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October 1, 1991

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
Update on Reactor Scram Due to 500 kV Circuit Breaker Fault
LER 91-005-01

GNRO-91/00169

Gentlemen:

Attached is Licensee Event Report (LER) 91-005-01 which is a final report.

Yours truly,

W T Cottle

WTC/RR/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)

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Grand Gulf Nuclear Station	0 5 0 0 0 4 1 6	1 OF 0 5

TITLE (4): Update on Reactor Scram Due to 500 kV Circuit Breaker Fault

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)
0 6	1 7	9 1	9 1	— 0 0 5 —	0 1	1 0	0 1	9 1	N/A	0 5 0 0 0 4 1 6
										0 5 0 0 0

OPERATING MODE (9)	1
POWER LEVEL (10)	0 7 3

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)	TS 3.5.1.h

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER AREA CODE	
Riley Ruffin / Licensing Specialist	6 0 1	4 3 7 - 2 1 6 7

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
X	F K	5 2	5 1 8 8	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 June 17, 1991 a fault occurred on the B-phase of breaker J5232 (500 KV). The fault resulted in a loss of power (LOP) to two balance of plant (BOP) busses and two engineered safety feature (ESF) busses. An automatic reactor scram occurred due to a main turbine control valve fast closure signal which was a result of the breaker fault. Power was restored to the BOP busses via their alternate supply.

The emergency diesel generator associated with each deenergized ESF bus responded to the bus undervoltage and functioned as designed. Power to the ESF busses was restored and the diesel generators were secured and placed in standby. The LOP also caused a Main Steam Line Isolation valve closure which resulted in a loss of feedwater and an increase in vessel pressure. Main Steam Line Safety Relief valves (SRV) were manually operated to control pressure. The High Pressure Core Spray System automatically initiated and injected into the vessel due to a low water level (reported per TS 3.5.1.h).

Based on an investigation of the occurrence, it was determined that the cause of the breaker fault is attributed to inadequate maintenance practices. There was no degradation of safety systems or components during the transient. Reactor water level was maintained at least 121 inches above the top of active fuel.

A. Reportable Occurrence

On June 17, 1991 at approximately 0933 hours, an automatic Reactor Protection System (RPS; EIIS Code: JC) actuation occurred due to a main turbine control valve (TCV) fast closure signal. During the transient, a low vessel water level condition resulted in an automatic actuation of the High Pressure Core Spray System (HPCS; EIIS Code: BG). These occurrences are reportable pursuant to 10CFR50.73(a)(2)(iv).

Additionally, the actuation and injection of HPCS is reportable pursuant to the action h. of the GGNS Technical Specification Limiting Condition for Operation 3.5.1.

B. Initial Conditions

The plant was in Operational Condition 1 at approximately 73.7% thermal power. A power ascension was in progress in accordance with plant operating procedures. The generator output was approximately 862 MWe.

C. Description of Occurrence

On June 17, 1991, plant operation personnel were continuing a plant power ascension in order to reach 100% thermal reactor power. At approximately 0932 hours, a loss of power (LOP) to Balance of Plant (BOP) buses, 11HD (6.9 KV) and 13AD (4.16 KV) (EIIS Code: EA), and Engineered Safety Features (ESF) buses, 16AB and 17AC (both 4.16 KV) (EIIS Code: EB), occurred.

The major BOP loads lost during the LOP were as follows:

Component	EIIS Code
1 Reactor Recirculation Pump	AD
1 Circulation Water Pump	KE
1 Main Turbine Control Fluid (EHC) Pump	TG
1 Condensate Pump	SC
2 Condensate Booster Pumps	SD
1 Heater Drain Pump	SN
1 Instrument Air Compressor	LD
1 Service Air Compressor	LF

NRC Form 366A (9-83)

LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

U.S. NUCLEAR REGULATORY COMMISSION
APPROVED OMB NO. 3150-0104
EXPIRES: 8/31/88

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

An electrical fault occurred on a 500 kV switchyard circuit breaker (J5232) (EIIS Code: FK) and caused the main generator protective relays to actuate which resulted in a main generator trip.

A main turbine trip signal was generated due to the generator trip. The turbine trip signal caused a TCV fast closure signal which resulted in an automatic RPS actuation.

The main steam isolation valve solenoids are powered from the RPS motor generator (MG) sets. The 'A' RPS MG was lost due to the loss of power to the 13AD bus. This resulted in a loss of power to the 'A' logic (Division I) solenoids. The Division II leak detection system logic was deenergized due to the loss of power to the 16AB bus which caused the 'B' logic (Division II) solenoids to deenergize. This resulted in a MSIV closure which caused a loss of feedwater (FW; EIIS Code: SJ). The maximum pressure indicated during the transient was 1080 psia.

