November 28, 1990

MEMORANDUM FOR: Chairman Carr

THRU: Hugh L. Thompson, Jr. /s/s
Deputy Executive Director for
Nuclear Materials Safety, Safeguards,
and Operations Support

FROM: Ben B. Hayes, Director
Office of Investigations DM for BH

SUBJECT: INVESTIGATION NOTIFICATION

The following information is submitted in compliance with the Commission directive that you be informed whenever the Office of Investigations (OI) opens a new investigation:

Vogtle Elec. Gen. Plant Case No. 2-90-020 Opened: 11/06/90

On November 1, 1990, the Regional Administrator, Region II, requested investigative assistance after two former employees of Georgia Power Company petitioned the NRC asserting that the licensee knowingly provided false statements intending to mislead the NRC with false assurances about the reliability of the diesel generator, and that SONOPCO knew the diesel generator had actually continued to experience an excessive number of trips, failures, and problems similar in nature to the failure which led to the March 20, 1990, station blackout. ECD 08/91.

If you desire more details on this or any other OI investigation, please let me know.

cc: Commissioner Rogers
Commissioner Curtiss
Commissioner Remick
J. Taylor, EDO
H. Thompson, Jr., DEDS
S. Chilk, SECY
D. Williams, OIG
W. Parler, OGC
S. Ebneter, RA:RII

Distribution:
s/f (2-90-020)
c/f
B. Barber, OI:HQ
P. Thompson, OI:RII
EDO r/f

OI:RII	OI:RII	OI:DD	OI:D
JVorse	DMurphy	RFortuna	BHayes
90/11/28	90/11/28	90/11/28	90/11/28

A1146

A1178

9610070007 960827
PDR FOIA
COLAPIN94-208 PDR

Larry,
Ron requested these
LERs
Doug
7-19-90/1705

Release
A/12/

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): VOGTLE ELECTRIC GENERATING PLANT - UNIT 1

DOCKET NUMBER (2): 0 5 0 0 0 4 2 4 | PAGE (3): 1 OF 0 9

TITLE (4): LOSS OF OFFSITE POWER LEADS TO SITE AREA EMERGENCY

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 3	2 0	9 0	9 0	— 0 0 6 —	0 1	0 6	2 9	9 0	VEGP - UNIT 2	0 5 0 0 0 4 2 5
										0 5 0 0 0

OPERATING MODE (9): 6

POWER LEVEL (10): 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)		T. S. 4.8.1.1.3
[ ]	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	
R. M. ODOM, NUCLEAR SAFETY AND COMPLIANCE	4 0 4	8 2 6 - 3 2 0 1

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)

On 3-20-90, Unit 1 was in a refueling outage and Unit 2 was operating at 100% power. At 0820 CST, the driver of a fuel truck in the switchyard backed into a support for the phase "C" insulator for the Unit 1 Reserve Auxiliary Transformer (RAT) 1A. The insulator and line fell causing a phase to ground fault. Both Unit 1 RAT 1A and Unit 2 RAT 2B High Side and Low Side breakers tripped, causing a loss of offsite power condition (LOSP). Unit 1 Diesel Generator (DG) 1A and Unit 2 DG2B started, but DG1A tripped, causing a loss of residual heat removal (RHR) to the reactor core since the Unit 1 Train B RAT and DG were out of service for maintenance. A Site Area Emergency (SAE) was declared and the site Emergency Plan was implemented. The Reactor Coolant System heated up to 136 degrees F from 90 degrees F before the DG was emergency started at 0856 CST and RHR was restored. The initial notifications were not made within the required 15 minutes due to the loss of power to the Emergency Notification Network (ENN). At 0915 CST, the SAE was downgraded to an Alert after onsite power was restored.

The direct cause of this series of events was a cognitive personnel error. The truck driver failed to use proper backing procedures and hit a support, causing the phase to ground fault and LOSP. The most probable cause of the DG1A trip was the intermittent actuation of the DG jacket water temperature switches.

Corrective actions include strengthening policies for control of vehicles, extensive testing of the DG, replacement of suspect DG temperature switches, and improvements in the ENN system.

NRC Form 366
(9-83)

9007050046 9pp

Release

A. REQUIREMENT FOR REPORT

This event is reportable per: a) 10 CFR 50.73 (a)(2)(iv), because an unplanned Engineered Safety Feature (ESF) actuation occurred when the ESF Actuation System sequencer started, and b) Technical Specification 4.8.1.1.3, because a valid diesel generator failure occurred. Additionally, this report serves as a summary of the Site Area Emergency event.

B. UNIT STATUS AT TIME OF EVENT

Unit 1 was in Mode 6 (Refueling) at 0% rated thermal power. The reactor had been shut down since 2-23-90 for a 45 day scheduled refueling outage. The reactor core reload had been completed, the initial tensioning of the reactor vessel head studs was complete, and the outage team was awaiting permission from the control room to begin the final tensioning. Reactor Coolant System (RCS) level was being maintained at mid-loop with the Train A Residual Heat Removal (RHR) pump in service for decay heat removal. The temperature of the RCS was being maintained at approximately 90 degrees F.

Due to the refueling outage maintenance activities in progress, some equipment was out of service and several systems were in abnormal configurations. The Train B Diesel Generator (DG1B) was out of service for a required 36 month maintenance inspection. The Train B Reserve Auxiliary Transformer (RAT 1B) had been removed from service for an oil change. The Train B Class 1E 4160 Volt switchgear, 1BA03, was being powered from the Train A RAT 1A through its alternate supply breaker. All non-1E switchgear was being powered from the Unit Auxiliary Transformers (UAT) by backfeeding from the switchyard. All Steam Generator (S/G) nozzle dams had been removed, but only S/G's 1 and 4 had their primary manways secured. Maintenance personnel were in the process of restoring the primary manways on S/G's 2 and 3. RCS level was being maintained at mid-loop for valve repairs and the S/G manway restorations. In addition, the pressurizer manway was removed to provide an RCS vent path.

C. DESCRIPTION OF EVENT

On March 20, 1990, at approximately 081. CST, a truck driver with a security escort entered the protected area in a fuel truck. Although not a member of the plant operating staff, the driver was a Georgia Power Company employee belonging to a service group used to perform various plant services. The driver checked the welding machine that was in the area and found that it did not need fuel. He returned to the fuel truck and was in the process of backing out of the area when he hit a support holding the phase "C" insulator for RAT 1A. The insulator and line fell causing a phase to ground fault, and the transformer breakers tripped.

