuclear Plant, Unit 1, Rochester G 05000244
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RECIP.NAME RECIPIENT AFFILIATION

JOHNSON,A.R. Project Directorate I-1 (PD1-1) (Post 941001)

SUBJECT: LER 95-005-00:on 950607,FW isolation on high SG level occurred.Caused by decrease in instrument air pressure due to an air leak in containment.FW flow switched to manual control.W/950707 ltr.

DISTRIBUTION CODE: IE22T COPIES RECEIVED:LTR 1 ENCL 1 SIZE: 9
TITLE: 50.73/50.9 Licensee Event Report (LER), Incident Rpt, etc.

NOTES:License Exp date in accordance with 10CFR2,2.109(9/19/72). 05000244

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ROBERT C. MECREDY
Vice President
Nuclear Operations

July 7, 1995

U.S. Nuclear Regulatory Commission
Document Control Desk
Attn: Allen R. Johnson
PWR Project Directorate I-1
Washington, D.C. 20555

Subject: LER 95-005, Instrument Air Leak in Containment Causes Feedwater Isolation
R.E. Ginna Nuclear Power Plant
Docket No. 50-244

In accordance with 10 CFR 50.73, Licensee Event Report System, item (a) (2) (iv), which requires a report of, "Any event or condition that resulted in a manual or automatic actuation of any engineered safety feature (ESF), including the reactor protection system (RPS)", the attached Licensee Event Report LER 95-005 is hereby submitted.

This event has in no way affected the public's health and safety.

Very truly yours,

Robert C. Mecredy

Robert C. Mecredy

xc: U.S. Nuclear Regulatory Commission
Mr. Allen R. Johnson (Mail Stop 14B2)
PWR Project Directorate I-1
Washington, D.C. 20555

U.S. Nuclear Regulatory Commission
Region I
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Ginna USNRC Senior Resident Inspector

IE22

NRC FORM 366 (5-92) U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB NO. 3150-0104 EXPIRES 5/31/95
LICENSEE EVENT REPORT (LER) (See reverse for required number of digits/characters for each block)	ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS INFORMATION COLLECTION REQUEST: 50.0 HRS. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE INFORMATION AND RECORDS MANAGEMENT BRANCH (MNBB 7714), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555-0001, AND TO THE PAPERWORK REDUCTION PROJECT (3150-0104), OFFICE OF MANAGEMENT AND BUDGET, WASHINGTON, DC 20503.

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TITLE (4) Instrument Air Leak in Containment Causes Feedwater Isolation

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
06	07	95	95	--005--	00	07	07	95	FACILITY NAME	DOCKET NUMBER

OPERATING MODE (9)	N	THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more) (11)							
			20.402(b)		20.405(c)	X	50.73(a)(2)(iv)		73.71(b)
POWER LEVEL (10)	097		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
			20.405(a)(1)(iii)		50.73(a)(2)(i)		50.73(a)(2)(viii)(A)		(Specify in Abstract below and in Text, NRC Form 366A)
			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 (Include Area Code)
John T. St. Martin - Technical Assistant	(716) 771-3641

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
B	LD	PSF	0000	N						

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)

On June 7, 1995 at approximately 1905 EDST, with the plant at approximately 97% steady state reactor power and the Instrument Air system isolated to Containment due to an air leak, Feedwater Isolation on high Steam Generator level occurred when levels went above 67% narrow range level in the Steam Generators.

Immediate corrective action was to manually control feedwater flow until levels in the Steam Generators were restored to their normal operating band.

The underlying cause of the inability to control Steam Generator levels was a decrease in Instrument Air pressure due to an Instrument Air leak in Containment, followed by restoration of air pressure with a demand signal to fully open main feedwater regulating valves.

This event is NUREG-1022 Cause Code (B).

Corrective action to preclude repetition is outlined in Section V.B.

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

I. PRE-EVENT PLANT CONDITIONS:

The plant was at approximately 97% steady state reactor power with no significant activities in progress. A soldered joint connection in the Instrument Air system in Containment failed, causing a decrease in Instrument Air pressure, and loss of control air to air-operated components, including the main feedwater regulating valves.

II. DESCRIPTION OF EVENT:

A. DATES AND APPROXIMATE TIMES OF MAJOR OCCURRENCES:

- June 7, 1995, 1856 EDST: Instrument Air system leak occurs in Containment.
- June 7, 1995, 1902 EDST: Instrument Air to Containment is isolated, restoring normal air pressure to air-operated components outside Containment.
- June 7, 1995, 1905 EDST: Steam Generator (S/G) levels increase above 67%. Event date and time.
- June 7, 1995, 1905 EDST: Discovery date and time.
- June 7, 1995, 1910 EDST: "A" and "B" S/G levels restored to pre-event normal operating band.

