NRC FORM 366 (6-89) U.S. NUCLEAR REGULATORY COMMISSION

APPROVED OMB NO. 3150-0104
EXPIRES: 4/30/92

ESTIMATED BURDEN PER RESPONSE TO COMPLY WTH 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.

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
R.E. Ginna Nuclear Power Plant	0 5 0 0 0 2 4 4	1 OF 1 1

TITLE (4) Feedwater Control Perturbations, Due to a Secondary Transient, Causes Steam Generator High Level Feedwater Isolations

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 5	1 8	9 2	9 2	— 0 0 6	— 0 1	0 7	2 9	9 3		0 5 0 0 0

OPERATING MODE (9)	N	THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more of the following) (11)			
POWER LEVEL (10)	0 9 7	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
Wesley H. Backus Technical Assistant to the Operations Manager	AREA CODE 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	MANUFAC-TURER	REPORTABLE TO NPRDS		CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	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 May 18, 1992 at approximately 1339 EDST, with the Reactor at approximately 97% full power, Main Feedwater Isolations occurred on the "A" Steam Generator (S/G). These feedwater isolations were caused by overfeeding the "A" S/G while in manual control, following a secondary system pressure transient.

Immediate operator action was to manually control the Main Feedwater Regulating Valves (FRV) to reduce the steam generator water levels and stabilize the plant.

The immediate cause of the event is due to a secondary system pressure transient.

The root cause of the event has been determined to be the Heater Drain Tank Level Control System. (This event is NUREG-1022 (X) cause code.)

Corrective actions taken or planned are discussed in Section V of the text.

I. PRE-EVENT PLANT CONDITIONS

The plant was at approximately 97% steady state power with no major activities in progress. Instrument and Control (I&C) Department personnel were investigating the Advanced Digital Feedwater Control System (ADFCS) due to some feedwater oscillations that had occurred at approximately 1110 EDST, May 18, 1992. These feedwater oscillations were being caused by the "A" and "B" Steam Generator (S/G) Main Feedwater Regulating Valves (FRVs) cycling (between 43% and 48%), at twice their normal frequency and in opposite phase from each other. This feedwater response (instability while in automatic control) was new and unexpected by the operators.

II. DESCRIPTION OF EVENT

A. DATES AND APPROXIMATE TIMES OF MAJOR OCCURRENCES:

- May 18, 1992, 1337 EDST: A Secondary Side Transient occurs.
- May 18, 1992, 1339 EDST: Event Date and Approximate Time.
- May 18, 1992, 1339 EDST: Discovery Date and Approximate Time.
- May 18, 1992, 1342 EDST: Control Room operators restore S/G levels to normal and place Main FRVs in automatic.

B. EVENT:

On May 18, 1992, at approximately 1337 EDST, with the Reactor at approximately 97% full power, a secondary side condensate system decreasing pressure transient occurred. This pressure decrease traveled throughout the entire secondary side and was reflected as low feedwater header pressure and corresponding decrease in differential pressure across both FRVs. The ADFCS responded to the low feedwater header pressure by increasing the open demand signal to the main FRVs in order to maintain the proper feedwater flow. At this same time, the "C" Condensate Pump automatically started on low condensate system header pressure and combined with a condensate header pressure low alarm, alerted the Control Room operators to an abnormal condition. The Control Room operators, after observing that the main FRVs open demand signals were higher than normal and increasing, and that feedwater flow to the "B" S/G was higher than appropriate, took manual control of the Main FRVs to prevent further opening of the valves.

An increased feedwater header pressure, caused by the auto start of the "C" Condensate Pump, coupled with manual operation of the Main FRVs, resulted in momentary overfeeding of the "A" S/G after the "B" FRV was throttled. At 1339 EDST, May 18, 1992, Main Feedwater Isolation on high level (i.e. >/= 67% narrow range level) occurred three (3) times to the "A" S/G over a period of ten (10) seconds. During this time period, the operator manually secured the "C" condensate pump to decrease feedwater header pressure.

The Control Room operators manually restored S/G levels to normal, and at approximately 1342 EDST restored the "A" and "B" Main FRVs to automatic control.

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

None.

D. OTHER SYSTEMS OR SECONDARY FUNCTIONS AFFECTED:

None.

E. METHOD OF DISCOVERY:

The event was immediately apparent due to alarms and indications in the Control Room.

F. OPERATOR ACTION:

The Control Room operators took immediate manual actions to control S/G levels and stabilize the plant. Subsequently, the Control Room operators notified higher supervision and the Nuclear Regulatory Commission per 10 CFR 50.72, Non-Emergency, 4 hour Notification.

G. SAFETY SYSTEM RESPONSES:

The "A" S/G Main FRV began to close automatically as a result of the feedwater isolation signal. Due to the short duration that the signal was present, the valve never fully closed.

III. CAUSE OF EVENT

A. IMMEDIATE CAUSE:

The feedwater isolation signal to the "A" S/G Main FRV was due to the "A" S/G narrow range level being >/= 67%.

B. INTERMEDIATE CAUSES:

The "A" S/G narrow range level was >/= 67% due to increased flow to the "A" S/G caused by the perturbations in main feedwater header pressure and manual FRV operation. This situation resulted in inadvertent overfeeding of the "A" S/G. It should also be noted here that just prior to this event, feedwater automatic control had been experiencing instability with flow oscillations and the operators were sensitive to abnormal indication with ADFCS in automatic control of the FRVs.

The perturbations in main feedwater header pressure were determined to be caused by a large reduction of Heater Drain Tank (HDT) pump flow. Since HDT pump flow comprises approximately one-third of the feedwater flow, this reduction caused available feedwater flow to the S/Gs to be reduced by approximately 16%. The ADFCS system was responding to this transient when Control Room operators took manual control of the main FRVs. This reduction in HDT pump flow scenario was tested on the Ginna Simulator and the results were similar to the observed plant response.