At 0820 CST, both Unit 1 RAT 1A and the Unit 2 RAT 2B High Side and Low Side Breakers tripped causing a loss of offsite power condition (LOSP) to the Unit 1 Train A Class 1E 4160 volt bus 1AA02, the Unit 2 Train B Class 1E bus 2BA03, and the 480 volt busses supplied by 1AA02 and 2BA03. The Unit 1 Train B Class 1E 4160 volt bus 1BA03 also lost power since RAT 1A was feeding both Trains of Class 1E 4160 volt busses. The loss of power caused the associated ESF Actuation System Sequencers to send a start signal to one Unit 1 and one Unit 2 Diesel Generator. DG1A and DG2B started and sequenced the loads to their respective busses. Further description of the Unit 2 response to this event is provided in LER 50-425/1990-002.

One minute and twenty seconds after DG1A started and sequenced the loads to the Class 1E bus, the engine tripped. This again caused an undervoltage (UV) condition to class 1E bus 1AA02. The UV signal is a maintained signal at the sequencer. However, since DG1A was coasting down from the trip, the shutdown logic did not allow the DG fuel racks or starting air solenoids to open and start the engine. This properly caused the engine starting logic to lock up, a condition that existed until the UV signal was reset. For this reason, DG1A did not automatically re-start after it tripped.

After the trip, operators were dispatched to the engine control panel to investigate the cause of the trip. According to the operator, several annunciators were lit. The operator briefly reviewed several instrument read-outs and detected no immediate problem. In order to restore emergency power, the operator reset the annunciators without delaying to evaluate or record the annunciators that were present. During this time, a Shift Supervisor (SS) and a Plant Equipment Operator (PEO) went to the sequencer panel to determine if any problems were present on the 1A sequencer. The SS pushed the UV reset button, then reset the sequencer by deenergizing and energizing the power supply to the sequencer. This caused the DG air start solenoid to energize for another 5 seconds which caused the engine to start. This happened 19 minutes after the DG tripped the first time. The engine started and the sequencer sequenced the available loads as designed. After 1 minute and 10 seconds, the breaker and the engine tripped a second time. It did not automatically re-start due to the starting logic being blocked as described above. By this time, operators, a maintenance foreman and the diesel generator vendor representative were in the DG room. The initial report was that the jacket water pressure trip was the cause of the trip. This report was discounted because the maintenance foreman and vendor representative observed that the jacket water pressure at the gauge was about 12-13 PSIG. The trip setpoint is 6 PSIG and the alarm setpoint is 8 PSIG. Also, the control room observed a lube oil sensor malfunction alarm.

Fifteen minutes after the second DG1A trip, DG1A was started from the engine control panel using the emergency start breakglass button. The engine started and loads were manually loaded. When the DG is started in the emergency mode, all the trips except four are bypassed. However, all alarms will be annunciated. During the emergency run, no trip alarms were noticed by the personnel either at the control room or at the engine control panel. The only alarms noted by the control room operator assigned for DG operation were lube oil pressure sensor malfunction and fuel oil level high/low alarm, neither of which would have tripped the diesel.

At 1040 CST, RAT 1B was energized to supply power to 4160 volt bus 1BA03. DG1A supplied power to 4160 volt bus 1AA02 until 1157 CST, at which time bus 1AA02 was tied to RAT 1B.

A Site Area Emergency was declared at 0840 CST, due to a loss of all offsite and onsite AC power for more than 15 minutes. The Emergency Director signed the notification form used to inform offsite government agencies of the emergency at 0848 CST. The shift clerk attempted to initiate offsite notification utilizing the primary ENN in the control room but found it inoperable due to loss of power. The shift clerk then went to the back-up ENN and initiated notification after roll call on this system at 0857 CST. Due to the loss of power, which rendered the primary Emergency Notification Network (ENN) inoperable, and some mis-communication, the initial notification was not received by all agencies until 0935 CST.

The Emergency Director instructed personnel to complete various tasks for restoring containment and RCS integrity. All work was accomplished and maintenance personnel exited containment by 1050 CST.

The SAE was downgraded to an Alert Emergency at 0915 CST after restoration of core cooling and one train of electrical power. By 1200 CST, plant conditions had stabilized with both trains of electrical power being supplied from an offsite source (RAT 1B). After discussions with the NRC and local government agencies, the emergency was terminated at 1247 CST and all agencies were notified by 1256 CST.

D. CAUSE OF EVENT

Direct Cause:

1. The direct cause of the loss of offsite Class 1E AC power was the fuel truck hitting a pole supporting a 230kV line for RAT 1A. This was a cognitive personnel error on the part of the truck driver. There were no unusual characteristics of the work location that directly contributed to this personnel error.

2. The direct cause of the loss of onsite Class 1E AC power was the failure of the operable DG, DG1A, to start and load the LOSP loads on bus 1AA02.

3. The direct cause of the failure of the primary ENN system in the control room was the loss of electrical power to Unit 1. The primary ENN in the control room is powered from Unit 1 Class 1E AC power. Therefore, when Unit 1 lost Class 1E AC electrical power, the primary ENN in the control room did not work.

Root Cause:

1. The truck driver met all current site training and qualification requirements, including holding a Class 2 Georgia driver's license. However, site safety rules, which require a flagman for backing vehicles when viewing is impaired, were violated.

2. The root cause for the failure of DG1A has not been conclusively determined. There is no record of the trips that were annunciated after the first trip because the annunciators were reset before the condition was fully evaluated. Therefore, the cause of the first trip can only be postulated, but it was most likely the same as that which caused the second trip. The second trip occurred at the end of the timed sequence of the group 2 block logic. This logic allows the DG to achieve operating conditions before the trips become active. The block logic timed out and the trip occurred at about 70 seconds. The annunciators observed at the second trip included jacket water high temperature along with other trips. In conducting an investigation, the trip conditions that were observed on the second DG trip on 3-20-90 could be duplicated by venting 2 out of 3 jacket water temperature sensors, simulating a tripped condition. The simulation duplicated both the annunciators and the 70 sec. trip time. The most likely cause of the DG trips was intermittent actuation of the jacket water temperature switches.

   Following the 3-20-90 event, all three jacket water temperature switches, which all have a design setpoint of 200°F, were bench tested. Switch TS-19110 was found to have a setpoint of 197 degrees F, which was approximately 6 degrees below its previous setting. Switch TS-19111 was found to have a setpoint of 199 degrees F, which was approximately the same as the original setting. Switch TS-19112 was found to have a setpoint of 186 degrees F, which was approximately 17 degrees F below the previous setting and was re-adjusted. Switch TS-19112 also had a small leak which was judged to be acceptable to support diagnostic engine tests and was reinstalled. The switches were recalibrated with the manufacturer's assistance to ensure a consistent calibration technique.