B. EVENT:

On June 7, 1995, at approximately 1856 EDST, with the plant at approximately 97% steady state reactor power, a soldered joint connection on a two inch Instrument Air (IA) line in Containment (CNMT) failed, resulting in leakage from the IA system and decrease in IA pressure. This decrease in IA pressure resulted in loss of control air to air-operated components, with valves beginning to travel to their respective "fail" positions. Among these components were the two main feedwater regulating valves (MFRV) (fail closed) which drifted towards the closed position as IA pressure at the valve actuator decreased.

At approximately 1902 EDST, the Control Room operators had diagnosed the probable location of the IA leak, and closed the IA CNMT Isolation valve, AOV-5392. With the closure of AOV-5392, the leak was isolated, and normal IA pressure was restored to components outside CNMT.

Due to the MFRVs drifting closed, feedwater (FW) flows and Steam Generator (S/G) levels decreased, resulting in an increasing "demand" signal to the MFRVs. Isolation of the IA leak resulted in restoration of IA pressure, and the MFRVs opened fully, responding to the increased demand signal. At the time the MFRVs went to the full open position, level was approximately 25% in the "A" S/G and 40% in the "B" S/G. The increase in FW flow resulted in increasing level in the "A" and "B" S/Gs.

Within three minutes narrow range level in the "B" S/G had increased to cause FW Isolation on high level in the "B" S/G (S/G level >/= 67 % narrow range level). The "B" MFRV closed in response to this FW Isolation signal as designed, and reopened when level decreased below 67%. For the next ninety seconds, there were several occurrences of FW Isolation for the "B" S/G as level cycled around 67%. During this time narrow range level in the "A" S/G also increased to cause FW Isolation on high level in the "A" S/G. For approximately twenty seconds, there were occurrences of FW Isolation for the "A" S/G as level cycled around 67%. This short term S/G level transient continued until the Control Room operators took manual control to restore S/G levels. At approximately 1910 EDST, levels in the "A" and "B" S/Gs were restored to their normal operating band.

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

The decrease in IA pressure resulted in loss of control air to air-operated components.

D. OTHER SYSTEMS OR SECONDARY FUNCTIONS AFFECTED:

Due to the failed soldered joint connection in CNMT, and the subsequent isolation of IA to CNMT, air-operated components in CNMT failed to their respective "fail" positions. These included several valves and ventilation dampers. In addition, the Reactor Compartment Cooling (RCC) fan motor tripped when the associated dampers failed closed.

For air-operated components outside the CNMT, there was a decrease in IA pressure throughout the plant for six minutes, until IA to CNMT was isolated. During this time, numerous air-operated components outside CNMT started to travel to their respective "fail" positions. With the exception of the MFRVs, this loss of control air did not adversely affect the ability of the Control Room operators to maintain plant conditions.

E. METHOD OF DISCOVERY:

This event was immediately apparent due to alarms and indications in the Control Room. In particular, Main Control Board annunciators C-17 (CONTAINMENT VENT SYSTEM) and H-8 (INSTRUMENT AIR LO PRESS 100 PSI) alarmed, indicating a problem with IA in CNMT.

F. OPERATOR ACTION:

The Control Room operators responded to Main Control Board annunciators C-17, H-8 and H-16 (INSTRUMENT AIR COMP), and referred to Alarm Response Procedures C-17, H-8 and H-16. They entered Abnormal Operating Procedure AP-IA.1 (LOSS OF INSTRUMENT AIR). The Control Room operators requested that the auxiliary operator start the standby diesel-driven air compressor. Following the steps of AP-IA.1 and with the knowledge that abnormal alarms were received on CNMT systems prior to those on secondary systems, the Control Room operators isolated IA to CNMT by closing the IA CNMT Isolation valve (AOV-5392). This action isolated the leak from the rest of the IA system, and IA pressure increased to normal pressure in the rest of the system.

After the FW Isolation, the Control Room operators transferred control of the MFRVs to "manual" to restore S/G levels to their normal operating band. When S/G levels and FW flows were stabilized, they transferred control of the MFRVs back to "automatic".

With loss of letdown flow, the operators manually decreased charging flow to minimum flow, and secured one charging pump. The Shift Supervisor made a decision to initiate a power reduction until the leak was located, isolated, and IA pressure returned to normal throughout the system. At approximately 1908 EDST, a power reduction was started at one percent per minute, per Normal Operating Procedure O-5.1 (LOAD REDUCTIONS).