C. ROOT CAUSE:

After extensive investigation the underlying cause of the reduction of HDT pump flow was determined to be a HDT Level Control System problem. Instrument and Control (I&C) technicians found a valve mispositioned which cross connected the existing Level Control System with the old Pneumatic Control System. This cross connect valve allows the old Pneumatic System to be used as a backup system. It was determined that cross connecting these control systems could produce unpredictable control perturbations such as was seen during this event. The HDT Level Control System stabilized following the closing of the before mentioned cross connect valve.

IV. ANALYSIS OF EVENT

This event is reportable in accordance with 10CFR50.73, Licensee Event Report system, item (a)(2)(iv), which requires reporting of, "any event or condition that resulted in manual or automatic actuation of any Engineered Safety Feature (ESF) including the Reactor Protection System (RPS)". The feedwater isolation of the "A" S/G 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 feedwater isolations because:

- The feedwater isolation signals occurred at the required S/G level.
- The plant was quickly stabilized and manual control of FRVs was accomplished to mitigate any consequences of the event.
- As the feedwater isolation signals occurred as designed, the assumptions of the FSAR for steam line break were met.

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

V. CORRECTIVE ACTION

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

- After S/G levels were stabilized, the Main FRVs were returned to automatic control.

B. ACTION TAKEN OR PLANNED TO PREVENT RECURRENCE:

To prevent recurrence of inadvertently cross connecting the existing Level Control System to the old Pneumatic System, administrative controls have been instituted to ensure this valve is not opened when the existing Level Control System is to be used for control.

After reviewing the results of troubleshooting and discussions with Westinghouse, the following is an outline of the corrective actions being taken or planned in response to the feedwater control instability problems (and ADFCS transient response for secondary system transient):

- Short Term Response:

  a) I&C technicians loosened and lubricated the actuator to stem connection block for the "B" FRV.

  b) Operations was directed to place the Main FRV bypass valves in full open position and in automatic control, to reduce the feedwater control instability problem.

  c) The packing on both FRVs was relaxed to minimum torque, to reduce any potential friction of the valve stem.

- Intermediate Term Response:

  a) Westinghouse has evaluated ADFCS "tuning constants" and has supplied new ADFCS gain values, in an attempt to reduce the ADFCS impact on feedwater control instability problems.

  b) The new ADFCS gain values have been installed in the ADFCS software. Instability problems have not been eliminated as a result of these new gain values.

c) Diagnostic testing of FRV performance was completed using the Fisher Flowscanner system. Testing focused on valve response to changes in current-to-pneumatic (I/P) transducers, valve positioners, and solenoids and associated instrument air tubing.

o Long Term Response:

a) Valve characteristic data (with and without valve shaft packing in place) will be reviewed.

Status: Complete

b) The vendors of the Main FRV actuator and positioner will be consulted, requesting their recommendations for valve packing and positioner operation, based on a more detailed review of the diagnostic testing.

Status: Complete

c) The calibration procedures for the Main FRVs will be revised to include valve response characteristics during scheduled calibrations.

Status: In Process

d) Engineering Work Request (EWR) 4773 will apply the information obtained from other actions, and is intended to enhance feedwater control and response to plant transients. It is envisioned that improvements will be obtained in both electronic and pneumatic performance, as a result of these enhancements.

Status: Complete

During the 1993 Annual Outage, subsequent start-up and power operation to date, the following actions and results were accomplished:

- The Fisher Flowscanner system was used on the FRVs to establish "As Found" conditions.

- EWR 4773 was performed to improve manual control of the FRVs by decreasing the manual demand signal response time. Also as part of this EWR, the existing FRV fail open logic was deleted and the FRV fail closed logic was replaced with actuation upon reactor trip as opposed to turbine trip to allow automatic control of the FRVs following a turbine trip only.

- EWR 4894 was performed for replacement of the FRV Solenoid Operated Valves (SOV) with new SOVs that have larger diameter internal parts that will permit faster FRV closing times upon receipt of a feedwater isolation signal.

- The "A" Main FRV was disassembled, inspected, refurbished (i.e. lapped, polished stem), reassembled, and repacked. Packing gland was torqued to the specified value.

- The "B" Main FRV was disassembled, inspected, refurbished (i.e. replaced plug, stem, connecting pin; lapped), reassembled and repacked. Packing gland was torqued to the specified value. The actuator was also replaced.

- After all maintenance was completed on the FRVs, I&C set up the Fisher Flowscanner system and obtained valve characteristic data.

- On the subsequent startup and return to power, the "A" Main FRV showed feedwater control instabilities and the following actions were taken to reduce these instabilities:

  1. Re-tuning ADFCS

  2. Adjusting valve packing

  3. Adjusting snubber on valve actuator

  These actions did not improve our ability to control the instabilities.

Based on the above actions, combined with further testing and assessment, it has been determined that the ADFCS system is performing its intended function.

- Based on the above determination, the following actions are planned as plant conditions permit to enhance system stability control:

  1. Replace the "A" FRV actuator with the rebuilt one.

  2. Replace the "A" FRV positioner

  3. Repack both FRVs using a lower signature torque value

- Longer term corrective action is being evaluated to determine if an upgrade to FRV internals is warranted.

VI. ADDITIONAL INFORMATION

A. FAILED COMPONENTS:

None.

B. PREVIOUS LERs ON SIMILAR EVENTS:

A similar LER event historical search was conducted with the following results: LER 91-009 was a similar event with a different root cause.

C. SPECIAL COMMENTS:

None.