During the subsequent test run of the DG on 3-30-90, one of the switches (TS-19111) tripped and would not reset. This appeared to be an intermittent failure because it subsequently mechanically reset. This switch and the leaking switch (TS-19112) were replaced with new switches. All subsequent testing was conducted with no additional problems.

A test of the jacket water system temperature transient during engine starts was conducted. The purpose of this test was to determine the actual jacket water temperature at the switch locations with the engine in a normal standby lineup, and then followed by a series of starts without air rolling the engine to replicate the starts of 3-20-90. The test showed that jacket water temperature at the switch location decreased from a standby temperature of 163 degrees F to approximately 156 degrees F and remained steady.

Numerous sensor calibrations (including jacket water temperatures), special pneumatic leak testing, and multiple engine starts and runs were performed under various conditions. After the 3-20-90 event, the control systems of both engines were subjected to a comprehensive test program. Additionally, the jacket water high temperature switches were sent to an independent laboratory, which found the switches set at temperatures ranging from 162 degrees F to 195 degrees F rather than the 200 degree F setting that was required. The calibration technique was changed and switches were re-calibrated and installed on DG1B on 5-23-90. However, another failure occurred on DG1B (See Technical Specification Special Report 1-90-4.). These switches were also sent to the independent laboratory, which found the settings to be from 164 degrees F to 169 degrees F. Subsequent to this testing, the onsite calibration procedure was again revised to provide a technique that is consistent with the actual operating conditions that the switches experience. Switches were calibrated using this new technique, installed and found to operate within the expected parameters. Since the event of 3-20-90 through 6-7-90, DG1B had received 12 valid tests with the one failure mentioned above, and DG1A had received 16 valid tests with no failures.

Based on the above facts, it is concluded that the jacket water high temperature switches were the most probable cause of both trips on 3-20-90.

The investigation and testing following the 3-20-90 event revealed that pressure sensors in the diesel generator lube oil system had not been replaced in accordance with a 10 CFR 21 notification from the manufacturer dated 5-12-88. The 10 CFR 21 notification was confusing relative to the requirements for their replacement. It was subsequently revised in an addendum dated 6-8-90. The pressure trip

sensors have been modified in accordance with the manufacturer's instructions. GPC does not believe that these sensors contributed to the diesel generator trip on 3-20-90.

E. ANALYSIS OF EVENT

The loss of offsite power to Class 1E bus 1BA03 and the failure of DG1A to start and operate successfully, coupled with DG1B and RAT 1B being out of service for maintenance, resulted in Unit 1 being without AC power to both Class 1E busses. With both Class 1E busses deenergized, the RHR System could not perform its required safety function. Based on a noted rate of rise in the RCS temperature of 46 degrees F in 36 minutues, the RCS water would not have been expected to begin boiling until approximately 1 hour and 36 minutes after the beginning of the event. Using more conservative assumptions and methods, but the same actual time of the event, the calculated worst case time to boiling was found to be approximately 1 hour and 11 minutes, and time to core uncovering was found to be approximately 11 hours and 5 minutes. This assumed no gravity feed from the RWST.

Restoration of RHR and closure of the containment equipment hatch were completed well within the estimated 1 hour and 36 minutes for the projected onset of boiling in the RCS. A review of information obtained from the Process and Effluent Radiation Monitoring System (PERMS) and grab sample analysis indicated all normal values. As a result of this event, no increase in radioactive releases to either the containment or the environment occurred.

Additional systems were either available or could have been made available to ensure the continued safe operation of the plant:

1. The maintenance on RAT 1B was completed and the RAT was returned to service approximately 2 hours into the event.

2. Offsite power was available to non-1E equipment through the generator step-up transformers which were being used to "back-feed" the Unit Auxiliary Transformers (UAT) and supply the non-1E busses. Provided that the phase to ground fault was cleared, Class 1E busses 1AA02 and 1BA03 could have been powered by feeding through non-1E bus 1NA01.

3. The Refueling Water Storage Tank could have been used to manually establish gravity feed to the RCS to maintain a supply of cooling water to the reactor.

Consequently, neither plant safety nor the health and safety of the public was adversely affected by this event.

F. CORRECTIVE ACTIONS

1. A management policy on control and operation of vehicles has been established.

2. Temporary barricades have been erected with signs which direct authorization for control of switchyard traffic to the SS.

3. The Loss of Offsite Power (LOSP) diesel start and trip logic has been modified on both Unit 1 and Unit 2 so that an automatic "emergency" start will occur upon LOSP. Therefore, non-essential diesel engine trips are blocked upon LOSP. Additionally, high jacket water temperature has been deleted as a trip signal in the emergency start mode.

4. The DG1A test frequency was increased to three times per week until 4-20-90 when the test frequency was changed to once every 7 days in accordance with Technical Specification Table 4.8-1. This frequency will be continued until 7 consecutive valid tests are completed with no more than one valid failure in the last 20 valid tests. Up to and including the two valid failures of the 3-20-90 event, there were a total of four valid failures in ~~60~~ 68 valid tests of DG1A.

5. The jacket water temperature switches for each DG were replaced or re-calibrated using a more appropriate technique prior to their installation.

6. A back-up ENN system powered from the AT&T system, which previously existed and was operational for South Carolina agencies, has been extended to include Georgia local and state agencies. Instructions have been given to Emergency Directors and Communicators concerning use of the emergency communication systems.

G. ADDITIONAL INFORMATION

1. Failed Components:

   Jacket Water High Temperature Switches manufactured by California Controls Company.
   Model #A-3500-W3

2. Previous Similar Events:

   None

3. Energy Industry Identification System Code:

Reactor Coolant System - AB
Residual Heat Removal System - BP
Diesel Generator Lube Oil System - LA
Diesel Generator Starting Air System - LC
Diesel Generator Cooling Water System - LB
Diesel Generator Power Supply System - EK
Safety Injection System - BQ
13.8 kV Power System - EA
4160 volt non-1E power system - EA
4160 volt Class 1E power system - EB
Chemical and Volume Control System - CB
Containment Building - NH
480 volt Class 1E Power System - ED
Engineered Safety Features Actuation System - JE
Radiation Monitoring System - IL

JUL-31-'90 14:50 ID:SONOPCO-VOGTLE TEL NO:1-205-877-7885 #254 P02

Post Office Box 2625
Birmingham, Alabama 35202
Telephone 205 877-7936

W. C. Ramsey, Jr.
Project Engineering Manager - Vogtle

Southern Company Services
the southern electric system

July 31, 1990

Vogtle Electric Generating Plant - Unit 1 and 2
Fire Event Safe Shutdown Evaluation of Diesel Generator Trips
File: REA VG-0047 Log: SG-9510
Security Code: NC

Mr. C. C. Miller
Manager of Engineering
Vogtle Project - Nuclear Operations
Georgia Power Company
Post Office Box 1295
Birmingham, Alabama 35201

Received JUL 1990 Nuc. Eng. & Lic. Vogtle

Dear Mr. Miller:

We have reviewed the potential emergency diesel generator ground fault trip described in our letter SG-9471 for requirements to notify the NRC per 10 CFR 50.72. As described in FSAR Paragraph 9.5.1.1.3 the fire protection program provides assurance that a fire will not cause the loss of function of safe shutdown systems with or without offsite power. A fire in fire areas 1/2-CB-LC-B was analyzed and protective measures were included in the design as described in FSAR Appendix 9A. Thus a fire in this area with an LOSP is within the design bases of the plant.

