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August 20, 2002

U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Attention: Document Control Desk

Subject: LER -2002-003-00 [Reactor Scram 104 on June 22, 2002 Due To Loss Of Service Transformer 21]

Grand Gulf Nuclear Station
Docket No. 50-416
License No. NPF-29

GNRO-2002/00069

Ladies & Gentlemen:

Attached is Licensee Event Report (LER) 2002-003-00 which is a final report.
This letter does not contain any commitments.

Yours truly,

William A Eaton

WAE/ACG:acg
attachment: ACG LER 2002-003-00
cc: (See Next Page)

cc:

Hoeg	T. L.	(GGNS Senior Resident)	(w/a)
Levanway	D. E.	(Wise Carter)	(w/a)
Reynolds	N. S.		(w/a)
Smith	L. J.	(Wise Carter)	(w/a)
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U.S. Nuclear Regulatory Commission Attention: Mr. E. W. Merschoff (w/2) Regional Administrator 611 Ryan Plaza Drive, Suite 400 Arlington, TX 76011-4005
U.S. Nuclear Regulatory Commission ATTN: Mr. David H. Jaffe NRR/DLPM (w/2) **ATTN: FOR ADDRESSEE ONLY** ATTN: U.S. Postal Delivery Address Only Mail Stop OWFN/7D-1 Washington, D.C. 20555-0001

NRC FORM 366 (7-2001)	U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB NO. 3150-0104 EXPIRES 7-31-2004
LICENSEE EVENT REPORT (LER) (See reverse for required number of digits/characters for each block)		Estimated burden per response to comply with this mandatory information collection request: 50 hours. Reported lessons learned are incorporated into the licensing process and fed back to industry. Send comments regarding burden estimate to the Records Management Branch (T-6 E6), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by internet e-mail to bjs1@nrc.gov, and to the Desk Officer, Office of Information and Regulatory Affairs, NEOB-10202 (3150-0104), Office of Management and Budget, Washington, DC 20503. If a means used to impose information collection does not display a currently valid OMB control number, the NRC may not conduct or sponsor, and a person is not required to respond to, the information collection.

1. FACILITY NAME	2. DOCKET NUMBER	3. PAGE
Grand Gulf Nuclear Station, Unit 1	05000 416	1 OF 6

TITLE (4): Reactor SCRAM Due To Loss Of Service Transformer 21.

5. EVENT DATE			6. LER NUMBER			7. REPORT DATE			8. OTHER FACILITIES INVOLVED	
MO	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REV NO	MO	DAY	YEAR	FACILITY NAME N/A	DOCKET NUMBER N/A
06	22	2002	2002	003	00	08	20	2002	FACILITY NAME N/A	DOCKET NUMBER N/A

9. OPERATING MODE	1
10 POWER LEVEL	100 %

11. THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check all that apply)

	20.2201(b)		20.2203(a)(3)(ii)		50.73(a)(2)(ii)(B)		50.73(a)(2)(ix)(A)
	20.2201(d)		20.2203(a)(4)		50.73(a)(2)(iii)		50.73(a)(2)(x)
	20.2203(a)(1)		50.36(c)(1)(i)(A)	X	50.73(a)(2)(iv)(A)		73.71(a)(4)
	20.2203(a)(2)(i)		50.36(c)(1)(ii)(A)		50.73(a)(2)(v)(A)		73.71(a)(5)
	20.2203(a)(2)(ii)		50.36(c)(2)		50.73(a)(2)(v)(B)		OTHER Specify in Abstract below or in NRC Form 366A
	20.2203(a)(2)(iii)		50.46(a)(3)(ii)		50.73(a)(2)(v)(C)		
	20.2203(a)(2)(iv)		50.73(a)(2)(i)(A)		50.73(a)(2)(v)(D)		
	20.2203(a)(2)(v)		50.73(a)(2)(i)(B)		50.73(a)(2)(vii)		
	20.2203(a)(2)(vi)		50.73(a)(2)(i)(C)		50.73(a)(2)(viii)(A)		
	20.2203(a)(3)(i)		50.73(a)(2)(ii)(A)		50.73(a)(2)(viii)(B)		