Subsequently, the Control Room operators notified maintenance personnel and higher supervision.

An auxiliary operator and Radiation Protection technician conducted a CNMT entry at power in an attempt to identify and isolate the leak. The leak was located on the main two inch IA header in CNMT. A temporary repair was made to the failed joint connection. This repair enabled the Control Room operators to restore some pressure to the IA system in CNMT, sufficient to allow operation of selected valves and ventilation dampers.

The NRC Operations Center was notified at approximately 2211 EDST, per 10CFR50.72 (b) (2) (ii), non-emergency four hour notification.

G. SAFETY SYSTEM RESPONSES:

The MFRVs and MFRV bypass valves closed automatically as a result of the FW Isolation signals. Ventilation dampers for the Containment Recirculation Cooling fans and RCC fans failed to their respective safeguards positions. CNMT Isolation valves for charging and letdown also failed to their safeguards positions.

III. CAUSE OF EVENT:

A. IMMEDIATE CAUSE:

The immediate cause of the FW Isolation was level in the S/Gs being >/= 67%. The high level was caused by increased FW flows when the MFRVs went full open in response to the valve demand signal. This situation resulted in overfeeding the S/Gs.

B. INTERMEDIATE CAUSE:

The intermediate cause of the open valve demand signal for the MFRVs was decreased FW flows and S/G levels as the MFRVs drifted toward the closed position as IA pressure decreased.

C. ROOT CAUSE:

The underlying cause of the decrease in IA pressure was the failure of a soldered joint connection in a two inch IA line in CNMT. This was caused by insufficient insertion of the pipe into a fitting during original construction. This event is NUREG-1022 Cause Code (B), "Design, Manufacturing, Construction / Installation". This event does not meet the NUMARC 93-01, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants", definition of a "Maintenance Preventable Functional Failure".

IV. ANALYSIS OF EVENT:

This event is reportable in accordance with 10 CFR 50.73, Licensee Event Report System, item (a) (2) (iv), which requires a report of, "Any event or condition that resulted in a manual or automatic actuation of any engineered safety feature (ESF), including the reactor protection system (RPS)". The FW Isolation of the "A" and "B" S/Gs was an automatic actuation of an ESF system.

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

There were no operational or safety consequences or implications attributed to the FW isolations because:

o The FW isolations occurred at the required S/G level.

o S/G levels were quickly stabilized and manual control of MFRVs was accomplished to mitigate any consequences of the event.

Based on the above, it can be concluded that the public's health and safety was assured at all times.

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

V. CORRECTIVE ACTION:

A. ACTION TAKEN TO RETURN AFFECTED SYSTEMS TO PRE-EVENT NORMAL STATUS:

- o MFRVs were returned to automatic after S/G levels were restored to their pre-event normal operating band.

- o A temporary repair was made to the failed joint connection in CNMT. This temporary repair enabled the Control Room operators to restore some pressure to the IA system in CNMT, sufficient to allow operation of selected valves and ventilation dampers.

- o After letdown flow was restored and the Control Room operators could control primary system volume, the load reduction was stopped.

- o Maintenance personnel installed a temporary modification designed by Engineering, which permitted isolation of the failed joint for permanent repair, while maintaining an air supply to the letdown valves. This allowed the operators to maintain letdown flow.

- o Maintenance personnel performed the permanent repair by replacing the failed joint connection and adjacent pipe sections, and removed the temporary modification. At the completion of these activities, normal IA was restored to CNMT.

B. ACTION TAKEN OR PLANNED TO PREVENT RECURRENCE:

- o A sample of joint connections in the IA system will be examined by non-destructive techniques to confirm adequate pipe insertion into fittings.

- o This event will be evaluated and compared against Plant Simulator response under controlled conditions. Any lessons learned and enhancements to the control of primary system pressure will be identified, and procedures changed, as appropriate.

LICENSEE EVENT REPORT (LER)
TEXT CONTINUATION

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

VI. ADDITIONAL INFORMATION:

A. FAILED COMPONENTS:

There were no component failures, in that the leak occurred when a soldered joint connection failed. This joint connected a two inch copper pipe to a two inch copper elbow fitting. The manufacturer of the pipe and fitting is not relevant, and the manufacturer of the solder is unknown.

B. PREVIOUS LERs 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 Nuclear Power Plant could be identified.

C. SPECIAL COMMENTS:

None