Due to a fire in these areas, A train cables may be damaged and the A train is assumed to be lost, however the B train cables are protected from the effects of the fire and no damage to B train equipment and cables required for safe shutdown would occur. Thus while an unanalyzed ground fault which could separate the diesel generator from the B train safety related bus might occur, the equipment required to achieve and maintain safe shutdown would remain undamaged and the plant configuration would be similar to a station blackout.

We believe that the corrective actions to isolate the ground fault and re-establish power to the B train are straight forward and readily accomplished within the time frame previously analyzed for a station blackout. Thus adequate time is availeble to provide power to the safety related equipment required to shut down the plant and the capability to meet the design bases of the plant is maintained.

In conclusion, were this to occur, it could represent an unanalyzed condition, but would not be a significant compromise of plant safety, and therefore is not reportable per the requirements of 10 CFR 50.72.

A/122

SCS Ramsey response
used to not report.

A/127

Mr. C. C. Miller
July 31, 1990
Page 2

File: REA VG-0047
Log : SG-9510

If you have any questions, contact Andy Wehrenberg at extension 6768.

Very truly yours,

Wm C Ramsey Jr

W. C. Ramsey, Jr.

WCRJr/cm

xc: Bechtel Power Corporation
S. Pietrzyk

Georgia Power Company
P. D. Rushton
NORMS

Southern Company Services, Inc.
C. R. Myer
R. E. Patrick
J. A. Wehrenberg
Document File

Date Issued	Standing Order VOGTLE ELECTRIC GENERATING PLANT	Georgia Power	Order No
7/20/90			1-90-11
	Unit One		Procedure Ref. N/A

Title: 1B D/G LOSP OPERATION

Approved: [signature] Operations Supervisor [signature] Operations Superintendent

DUE TO THE SETPOINT OF THE 1B D/G NEUTRAL GROUND OVERCURRENT RELAY A FIRE IN ZONE 80, CAUSING DAMAGE TO THE FEEDER CABLE TO 1NB10, COULD RESULT IN THE D/G TRIPPING BEFORE 1NB10 TRIPS. IF A LOSP OCCURS ON UNIT 1, VERIFY THERE IS NO FIRE ALARM IN ZONE 80. IF A FIRE ALARM IN ZONE 80 EXISTS, IMMEDIATELY TRIP 1NB10. THIS PROBLEM EXISTS ON "1B D/G" ONLY AND WILL BE CORRECTED IN THE NEAR FUTURE BY DELETING THE NEUTRAL GROUND OVERCURRENT D/G TRIP.

[redacted]

FOR INFORMATION ONLY

Unit 0

Georgia Power

1-90-98
Procedure Ref.

Title UNIT 1 & 2 "B" D/G [illegible] OPERATION

Approved [signature] Operations Supervisor [signature] Operations Superintendent

DUE TO THE SETPOINT OF THE UNIT 1 & UNIT 2 "B" D/G NEUTRAL GROUND OVERCURRENT RELAY, A FIRE IN ZONE 80 CAUSING DAMAGE TO THE FEEDER CABLE TO NB10 COULD RESULT IN THE "B" D/G ON THE AFFECTED UNIT TRIPPING BEFORE NB10 TRIPS. IF A LOSP OCCURS ON EITHER UNIT, VERIFY THERE IS NO FIRE ALARM IN THE AFFECTED UNIT'S ZONE 80. IF A FIRE ALARM IN ZONE 80 EXISTS, IMMEDIATELY TRIP NB10 ON THE AFFECTED UNIT. THIS PROBLEM EXISTS ON BOTH UNIT'S "B" D/G AND WILL BE CORRECTED IN THE NEAR FUTURE BY DELETING THE NEUTRAL GROUND OVERCURRENT D/G TRIP.

FOR [illegible] [illegible]

704854 MCS 4C

Southern Electric System Performance Pay Plan
Survey 1990

Larry: This is a standard survey done for all Southern Company divisions by a company that specializes in surveys & opinion questionairs. This company also does numerous Fortune 500 companies.
Bockhold gave us the results recently and said Vogtle had the "worst results" in the company. Answers to questions #38, 51, 54, 57 55 56, 58 and 65 are revealing.

SONOPCO Project

VOGTLE PPP SURVEY FEEDBACK

PLANT VOGTLE

SEE SUMMARY TAB:

None of the overall dimensions were positive, however, all either improved or remained the same since 1989

POSITIVE FINDINGS:

THE 1990 PAY PLAN (PPP):

1. Has been explained to me in such a way that I understand it.

WITH REGARD TO GOAL SETTING IN THE PPP:

6. I understand the corporate goals used in the PPP.
21. Accomplishing my individual goals contributes to the achievement of our organizational goals.

IN GENERAL:

61. Increasing competition represents a major threat to our company.
64. I would like to continue working under some type of performance pay plan.

NEGATIVE FINDINGS:

THE 1990 PERFORMANCE PAY PLAN (PPP):

2. Is not having positive effects on the company's performance.
3. Is not having a positive effect on my work habits and performance.
4. Doesn't offer enough money in the incentive portion to make it work.
5. Will not appropriately compensate me for my contributions.

Vogtle had a good payout. Why did the people feel this way?

Did the large numbers of people rated in the top two blocks have an affect on this? They might have believed they were going to receive a 15-20% incentive increase. Inflated performance ratings apparently hurt.

VOGTLE PPP SURVEY FEEDBACK

WITH REGARD TO GOAL SETTING IN THE PPP:

7. I don't see the connection between my individual goals and the corporate goals.
8. My manager has not asked for my input in understanding the results of last year's survey.
9. My PPP incentive payment was not based on my job performance.
11. The PPP goals of my organization are more difficult than the goals of other organizations.
13. It is not possible to achieve our PPP organizational goals at the top level. (5 level of performance)
14. Work groups inside my organization do not cooperate in accomplishing our organizational goals.
24. My individual goals leave out important aspects of my job that are difficult to measure.
25. As a result of having individual goals, my co-workers are less willing to help each other.
26. To reach our 1989 organizational goals, people made decisions which will hurt my organization in the long run.
27. Accomplishing my individual goals is beyond my control.
28. As circumstances change, I will not be able to change my individual goals to make them more appropriate.