12. LICENSEE CONTACT FOR THIS LER

NAME	TELEPHONE NUMBER (Include Area Code)
Avinash Goel, Senior Engineer (Plant Licensing)	601-437-6296

13. COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT

CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX

14. SUPPLEMENTAL REPORT EXPECTED			15. EXPECTED SUBMISSION DATE	MONTH	DAY	YEAR
	YES (If yes, complete EXPECTED SUBMISSION DATE).	X NO				

16. ABSTRACT (Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines)

On June 22, 2002 at approximately 2217, while operating at steady state conditions of 100 percent rated thermal power and 88.7 percent total core flow, a reactor scram occurred on Turbine Control Valve fast closure. It was the result of a ground fault on the secondary (i.e. 34.5 kV side) of Service Transformer 21 Engineered Safety Feature busses 16 and 17 were de-energized. Division 2 and 3 diesel generators started and energized busses 16 and 17, respectively. No Emergency Core Cooling System initiated. A partial loss of plant non-safety power busses occurred. Those powered by Service Transformer 21 were lost, while those powered by Service Transformer 11 remained energized. Important loads lost were both Reactor Recirculation Pumps, and the running Electro-hydraulic Control (EHC) pumps. A third EHC pump auto started on low fluid pressure. Condensate Pumps B and C, and Condensate Booster Pump C tripped due to loss of buss 14AE. Numerous Division 2 isolations occurred due to loss of 16AB. The lowest reactor level noted during the transient was minus 7 inches, approximately 18 inches below the low level automatic scram setpoint of 11 inches. Auxiliary site power loop was also de-energized. Important loads lost were the Emergency Operations Facility and a partial loss of security lighting.

Both Reactor Feed Pumps (RFPs) tripped on low suction pressure about 8 minutes after the scram. One RFP was immediately restored.

Safety related control room air conditioner "B" failed to sequence back on after the diesel re-energized buss 16. The air conditioner was started manually.

A. REPORTABLE OCCURRENCE

Reactor scram 104 occurred at 2217 on June 22, 2002 due to loss of Service Transformer 21 (ST21) [FK]. The ST21 lockout caused loss of busses 11HD, 14AE, 16AB, 17AC, and 28AG. The running Electro-hydraulic Control (EHC) Pumps lost power resulting in Turbine Control Valve fast closure. Division 2 and 3 Diesel Generators [EK] started as required and energized their respective busses.

<u>LER Reportable Events:</u>

1. Automatic reactor scram from steady state condition resulting from a lockout trip on ST21 and subsequent Turbine Control Valve (TCV) fast closure.
2. De-energization of Engineered Safety Feature (ESF) busses 16 & 17. Division 2 and 3 diesel generators started and energized busses 16 and 17, respectively.
3. Reactor Low Level for a short period of time due to perturbation on the Feedwater System.
4. Division 2 Control Room AC [VI] failed to automatically restart and required manual breaker closing to start.

Notification was made to the NRC's Emergency Notification System (ENS) reporting this condition pursuant to 10CFR50.72(b)(2)(iv)(B), 10CFR50.72(b)(3)(iv)(A) and 10CFR50.72(b)(3)(xiii) and is being reported under 50.73(a)(2)(iv)(A) – "Any event or condition that resulted in manual or automatic actuation of any of the systems listed in paragraph (a)(2)(iv)(B) of this section......".

Additionally, pursuant to 10CFR50.73(a)(2) the following events are being included in the LER:

a) Both Reactor Recirculation Pumps [AD] received an End of Cycle Recirculation Pump Trip (EOC-RPT) signal on the loss of TCV trip fluid pressure. In addition, loss of power to balance of plant (BOP) busses prevented operation on Low Frequency Motor Generators (LFMGs), causing reactor cooldown and heatup limits of 100 degrees to be exceeded.
b) The BOP diesel generator that supplies backup power to the Emergency Operations Facility (EOF) [NC] failed to start. The backup EOF remained available.
c) Partial loss of security lighting.

