NRC Form 366 (9-83)

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

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): Inadvertent Control Rod Withdrawal

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 7	3 0	8 6	8 6	— 0 2 6	— 0 0	0 8	2 2	8 6	NA	0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9): 1

POWER LEVEL (10): 0 6 9

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)	[ ] 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)	Voluntary Special Report
[ ] 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)
Ronald Byrd/Licensing Engineer	6 0 1 4 3 7 - 2 1 4 9

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
Y	A A	F S V	A 6 1 0	N					

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)

On July 30, 1986, a single control rod continued drifting out of the core after receiving a single notch withdrawal command. The rod failed to insert on demand and continued to withdraw to the full-out position. Inspections and bench checks were performed on the withdraw directional control valve. It is concluded that temporary particulate accumulation on the valve seating surface caused an incomplete closure of the valve when the withdraw command was terminated, allowing drive water pressure to leak past the valve and force the drive piston downward. The valve was replaced and the Off-Normal Event Procedure was changed to provide specific operator instructions in this situation. General Electric Service Information Letter (SIL) #292 provides additional details of such occurrences.

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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)

A. REPORTABLE OCCURRENCE

On July 30, 1986, a single control rod continued drifting out of the core after receiving a single notch withdrawal command. The rod failed to insert on demand and continued to withdraw to the full-out position. This report is submitted as a voluntary report. No safety function was affected and no safety limits were exceeded.

B. INITIAL CONDITIONS

The plant was operating in Mode 1 at approximately 69 percent power.

C. DESCRIPTION OF OCCURRENCE

At 2143 on July 30, 1986, an operator withdrew rod 20-45 one notch from notch position 08 to notch position 10 (notch positions are even numbered). The "Rod Drift" and "Rod Block" alarms occurred and the operator observed that rod 20-45 was at notch position 12 and continuing to withdraw. The operator pressed the insert pushbutton several times and observed the "in" light and the "settle" light to illuminate. The repeated notch insertion attempts slowed the rod outward movement, but the rod continued to withdraw to the full-out position at notch position 48. It is estimated that the control rod took 3 minutes and 10 seconds to travel from position 08 to position 48.

The operator carried out the actions required by the Alarm Response Instructions. As a conservative measure, reactor power was reduced to 60 percent for thermal limit concerns, and a coupling check was performed. Once control of rod insertion was regained, the rod was placed at position 44 and back to position 48 to test the Rod Withdrawal Limiter. The rod was declared inoperable, fully inserted, and hydraulically disarmed.

D. APPARENT CAUSE

Inspections and bench checks were performed on the withdraw control valve, valve C11-F422. The valve demonstrated no sign of abnormal operation, and no fouling of the valve seat surface was evident. It is concluded that temporary particulate accumulation on the valve seating surface caused an incomplete closure of the valve when the withdraw command was terminated, allowing drive water pressure to leak past the valve and force the drive piston downward.

E. SUPPLEMENTAL CORRECTIVE ACTIONS

The C11-F422 valve, manufactured by Automatic Switch Company (ASCO), GE Part Number 105D6025P1, was replaced with a new valve, and the faulty valve was disassembled for inspection. The newly installed valve was retested satisfactorily. The control rod was restored to service at 2035 on July 31, 1986.

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

During the upcoming refueling outage, system drive water filters will be checked and low stagnant points in the system will be sampled and analyzed for particulates. Appropriate actions will be taken based on the results of the samples and filter inspections.

During the investigation it was determined that a General Electric Service Information Letter (SIL) #292 had been issued in July, 1979 addressing this situation and providing additional recommended operator actions to be taken should this occur. Appropriate additional operator instructions have been added to the Off-Normal Event Procedure for Control Rod/Drive Malfunctions based on recommendations from SIL #292. The procedure revisions included the following actions to be taken as necessary:

1. Application of continuous control rod insert signal,

2. Manual scram of individual control rod, and

3. Isolation of affected control rod drive.

A special review of all G. E. SIL's is being conducted. Preliminary reviews are complete. Final reviews are being conducted by affected departments. Implementation of procedural related changes resulting from these reviews are targeted for September 30, 1986.

F. SAFETY ASSESSMENT

The situation did not constitute a threat to plant safety nor to the safety function of the control rod drive system. The inadvertent control rod withdrawal did not cause fuel safety limits to be exceeded and did not compromise fuel preconditioning limits. The apparent valve malfunction did not affect the ability of the rod to scram.

FSAR 15.4 discusses the evaluation of reactivity and power distribution anomalies. The control rod drift event that occurred at GGNS is most similar to the FSAR 15.4 2 scenario of a Rod Withdrawal Error (RWE) transient which is postulated to occur at power.

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The RWE transient at power results from a procedural error and is effectively mitigated by the single failure proof rod withdrawal limiter (RWL) mode of the rod control and information system (RCIS). In this scenario, the RWL mode limits the extent of rod (or gang) withdrawal by initiation of a control rod block signal. This transient is classified as an event of moderate frequency and was determined not to exceed the MCPR safety limit. The July 30 rod drift event at GGNS was bounded with respect to LHGR and MCPR by the RWE at power transient analyzed in FSAR 15.4.2.

Certain failures in the control rod system can result in the drifting of an individual rod past the selected position. As occurred at GGNS, the rod can drift to a full out location. Even so, such an event is considered to be a low frequency incident based on overall BWR operating experience. This position was provided to the NRC by MP&L in response to NRC Question #232.4 in June of 1979 (FSAR Amend. 31) and is supported by BWR operating experience since that time.

The frequency of an unmitigated rod drift event which exceeds the MCPR safety limit was determined to be quite low, such that this event is given an abnormal or infrequent incident classification for the purposes of FSAR Chapter 15 analyses. Such an event has been calculated to occur less than once per 500 reactor years.

It should be noted that even with the exceeding of the MCPR safety limit, the consequences are significantly less severe than the limiting control rod related event analyzed under the category of reactivity and power distribution anomalies, namely the control rod drop accident (FSAR 15.4.9).