MY 1990 INDIVIDUAL GOALS DO NOT REQUIRE ME TO:

29. Be more flexible in dealing with my customers, supervisor, and co-workers.
34. Take intelligent risks.

MY MANAGER/SUPERVISOR IS NOT EFFECTIVE AT:

35. Treating all employees in my group fairly and consistently.
* 36. Demonstrating a strong belief that changes going on in the company are necessary.
37. ~~Recognizing and rewarding people based on performance.~~
* 38. Encouraging and environment of open communications.
39. ~~Giving me meaningful~~ and honest ~~feedback on how I~~ perform my job.
41. Helping me, in a positive way, to understand how the PPP works.
42. Getting my work group to function as a team.
43. ~~Conveying the results he/she expects of me.~~
44. Communicating about organizational goals and my role in helping to achieve them.
45. Giving me the authority to make decisions I need to make.
46. Taking the right action with poor performers.

* Denotes items of high interest 20-Jul-1990

IN GENERAL:

47. I am not paid fairly compared with other employees in my company who do similar work.
48. Employees are not rewarded for acquiring new skills.

Are people told in performance feedback sessions that they have some developmental needs and they should improve in certain areas. Then, either management doesn't provide developmental opportunities or reward the acquisition of new skills?

49. Written information on the PPP has not helped me to understand the program.

* 51. Employees are afraid to voice an opinion management does not want to hear.

52. There are not ways for employees to formally participate in solving problems for the company.

* 54. People in my organization do not trust each other.

55. Unnecessary change is occurring within our company.

Management is not supporting the changes going on.

* 56. The officers of my company are not aware of the problems at my level.

57. When I make suggestions for improvement, I am ignored.

* 58. There is not good up and down communications throughout the company.

59. I was not given the training I needed to make the PPP work.
60. I was not giving training in how to set effective goals.
62. Managers did not use last year's survey data to improve the operation of the organization.

* 65. I am afraid to voice an opinion that my management doesn't want to hear.

66. I do not understand how my base salary increase was determined.

WRITTEN COMMENTS:

POSITIVE:

People liked being judged on merit and the money.

NEGATIVE:

They didn't like the lack of fairness and favoritism of the allocations. They felt the goal quality was poor, too trivial, too easy to achieve, and too subjective.

This is the kind of intimidation people receive from management. throwing back the burden on the

It also reflects on managements lack of conservative decision making and willingness to admit guilt and accept blame.

I also opposed the PRB majority on this vote with JGA.

BOCKHOLD HAS TO WRITE EXPLANATORY MEMOS ON SPLIT PRB REPORTABILITY DECISIONS. (BY PROCEDURE?)

HE HAS J.G.A. DRAFT THEM FOR HIM.

IN THIS CASE, JGA WAS A DISSENTING PRB MEMBER.

Interoffice Correspondence

If you do not agree, see me.

George

DATE:

RE: Reportability Determination for Deficiency Card 1-90-003
Log: NOTS-00383

FROM: G. Bockhold, Jr

TO: PRB ~~Chairman~~ members and SRB members

On January 5, 1990, the [illegible] met to review the reportability of Deficiency Card 1-90-003. The deficiency involved a potential violation of a licence condition, specifically the reactor power limit of 3411 MW_{th}. The board determined by a split decision that the item was not reportable.

Subsequently, ~~on January [illegible], Mr. J. G. Aufdenkampe issued a minority opinion taking issue with the board's decision. After reviewing the issue including the minority opinion, I have determined that the board decision was non conservative based on the information available at the time of the decision. Information~~ obtained during additional reviews verified that the item was not reportable ~~but at the time of the board meeting this information was not available~~.

~~The board's decisions should always be made in the conservative direction. When information is not available which clearly indicates an item is not reportable, the decision should default to the conservative or not be made until sufficient information is available.~~

If you have any questions ~~or additional comments, please let me know.~~

~~This was not [illegible]~~
The reviews were completed and showed that the item was not reportable.

GB/JGA/chd

xc: NORMS

Mr J. G. Aufdenkampe, on January 10, 1990, issued a minority opinion questioning the recommendation of the PRB. My decision not to report this item on January 5, 1990, was based upon the majority recommendation and my belief that more information would be expeditiously provided by Tech. Support, the organization [illegible]

that Mr. G. [illegible]kampe is responsible
for.

My Circles,
Not on Original

DC's identifing
problem on Unit 2

Deficiency Card

Card # 2-90-080

Common

1: Description of Deficiency

Additional Sheets Attached? ☐Yes ☐No

THERE ARE SEVERAL [illegible] CABLES IN 2NBIO (NB10) SWITCHGEAR ROOM. THESE [illegible] UNDER THE SAME FIRE ZONE (ZONE 55). 2B OUTPUT BREAKER TO TRIP [illegible] RELAY (SSN) AND PART OF TRAIN A 4-85 [illegible] (WITH TRAIN A CABLES IN ZONE 85) MAY [illegible] INDUCED FAULT.

Completed By Initiator

Location Of Deficiency? 2NBIO SWITCHGEAR ROOM

What is Affected By The Deficiency? D/G 2NBIO [illegible] ("CURRENT WITH LOSP")

How Was The Deficiency Discovered? REVIEWING [illegible] OVERCURRENT RELAY FUNCTION DURING THE FIRE

Event Time 4-20 pm SAT. Date 7-23-90 Discover Time [illegible] Date 7-23-90

Discovered By? NPSN / KOCHERY Phone # 3132 Dept. ENGG SUPPORT

2: Shift Supervisor Review

Name Of USS Reported To? JOHN BOWLES Time 1600 Date 7-23-90

Plant Mode/Condition: 1/100%

Is Immediate Notification Required? ☐Yes ☐No

If Yes, ☐1 Hour, ☐4 Hour, or ☐24 Hour Reported Date N/A Time

Tech. Spec. Required Action Taken? ☐Yes ☐No N/A

Within 2 Hours

Summarize Compensatory Action Taken

DCP-290V2N0180 issued by Engineering to prevent the occurrence. Refer to 1C 1980 299 DCP-290V2-0180

LCO Initiated: ☐Yes ☐No # Type Info LCO Fire

WRT Initiated: ☐Yes ☐No #

Signature Of USS [signature] Date 7-23-90 Time 1603

Completed By

706686 1238

3: Technical Support Review

NSAC Evaluation/Review

A.		No Deficiency
B.	X	Reportable Deficiency
C.		Deficiency Not Reportable

Explanation: The ... (a)(2)(v)(A) ...