B. Initial Conditions

At the time of the event, the reactor was in OPERATIONAL MODE 1 with reactor power at approximately 100 percent. Moderator temperature, reactor pressure vessel (RPV) pressure and RPV water level were at approximately 549 degrees F, 1025 PSIG and 35.9 inches, respectively. There were no additional inoperable structures, systems, or components at the start of the event that contributed to the event.

C. Description of Occurrence

On June 22, 2002 at approximately 2217, due to a ground fault on the secondary (i.e. 34.5 kV) side of Service Transformer 21 (ST21) [FK] a reactor scram occurred at Grand Gulf Nuclear Station. The 34.5 kV System is non-safety related. It provides power from the 500 kV system through step-down Station Service Transformers to various plant BOP and selected ESF Transformers. The ST21 lockout caused loss of busses 11HD, 14AE, 16AB, 17AC, and 28AG. The loss of the 14AE buss caused the running EHC pumps to trip. The resulting perturbation on the EHC system resulted in loss of EHC system pressure and subsequent RPS trip on TCV fast closure (low trip fluid pressure). All withdrawn Control Rods inserted full in on the scram. Concurrent with this the safety related diesels responded to the under voltage condition on their busses. The Reactor Recirculation Pumps received an End of Cycle Recirculation Pump Trip (EOC-RPT) signal on the loss of TCV trip fluid pressure. The loss of the 14AE buss also caused a loss of two Condensate Pumps and one Condensate Booster pump [SD]. This loss of Condensate and Condensate Booster Pumps left only one Condensate and two Condensate Booster Pumps supplying water to the two Reactor Feed Pumps. Residual Heat Removal “B” [BO] (running in suppression pool cooling mode) tripped.

The affected busses were lined up to other available power sources. The safety related busses were synchronized back to the grid and Diesel Generators [EK] were secured. The Under Voltage lockouts on the 11HD and 14AE busses were reset after the busses were tied to available power sources. This allowed desired loads to be restarted. The 28AG buss was cross tied to 18AG, and additional Plant Service Water pumps were started.

Reactor level was controlled by the Feedwater System using the Startup Level Control Valve. The lowest Reactor level reached was approximately minus 7 inches, approximately 18 inches below the low level automatic scram setpoint of 11 inches. The maximum Reactor pressure reached was about 1120 psig due to a momentary loss of Hydraulic Control Fluid pressure that caused Turbine Control/Stop Valves to close and delayed Bypass valve response. This pressure rise was mitigated by the operation of 11 of the Main Steam Safety Relief Valves (MSRVs) in the relief mode. After the initial pressure transient, Reactor pressure was controlled by the Main Turbine Bypass Control Valves and no more MSRVs lifted.

About 8 minutes after the scram, a decreasing level trend caused the Reactor Feed Pumps to speed up to provide more feed flow. Due to the abnormal Condensate/Booster pump lineup and low flow conditions after the scram there was insufficient suction pressure available to the Reactor Feed Pumps. As a result, both RFPs tripped on low suction pressure. One Reactor Feed Pump was immediately restored, and no further feedwater problems occurred.

The loss of the Reactor Recirculation Pumps and Reactor Water Cleanup System (RWCU) in conjunction with the cooler water injected by the Control Rod Drive System resulted in thermal stratification in the lower portions of the RPV. When the RWCU system was restarted, this mixed the stratified water. The stratification and subsequent restart of the recirculation pumps caused the 100 degrees F heatup/cooldown limits to exceed.

Safety related control room air conditioner “B” failed to sequence back on after the diesel re-energized the ESF buss 16. Operators were able to manually restart the air conditioner.

The BOP diesel that supplies backup power to the EOF failed to start. The backup EOF remained available.

Plant busses, the auxiliary site power loop and the EOF were re-energized from alternate power sources.

No plant conditions or evaluations in progress at the time of the scram had an effect on the events leading to the scram or on the consequences of the scram.

