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January 9, 1991

W. T. Cottle
Vice President
Nuclear Operations

U.S. Nuclear Regulatory Commission
Mail Station P1-137
Washington, D.C. 20555

Attention: Document Control Desk

SUBJECT: Grand Gulf Nuclear Station
Unit 1
Docket No. 50-416
License No. NPF-29
Automatic Scram Due To Instrument Air System
Piping Joint Failure
LER 90-028

GNRO-91/00009

Gentlemen:

Attached is Licensee Event Report (LER) 90-028 which is a final report.

Yours truly,

W T Cottle

WTC/BAB/cg
attachment

cc: Mr. D. C. Hintz (w/a)
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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)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Grand Gulf Nuclear Station - Unit 1	0 5 0 0 0 4 1 6	1 OF 0 4

TITLE (4): Automatic Scram Due To Instrument Air System Piping Joint Failure

EVENT DATE (5) MONTH	DAY	YEAR	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	REPORT DATE (7) MONTH	DAY	YEAR	OTHER FACILITIES INVOLVED (8) FACILITY NAMES	DOCKET NUMBER(S)
1 2	1 0	9 0	9 0	0 2 8	0 0	0 1	0 9	9 1	NA	0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9): 1

POWER LEVEL (10): 1 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)	X 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	
Bruce A. Burke / Licensing Engineer	6 0 1	4 3 7 - 6 3 3 3

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)

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)

Actuation of Engineered Safety Features occurred on December 10, 1990, including the reactor protection system (RPS) and the emergency core cooling system (ECCS) high pressure core spray (HPCS) system. The subsequent reactor scram from full power was caused indirectly by a failed solder joint in the instrument air system. This report is being submitted pursuant to 10CFR50.73(a)(2)(iv) and plant Technical Specification 3.5.1 Action Item h.

The failed pipe joint was investigated. A leaking root valve apparently caused a slight pressure differential across the joint and precluded optimum capillary action during the soldering process. It is believed that vibrations in combination with system pressure caused the resultant separation of this inadequately soldered joint. This joint had been soldered on November 9, 1990. Applicable plant administrative procedures will be revised to preclude inadequate welding and similar failures.

The actuation of the RPS and the resultant automatic scram occurred as designed. Actuation of the ECCS initiated as designed by injecting condensate from the condensate storage tank into the reactor vessel via HPCS. The safety of the general public was not compromised by this event.

A. Reportable Occurrence

Actuation of the reactor protection system (RPS) initiated an automatic reactor scram on December 10, 1990 due to a Level 3 reactor water condition. Actuation of the emergency core cooling system (ECCS) also occurred; the high pressure core spray (HPCS) system injected into the reactor vessel due to a Level 2 reactor water condition. This report is being submitted as Engineered Safety Feature (ESF) actuations pursuant to 10CFR50.73(a)(2)(iv) and plant Technical Specification 3.5.1 Action Item h. Initial notification was made on December 10, 1990 in accordance with 10CFR50.72(b)(1)(iv).

B. Initial Conditions

The plant was in Operational Condition 1, power operation, with reactor water at approximately 530 degrees F and 1034 psig.

C. Description of Occurrence

On December 10, 1990 at approximately 1033 hours while operating at full power, actuation of the RPS (EIIS Code: JC) occurred. The resultant automatic reactor scram was initiated when both RPS divisions were tripped due to low reactor water level. The scram was caused indirectly by a failed solder joint in the instrument air system (IAS) (EIIS Code: LD) piping.

Upon failure of the piping joint, IAS pressure decreased rapidly. The loss of air pressure caused the level control valves for the feedwater heaters, moisture separator drain tanks, and the heater drain tank to fail close and their dump valves to fail open. This subsequently caused the heater drain pumps to trip. The loss of IAS pressure also caused both feedwater pump minimum flow control valves to fail open which diverted feedwater flow to the condenser. This decreased the feedwater flow to the reactor vessel and increased condensate system flow. The high condensate system (EIIS Code: SD) flow rate induced a low suction pressure trip of one condensate booster pump. Both main feedwater system (EIIS Code: SJ) pumps then tripped on low suction pressure and the reactor water level decreased rapidly. At +11.4 inches on the narrow range instrumentation (i.e., Level 3), the reactor scrammed automatically as designed. All four RPS channels of reactor water level had been tripped.

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

Manual initiation of the reactor core isolation cooling (RCIC) system (EIIS Code: BN) was in progress to stabilize reactor water level as level approached -41.6 inches as indicated on the wide range instrumentation (i.e., Level 2). The level diminished to Level 2 before RCIC flow was established and automatic system actions occurred as designed, including initiation of HPCS (EIIS Code: BG), isolation of the reactor water cleanup (RWCU) system (EIIS Code: CE), and tripping of both reactor recirculation system (EIIS Code: AD) pumps. Reactor water level was restored and stabilized using HPCS, RCIC, and the feedwater system.

D. Apparent Cause

The failed pipe joint was investigated. This joint had been soldered on November 9, 1990 during the refueling outage as part of a modification to install particulate filters in individual IAS branches. Nondestructive examination of the joint had been performed as part of the post-installation testing; snoop testing had indicated an acceptable joint.

That investigation determined that the failed joint had been soldered inadequately. A leaking root valve apparently caused a slight pressure differential across the joint and precluded optimum capillary action during the soldering process. The flow of solder between mating pieces did not form a complete bond. It is believed that vibrations in combination with system pressure caused the resultant separation of this inadequately soldered joint.

E. Supplemental Corrective Actions

All soldered piping joints which were part of the recent modification were reinspected as a result of the failed joint. Ultrasonic testing (UT) was performed on these joints after the UT method of inspection was evaluated and accepted for these soldered copper piping joints. One additional soldered joint was defective as detected by an audible leak prior to the UT examination of that joint. This joint leak was not sufficient to contribute to the loss of IAS pressure. All other soldered joints inspected were acceptable as indicated by the UT method. Satisfactory repairs were made on both defective joints with the piping system vented.

NRC Form 366A
(9-83)
U.S. NUCLEAR REGULATORY COMMISSION
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LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

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

Applicable plant administrative procedures will be revised to ensure that a suitable vent path is provided, as appropriate, to the affected piping or component to preclude inadequate welding and similar failures. These procedures will also be revised to require reporting to the responsible welding engineer of any abnormality (including fluid flow through the joint) which occurs during any welding, soldering, or brazing process. All solder welders will be instructed that any fluid flow (air or liquid) through the solder joint must be reported to the responsible welding engineer prior to making the joint.

F. Safety Assessment

The actuation of the RPS and the resultant automatic scram occurred as designed. Actuation of the ECCS initiated as designed by injecting condensate from the condensate storage tank into the reactor vessel via HPCS. Reactor water level decreased to -72 inches as indicated on the wide range instrumentation which corresponds to approximately 94 inches of water above the core. Eight HPCS system actuation cycles have occurred with elevated reactor temperature including this event. The safety of the general public was not compromised by this event.

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