Completed in 1 Day

Responsible Dept:

NSAC Reviewer

NSAC Supervisor

4: Disposition for Deficiencies

Completed in 1 Month By Responsible Dept

Cause Code:

Causing Dept(s):

Department Manager:

706886 236

Or

My Note

QC's identifying problem on Unit 1

My Circles
Not on Original.

CWO # 2-90-299

1: Description of Deficiency

THERE ARE SEVERAL ...

(NON IE B TRAIN FED) SWITCHGEAR ... TRAIN A CABLES

AND CABLES TO AND FROM 1NB10 ARE UNDER THE SAME FIREZONE

(ZONE 28B). A FIRE IN ZONE 28B ... BUT

PUT BREAKER IN TRIP CLOSING

OVERCURRENT RELAY (51N) AND ...

120V ELECTRICAL SYSTEM (ASSOCIATED WITH ABOVE MENTIONED

TRAIN A CABLES) MAY NOT BE AVAILABLE DUE TO FIRE.

Location Of Deficiency? 1NB10 SWITCH GEAR Room

What Is Affected By The Deficiency? D/G 1B AND PART D/G 1A

How Was The Deficiency Discovered? DURING A REVIEW OF NEUTRAL

OVERCURRENT RELAY COORDINATION IN FIRE

Event Time 3-30 Pm Date 7-19-90 Discovery Time 3-30-90 Date 7-19-90

Discovered By? NPVS / KOCHERY ... Dept. ENGG. SUPPORT

2: Shift Supervisor Review

Name Of USS Reported To? CC Elbert Time 1712 Date 7-19-90

Plant Mode/Condition: 1

Is Immediate Notification Required? ☐Yes ☒No

If Yes, ☐1 Hour, ☐4 Hour, or ☐24 Hour Reported Date N/A Time

Tech. Spec. Required Action Taken? ☐Yes ☐No ☒N/A

A Temp ... 7-19-90 to prevent the

occurrence. An ... order ... to render the ...

LCO Initiated: ☐Yes ☒No # ...

WRT Initiated: ☐Yes ☒No #

Signature Of USS CC Elbert Date 7-19-90 Time 1714

706880 1238

3: Technical Support Review

NSAC Evaluation/Review (Check Appropriate Box) Date Received: 7-25-90

A		No Deficiency Card Required. Send Copy To Responsible Dept.
B.	X	Reportable Deficiency, Report # DICR # 90-24
C.	~~X~~	Deficiency, Not Reportable

Explanation: A fire approaching a cable tray that alone could prevent the fulfillment of the safety function of a system needed to shutdown the reactor and maintain it in a safe shutdown condition. This is reportable per 10CFR 50.73 (a)(2)(v) (A).

Responsible Dept: Tech Support

NSAC Reviewer: Tim Wakk Date: 7-25-90

NSAC Supervisor: Date:

Completed in 1 Day

4: Disposition, For Deficiencies in Item 3C Above Only.

Completed in 1 Month By Responsible Dept.

Cause Code: Event Co

Causing Dept(s):

Department Manager:

Should have been 4 hour report per 50.72 b.2 iii A

Or

1 Hour Red Phone per 50.72 b.1.ii B

"In a condition outside the design basis"

My Note

New Issue on

Fire Event Safe
Shutdown Design
and
Reporting of it.

COPY OF
WHOLE
PACKAGE
GIVEN
TO URYC
8/6/90

LLR

OPERATION OUTSIDE THE DESIGN BASIS
FIRE PLUS LOSP DEFEATS SAFE SHUTDOWN DESIGN CAPABILITY

On 7-19-90 two deficiency cards (DC's 1-90-299 and 2-90-080) were written by a plant electrical engineering supervisor. The DC's identified that 1E electrical train cable separation problems existed in the 1NB10 switchgear room for Unit 1 and in the 2NB10 switchgear room for Unit 2. The DC's stated that a fire in Zone 80 could cause Diesel Generator 1B or 2B to trip thru the neutral over current relay and Safety Related A train 480 V and 120 V circuits would be lost due to fire damage because cables affecting both A train and B train were located in the same fire zone.The train A cables include "Safe Shutdown" cables.
This condition (of having a design where a single fire event plus loss of offsite power would result in the loss of Safe Shutdown functions) is a condition outside the design basis of the plant. It violates the general design criteria of 10CFR50 Appendix A criterion 3, 17,21 and the requirments of Branch Technical Position CMEB 9.5.1 of NUREG 800.In addition the operability of effected equipment so designed is also in question.Operation outside the safety design basis essentially places the plants in "motherhood".

To the operator the observed condition would be extremely confusing.In the event of a fire in zone 80, for example, the safe shutdown procedures direct the operator to use train B.The operator would find the B train diesel tripped with a ground fault that could not be cleared.A train would be unavailable due to fire damage.

Fire areas were evaluated in the Vogtle FSAR Fire Hazzards Review and are described in the Vogtle FSAR sections 9A.2.29,9A.1.40,9A.1.50,and 9A.1.47 (see chart below)These sections are incorrect.
They fail to consider the design dificiency described above and provide guidance for "safe shutdown" to use a train that would be faulted, tripped,and unavailable.

Despite the above conditions being reported to the Shift Supervisor thru the DC's the Shift Supervisor did not make the required prompt notification to the NRC within 1 hour under 10CFR50.72(b)(1)(ii)(A)or(B).This condition would also be considered reportable under10CFR50.72(b)(2)(iii)(A) which requires a 4 hour report.
Operations did initiate a "Standing Order" (1-90-11) on 7-19-90 that describes the problem as being only on Unit 1.The standing order is incomplete and does not address all the fire zones (only Zone 80) where the electrical separation design is deficient(see chart).
It relies on operator action and may not mitigate the deficiency.

RECD @
RESIDENCE
& GIVEN TO
URYC ON
[illegible]

On 7-30-90 a second standing order was issued correcting the first one, stating that both unit 1 and unit 2 were affected.It still only addresses Fire Zone 80.

By 7-27-90 the Vogtle NSAC supervisor had determined that the event was a 1 hour reportable and the DC's had been elevated to SONOPCO licensing and discussions up to and including the Vice President had been held but actions to place the Plants in a safe condition and to notify the NRC within 1 hour were not initiated.