D. Apparent Cause

The ground fault was the result of a raccoon bridging the 34.5 KV phase "B" to a grounded stanchion in the vicinity of switch 589-21GT. This switch is a manually operated disconnect switch, located electrically between the secondary of ST21 and the Grounding Transformer for ST21. ST21 is a 500 kV (primary) to 34.5 kV (secondary) transformer. (See Attachments 1, 2 and 3).

The scram signal was from control valve fast closure due to the loss of the running EHC pumps. Plant response was verified to be normal for a turbine trip without bypass valves transient. Turbine bypass valves were automatically restored to service when the standby EHC pump started.

<u>Component Failure</u>

10CFR50.73(b)(6) defines component failure as the termination of the ability of a component to perform its required function. The following details and evaluates the possible component failures during this event:

a. Service Transformer 21 (ST21) - A technical evaluation concluded that no functional impairments to ST21, the disconnect switch, the bus structures, (including insulators), the Grounding Transformer, or the protective relaying has occurred or has been exhibited by this event. The Service Transformer was subsequently re-energized without incident.

b. Control Room Air Conditioner "B" - The Air Conditioner failed to sequence back on after the diesel re-energized the buss 16 but started manually. Because, the unit could be manually started it was capable of performing its safety function and is not considered a failure.

c. EOF Diesel Generator – The EOF Diesel Generator is not part of plant equipment and its failure is not required to be reported in an LER. It is included in the LER because of interest in this issue expressed by the NRC.

E. Corrective Actions

<u>Immediate Corrective Actions:</u>

a. Service transformer protective relaying was verified to be functioning correctly. Checks were made of the service transformer and related equipment to verify no damage was done to any switchyard components. The service transformer was subsequently re-energized without incident.

b. An Engineering evaluation was completed on exceeding the heat up and cooldown rates and evaluate the cumulative fatigue usage factor including this scram. The evaluation concluded that the ASME Code allowable is maintained and there is no impact on the current operational basis.

c. A Load Shedding Agastat relay was replaced on the "B" control room air conditioner and was restored to service. As a follow up action the Load Center Breaker 52-16606 was also refurbished, retested and reinstalled under maintenance activity.

d. The EOF backup diesel was repaired and returned to service restoring normal operation of the EOF.

e. Issued Operating Experience Report (OE-14063 on June 24th, 2002).

Long Term Corrective Actions:

Condition Report GGN-2002-1106 and GGN-2002-1110 were written to address any additional action.

F. Safety Assessment

All safety systems responded as designed in this event with the exception that the Control Room Air Conditioner B which did not restart when Division 2 Load Shedding and Sequencing connection sequence occurred. This system is intended to protect the control room environment under accident conditions. This system was manually started. There was no impact to plant operators or control room equipment. No ECCS initiation or Primary/ Secondary Containment isolation setpoints were reached during this event. All system isolations were due to loss of power and not actual isolation signals. The power lost to the Emergency Operations Facility had no impact on safety. The cooldown and heatup limits that were exceeded did not compromise the integrity of the reactor coolant system (RCS) pressure boundary or reactor pressure vessel (RPV) components and is discussed below.

As discussed in the Technical Specification (TS) Bases, the Pressure/Temperature (P/T) limits prescribed by TS 3.4.11 are not derived from Design Basis Accident (DBA) analyses. They are prescribed during normal operation to avoid encountering pressure, temperature, and temperature rate of change conditions that might cause undetected flaws to propagate and cause non-ductile failure of the reactor coolant pressure boundary (RCPB), a condition that is unanalyzed.

All components of the RCS are designed to withstand effects of cyclic loads due to system pressure and temperature changes. These loads are introduced by startup (heatup) and shutdown (cooldown) operations, power transients, and reactor trips. Pressure and temperature changes during RCS heatup and cooldown are limited to within the design assumptions and the stress limits for cyclic operation.