Reactor Core Isolation Cooling (RCIC; EIIS Code: BN) was manually initiated for pressure control and to restore water level. Additionally, Main Steam Line Safety Relief Valves (SRV) were manually opened to assist in vessel pressure control. The pressure was maintained at approximately 865 psia. Two loops of the Residual Heat Removal System (RHR, EIIS Code: BO) were placed in the suppression pool cooling mode of operation to maintain suppression pool temperature below required limits.

The loss of vessel inventory, due to SRV operation, resulted in an automatic initiation of HPCS on low water level (-41.6). HPCS actuated and injected into the vessel increasing vessel water level to approximately 40 inches prior to being secured. Reactor water level reached a minimum mean level of -42 inches as indicated by control room level indication (1B21R623A:-39 inches and 1B21R623B:-45 inches). Auxiliary and Containment Building isolation logic level instruments did not sense the low water level condition due to the actual water level remaining near the setpoint and the duration of the low water level signal. The two BOP busses were restored using the alternate service transformer which supplies power to the plant loads.

Additionally, the remaining reactor recirculation water pump tripped due to an Anticipated Transient Without a Scram - Alternate Rod Insertion Recirculating Water Pump Trip (ATWS-ARI RPT) actuation on low water level (-41.6 inches).

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

Two ESF busses were also affected by the loss of offsite power. Upon the LOP to bus 16AB, the Division II diesel generator (D/G) started and assumed the load on it's associated bus. Division III D/G also responded due to the LOP as sensed by bus 17AC. Power was restored to the two ESF busses from an alternate ESF transformer via a 115 KV supply and the diesels were placed in standby.

All safety systems responded as designed.

D. Apparent Cause

The electrical fault caused a main generator trip which resulted in a main turbine trip. The RPS actuation was a result of a TCV fast closure signal which was generated by the turbine trip.

The LOP was due to a fault on the B-phase of circuit breaker J5232. The fault is attributed to an arc which occurred in high pressure tank No. 3.

Breaker J5232 (Siemens-Allis 550 KV GHO) uses high pressure Sulfur Hexafluoride (SF6) to extinguish the arc which occurs during circuit interruptions. The breaker assembly interrupter is located inside a grounded low pressure tank. As the interrupter contacts open, the blast valve opens which allows high pressure SF6 to extinguish the arc. The interrupter and blast valve assembly sit on a high pressure tank which contains high pressure SF6. The high pressure tank also acts as an insulator between the low pressure tank and the high voltage portions of the breaker assembly.

During an inspection of the No. 3 high pressure tank, personnel identified burn marks on the inside wall of the tank and a large hole in the corona ring located on the tank. Both resulted from the fault which occurred in the No. 3 tank. The investigation also revealed a thin film of gas compressor oil covering the inside walls of two of the three adjacent high pressure tanks. The most probable cause of the arc, as determined by plant personnel, is a layer of oil on the inside wall of high pressure tank No. 3, but this could not be confirmed until further analyses were performed.

The oil analysis determined that the oil contained small micron particles. A micron particle count was performed on oil samples taken from three breakers and new oil. Comparing the particle content of each sample it was found that particle content in the sample taken from breaker J5232 was approximately thirty-five times the particle content of the new oil. This excessively high particle count allowed an arc to be drawn from the high voltage section of the breaker to the ground. Oil samples were analyzed from the balance of the Siemens-Allis breakers and the particle counts determined to be acceptable.

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

The two sets of filters are located in a series between the compressor and the high pressure tanks to avoid the introduction of contaminants to the tanks. No abnormalities were observed during an inspection of the filters. Currently the filters are replaced at a frequency in accordance with vendor recommendations.

However, oil and micron particles were able to slowly pass through the filtration system and terminate in the piping connecting the high pressure tank. During operation of the blast valve, the deposited oil was allowed to spread into the high pressure tank due to the differential pressure. The oil build-up continued for several years until a tracking path to ground was developed.

All high pressure tanks, of the Siemens-Allis breaker, were inspected and cleaned during 1982 and 1983. Subsequent to 1983, the tanks were only inspected during maintenance on a tank or associated breaker internals.

Therefore, the cause is attributed to inadequate preventive maintenance.

E. Supplemental Corrective Actions

The high pressure tank was replaced and the 3 adjacent high pressure tanks within the B-Phase were inspected and cleaned. The high pressure tanks of the remaining breaker poles were also inspected.

Mississippi Power and Light, who has responsibility for performing maintenance on switchyard equipment, has agreed to inspect and clean the Siemens breakers during Refueling Outage 5. Subsequent to RFO5, the Siemens breakers will be inspected and cleaned every six years.

Analyses will be performed on the compressor oil removed during each oil change.

F. Safety Assessment

A review of the event confirmed all safety systems performed as designed. HPCS initiated on low water level and operated for approximately 2.5 minutes and restored vessel water level to 40 inches. Current usage factor for HPCS is less than the 0.70 limit as described in GGNS Technical Specification 3.5.1.h. This was the ninth HPCS actuation cycle which has occurred at GGNS. The minimum water level reached was approximately 121 inches above the top of active fuel. There was no degradation of safety components or systems during the transient.

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