On the morning of 7-30-90 the Vogtle General Manager was discussing the reportability of this event with the Technical Support Manager and the Operations Manager.He had been told that SONOPCO engineers thought it was outside the design basis and this angered him.He stated that "Whatever engineer thinks it is outside the design basis should write a DC and sign his name on it and show exactly what criteria is violated.".This seemed very strange since a DC had already been written 11 days before identifing the deficient condition. It was the General Manager's and his operations staff's job to promptly evaluate the existing DC and make appropriate reportability calls,not to atempt to reset the time clock with another DC or make intimidating statements about the engineers.

By 7-30-90 the SONOPCO General Manager Plant Support,the Engineering and Licensing Manager, and the Licensing Manager had all concluded that it was a 1 hour reportable but still no report was made .It was said that Southern design engineering was still looking at it.

On 7-31-90 the SOPNOPCO Engineering and Licensing Manager informed the Plant that SONOPCO and SCS had decided that this condition was not outside the design basis and was not reportable as a 1 hour,a 4 hour and not even a LER,it was merly a deficiency that should be corrected.

FSAR CHAPTER 9A

ZONE	AFFECTED TRAIN	FSAR FIRE EVENT SAFE SHUTDOWN EVALUATION says to use
1-80	B	B train
1-83	B	A or B train
2-80	B	B train
2-83	B	A or B train
2-[illegible]	A	A or B train

Since the condition was initially discovered there has been the opportunity to actually fix the deficiency.The fix is simple (lifting 2 leads) and can be done at power.In addition Unit 1 had tripped off line on 7-23-90 to 7-26-90 and the work could be done as part of the forced outage. Management has not pushed, to lift the leads and the fact that the unsafe condition was not corrected immediately is apalling but also confusing.It would appear that the deemphasis is designed to support the decision to not report.

RESPONSE TO HOBBY/MOSBAUGH § 2.206 PETITION, SECTION III.3

I. Petitioners' Allegations.

The petitioners assert that GPC, through the SONOPCO Project, submitted known, false statements to the NRC intended to mislead the NRC about the reliability of the Vogtle emergency diesel generators. As basis, the petitioners allege that on April 10, 1990 Mr. Mosbaugh informed the General Manager that the "diesel air quality"[1] statements made in a Confirmation of Action response letter were false and that on April 19, 1990 Georgia Power submitted a Licensee Event Report ("LER") after Mr. Mosbaugh advised the Senior Vice President, a corporate officer located in Birmingham, that the information contained in the LER was incorrect. The petitioners further allege that the Company "intentionally delayed" revising the LER until after a June 8, 1990 presentation to the Commissioner, drafted multiple transmittal letters for the revised LER which contained "false explanations" in an attempt to "cover up" errors in the original LER, and retaliated against Mr. Mosbaugh for identifying the alleged false information submitted to the NRC.

II. GPC Response to Petitioners' Allegations.

A. Diesel Generator Pneumatic System Air Quality.

The petitioners' allegation that the Confirmation of Action response letter, dated April 9, 1990 (the "COAR"), was false and the implication that Mr. Mosbaugh informed the General Manager of the inaccurate statements in the letter prior to its transmittal to the NRC are without merit. First, the COAR was dated April 9, 1990.[2] Mr. Mosbaugh's memorandum to the Vogtle General Manager which addresses "air quality" (Exhibit 1) is dated April 10,

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[1]Diesel air quality refers to the dryness of the air in the pneumatic control system of the diesel engines.

[2]The COAR states, in part, the following:

In addition, the following actions have been or are being implemented to ensure a high state of diesel reliability . . . . GPC has reviewed air quality of the D/G air system including dew point control and has concluded that air quality is satisfactory. Initial reports of higher than expected dew points were later attributed to faulty instrumentation. This was confirmed by internal inspection of one air receiver on April 6, 1990, the periodic replacement of the control air filters last done in March, 1990 which showed no indication of corrosion[,] and daily air receiver blowdowns with no significant water discharge.

1990. Thus, Mr. Mosbaugh's alleged notification to the General Manager that the "diesel air quality statements made in the letter were false" would have occurred after the letter was transmitted to the NRC. Second, the memorandum of Mr. Mosbaugh (Exhibit 1) does not mention the COAR nor state that the air quality at that time was not satisfactory. Rather, the memorandum identifies three types of historic problems.

Third, the basis of Mr. Mosbaugh's memo is believed to be a document, consisting of five pages (Exhibit 2), also dated April 10, 1990. This memorandum also discusses historic maintenance of the diesel air dryers and suggests that dew point measurement practices needed to be investigated to ensure reliable results. The memorandum does not, however, conclude that the then current air quality was deficient.

Fourth, the COAR acknowledges initial concern associated with air quality (i.e., "initial reports of higher than expected dew points") and deficient measurement of dew point (i.e., "attributed to faulty instrumentation"). Mr. Mosbaugh, it appears, was focused on historic air quality issues based on maintenance history, and was unaware of GPC efforts relative to better instrumentation and measurement. These efforts included obtaining instrumentation from another plant.

Fifth, the COAR lists some of the activities which form the basis for the conclusion that the air quality was satisfactory (April 6, 1990 internal inspection, replacement of control air filters and daily air receiver blowdowns). In addition, Mr. Mosbaugh apparently was unaware of other technical considerations, including the views of knowledgeable engineers that the air quality of the pneumatic system was satisfactory.

Finally, the NRC Staff, thought to include Mr. Pete Taylor, reviewed the issue of the possible contribution of moisture in the diesel engine's pneumatic control system to the March 20, 1990 event. The NRC Staff, then, is more aware of the verification of adequate diesel engine air quality based on personal review than Mr. Mosbaugh, who bases his conclusion on dated information.

B. Diesel Generator Start Information.

Petitioners' allegation that the April 9, 1990 COAR and a Licensee Event Report ("LER") (90-006) dated April 19, 1990 contained known false statements intended to mislead the NRC about the reliability of the VEGP diesel generators is without merit.

The COAR states, in part, that the "A" Unit 1 diesel generator had been started 18 times, and the "B" diesel generator had been started 19 times and that no failures or problems had

occurred during any of these starts. The LER refers to both diesel generators as "having been started at least 18 times each and no failures or problems have occurred during any of these starts." As can be confirmed in statements in the custody of the NRC's Operational Safety Inspection team which reviewed this matter in August, 1990, unit control logs and shift supervisor logs were the source of the data used in developing the numbers "18" and "19" found in the COAR and the original LER. The numbers originally were included in a transparency developed by Vogtle plant personnel; this transparency was included in handouts at an April 9, 1990 meeting with the NRC in Atlanta, Georgia. The COAR, written the same day as the meeting with NRC representatives in Atlanta, adopted the "18" and "19" numbers. The LER, written later, also was predicated upon the "18" and "19" start count. Statements in the custody of the Operational Safety Inspection team confirm that both documents basically used the information developed for the April 9 transparency.