The consequence of violating the LCO limits is that the RCS has been operated under conditions that can result in brittle failure of the RCPB, possibly leading to a non-isolable leak or loss of coolant accident. For this event, an evaluation was performed to determine the effect on the structural integrity of the RCPB components. Stress and fatigue analyses of the affected RCPB components (RPV components and RWCU piping) were evaluated. The evaluation concluded that heatup/cooldown rates assumed for design basis events are bounding from a component stress standpoint. The evaluation also determined that the fatigue usage factor and cumulative usage factor are below code allowable values. Therefore, the structural integrity of the RCPB components was not compromised.

The health and safety of the public was not compromised by this event.

G. Additional Information

Pursuant to 10CFR50.73(b)(5) the licensee considered this event to be an infrequent event. There has not been any occurrence of the same underlying concern in the past two years at Grand Gulf Nuclear Station. However, at other Non-Entergy plants a related event was recorded in Licensee Event Report (LER 482-00003, dated 10/04/2000). That event at Wolf Creek 1 was the result of a squirrel contacting the transient recovery voltage capacitors on the Unit Auxiliary Transformer. This resulted in both phase-to-phase and phase-to-ground faults and a subsequent transformer fire with installed suppression used to extinguish same. There was damage to the transformer from that event.

Note:

Energy Industry Identification System (EIIS) codes are identified in the text within brackets [ ].

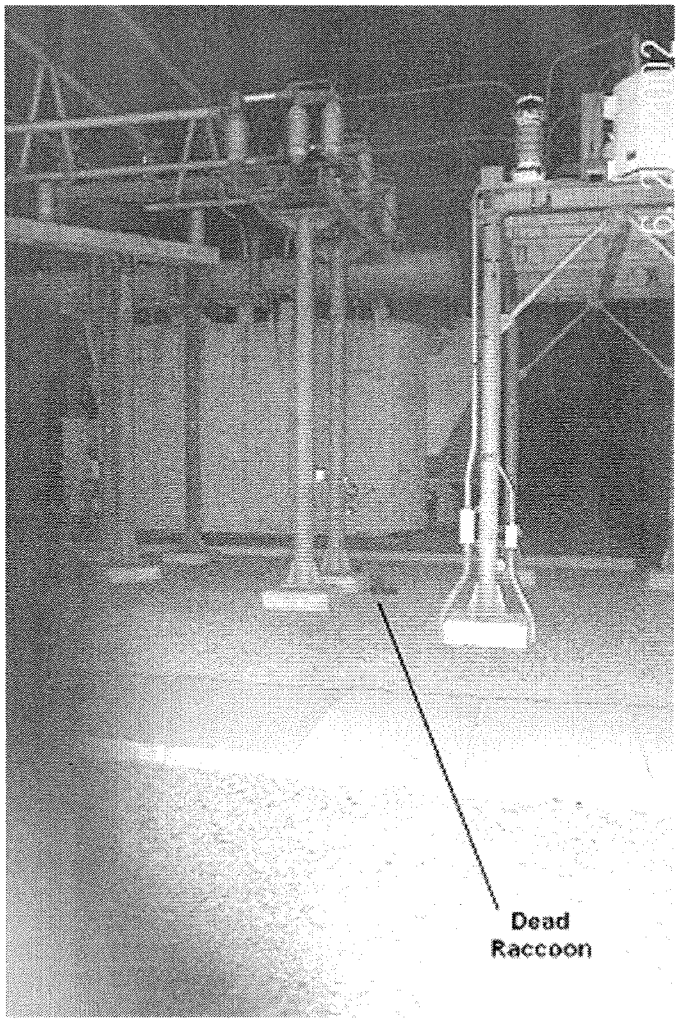
Attachments:

Attachment 1 – Picture of the dead raccoon after bridging the 34.5 kV phase "B".
Attachment 2 – Location of the dead raccoon shown on 34 kV switchyard drawing.
Attachment 3 - GGNS Distribution Switchyard.

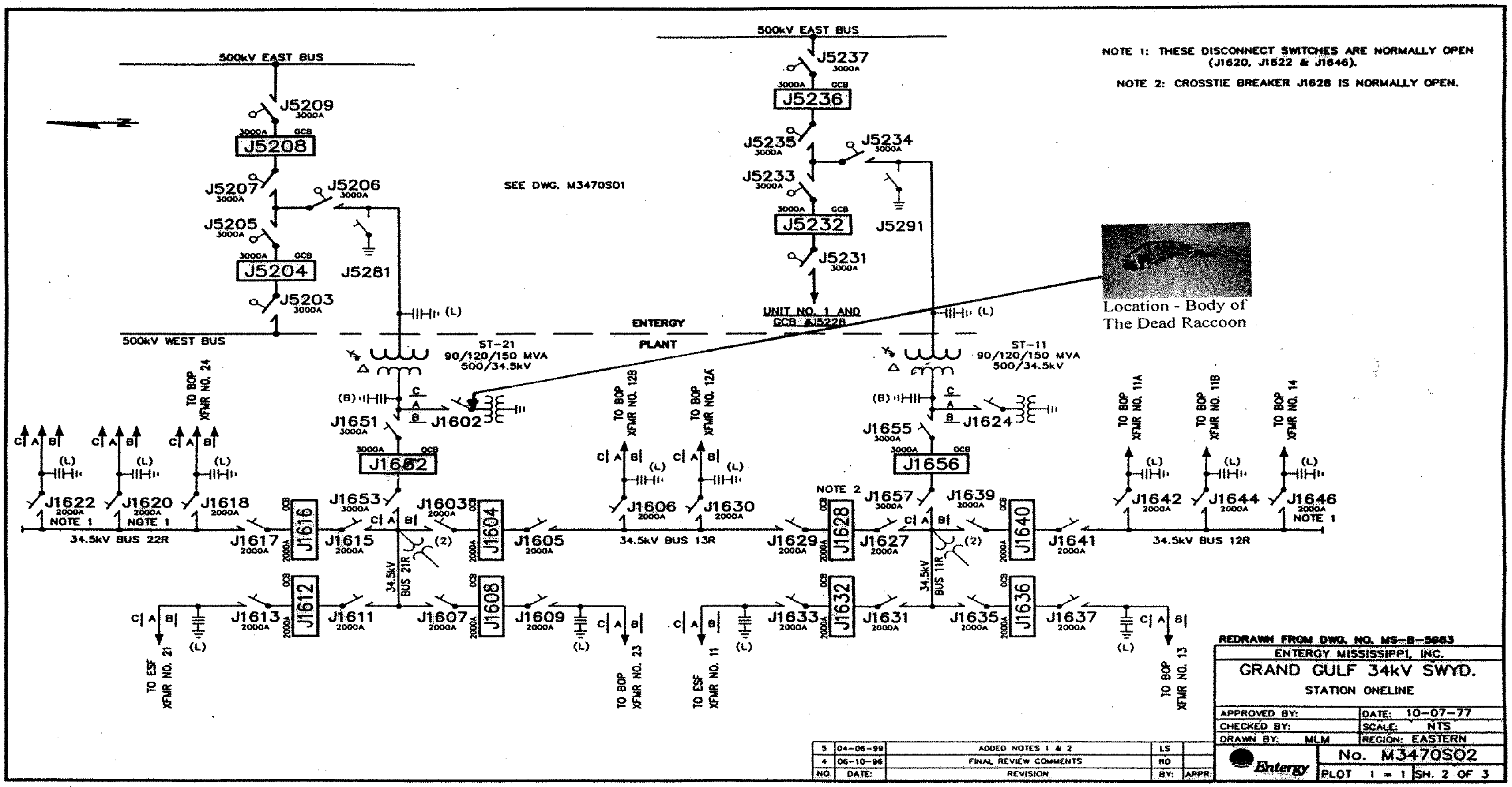
LER 2002-003-00

Pictures of dead Raccoon after bridging the 34.5 kV phase "B" to a ground stanchion On June 22nd, 2002 at Grand Gulf Nuclear Station

(Attachment 1)

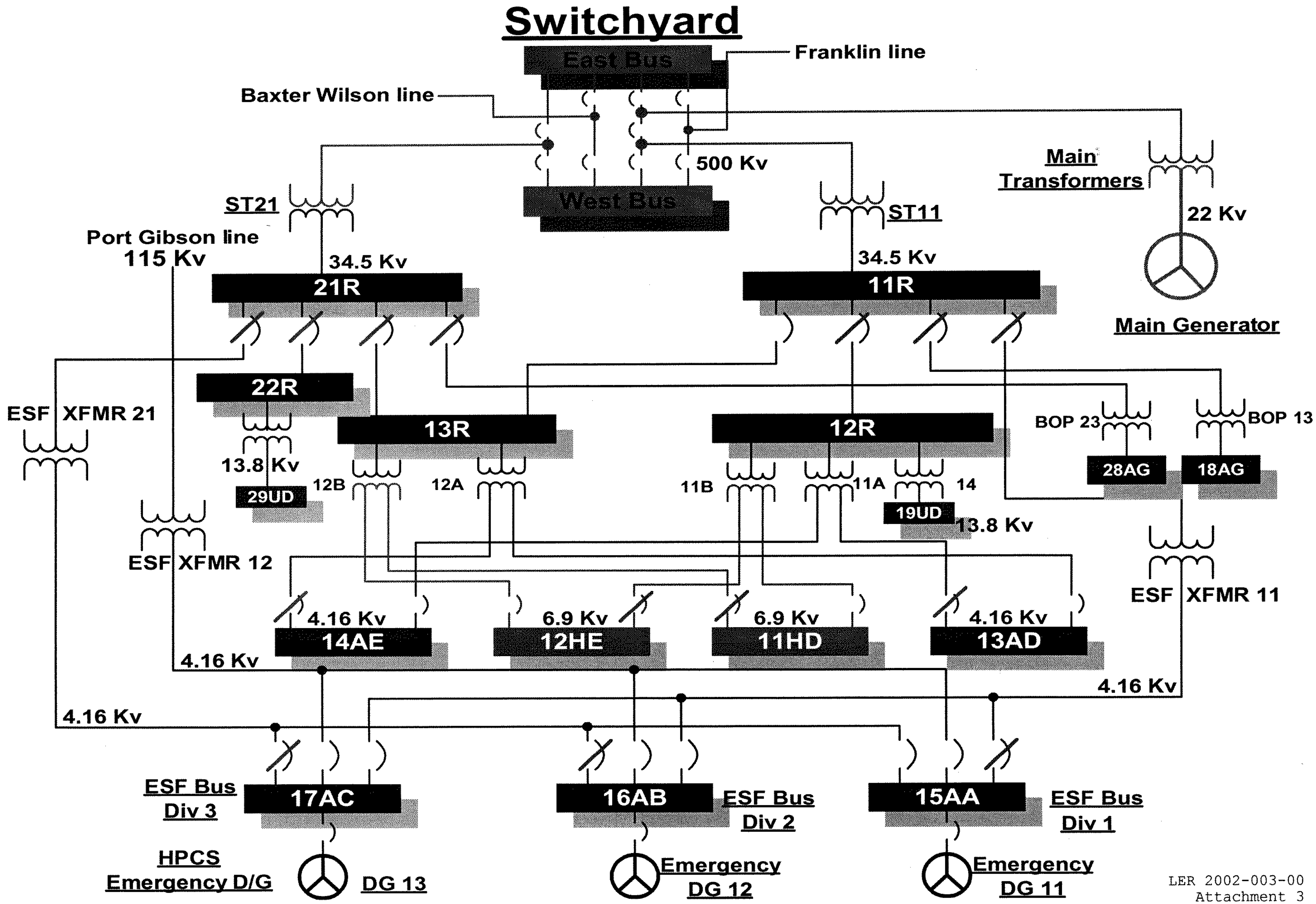


Entergy Mississippi, Inc.
Grand Gulf 34kV Switchyard
Location of the dead Raccoon



LER 2002-003
Attachment 2

Grand Gulf Nuclear Station
Distribution Switchyard



LER 2002-003-00
Attachment 3
(Ref. UFSAR Figure 8.1-001)