In addition, successive draft revisions of the LER have been reviewed by GPC. A version of the LER prepared by the site, dated April 17, 1990, identified "several starts" rather than specifying a number of starts (Exhibit 3, p. 6 of draft LER). An attendee at the Plant Review Board stated that a specific number should be used and the next draft version of the LER stated that the number was "more than 20 times" (Exhibit 4, p. 6). The "more than 20 times" phraseology was provided to corporate representatives in Birmingham (Exhibit 5). These representatives, with knowledge of the 18 and 19 numbers used in the transparency on April 9, questioned the "more than 20 times each" language provided by the site. More specifically, Mr. Hairston, the Senior Vice President, requested the corporate LER coordinator to "verify >20 starts." Retained copies of the LER drafts confirm other efforts by corporate representatives to verify other information and assure the accuracy of the LER (Exhibit 6).

Additional diesel generator starts had occurred subsequent to April 9, 1990 (the date of the GPC meeting in Atlanta with NRC representatives), and the final April 19th LER wording stated that each diesel engine had been started "at least 18 times each."[3] GPC was aware that NRC inspectors had followed the Company's efforts to troubleshoot and test the operability of the diesel generators and believed that the NRC had all relevant information on the diesel generators' operability and reliability. Nevertheless, either before or concurrent with the

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[3]The wording was reviewed by corporate and site representatives in a telephone conference call late on April 19, 1990. Although Mr. Hairston was not a participant in that call, he had every reason to believe the final draft LER presented to him after the call was accurate and complete.

transmittal of the LER to the NRC, the Senior Vice President instructed Mr. C. Ken McCoy, the Vice President for Vogtle, to call the NRC's Mr. Ken Brockman and to discuss the fact that the number of starts indicated in the LER differed from the number on the April 9 transparency. A phone call from Mr. McCoy was placed to Mr. Brockman on April 19, in the afternoon.

Mr. Mosbaugh and employees who reported to him controlled the development of the original LER. To the extent Mr. Mosbaugh actually had concerns about the substance of this document, he had direct and immediate ability to change the information contained in it. His own actions relative to the LER establish this fact. Indeed, as reflected in the PRB comment review sheet for its meeting No. 90-59, held on April 18, 1990 (Exhibit 4), Mr. Mosbaugh directed three changes to the draft LER, two of which he directed as "mandatory" word changes. He, therefore, had an opportunity to require any other correction. Similarly, on April 19, 1990 in a telephone conversation between the site representatives and Corporate Office representatives, he had the opportunity to suggest corrective language but, apparently, failed to do so. Not until April 30, 1990 does it appear that Mr. Mosbaugh articulated for the benefit of his management that the diesel engine start count data contained in the LER was inaccurate. At that time, he was assigned, in writing, to correct the NRC documentation (Exhibit 7). He, therefore, was tasked with correcting the inaccuracy which his Technical Support group had created by supplying "more than 20 times" wording to the Corporate Office.

In September or October, 1990, in the presence of Mr. Brian Bonser, an NRC Resident Inspector, Mr. John Aufdenkampe (the former Technical Support Manager under Mr. Mosbaugh responsible for LER development in April, 1990), stated his opinion that the LER used the numbers in the transparencies developed for the April 9, 1990 meeting with the NRC in Atlanta and that his group had merely added additional starts from and after April 9 to reach the conclusion that "more than 20" successful starts had occurred during the relevant time frame.

In addition to directing changes in documents as required, on April 30, 1990 the General Manager also verbally notified the NRC Resident Inspector of the erroneous data, as he testified to the Operational Safety Inspection team. Further, Mr. Hairston called Mr. Stewart Ebneter, the NRC Regional Administrator, on May 14 and May 24, 1990. He believes that in the longer call on May 24 he informed Mr. Ebneter that the count of successful starts in the LER was in error. He further recalls that he conveyed the then-current "correct" numbers at that time to Mr. Ebneter and informed him that revisions to the LER would be forthcoming. Mr. McCoy recalls calling the NRC's Mr. Ken Brockman about the same time and informing him of the error; telephone billing reports reflect several telephone calls from

Mr. McCoy to Mr. Brockman on May 24. The petitioners' allegation that an intentional delay in revising the LER until after a June 8, 1990 NRC meeting is founded, then, on the false premise that the revised LER was the mechanism by which the NRC first learned of the inaccuracy of the LER. Such was not the case.

On or about May 9, 1990, Mr. Mosbaugh provided a revised draft of the LER language which addressed diesel generator starts. The revised language proposed by Mr. Mosbaugh (Exhibit 8) conveys the same substantive message as the language in the April 9th COAR and the original April 19th LER. All three state that each engine had been started successfully, and none indicated failures or problems indicative of an unreliable diesel engine. Mr. Mosbaugh's proposed revision, in pertinent part, states "including the under-voltage test each engine has been successfully started eleven times with no start failures." If, as he now alleges, Mr. Mosbaugh truly had concerns related to the original LER, his inaction on April 18 (at the PRB), in the April 19 telephone conference, and his April 30 assignment from his General Manager to provide revised LER language provided him with numerous opportunities to direct revision or to revise the alleged "false statements." This he failed to do.

The allegation that GPC officers would attempt to mislead the NRC with incorrect information is, in a word, absurd. As Appendix B to NUREG-1410 indicates, from March 26, 1990 through April 17, 1990 numerous interviews and meetings were held concerning the event at Vogtle, including transcribed diesel generator meetings between the NRC and GPC. The Incident Investigation Team (IIT) reviewed voluminous plant records, including records associated with historic diesel generator operations and maintenance. Numerous informal discussions concerning diesel operability also occurred, including discussions concerning operability of the diesel generators between the General Manager and the NRC's Mr. Allen Chaffee. Extensive telephone discussions were also held between NRC and GPC after the March 20 event, including 25 calls to IIT representatives in Bethesda between April 6 and May 11, 1990 concerning the diesel generators' sensors. Many of these calls lasted for over an hour and typically involved several IIT team members. Given the widespread and extensive discussions between GPC and NRC representatives at various functional levels, the suggestion that GPC officers or upper level managers, who were aware of these efforts, would knowingly provide false information is ludicrous. The converse is the truth; their staffs were tasked to verify the information provided from the site group which was under Mr. Mosbaugh's direction and control. And, when it became apparent that information provided to the NRC was inaccurate, various GPC representatives informed the NRC of the fact.

Finally, the Petition fails to point out that Mr. Mosbaugh was removed from the PRB on May 11, 1990 as a consequence of the permanent Assistant General Manager - Support reassuming his position after completion of SRO training, not as a consequence of identifying an inaccuracy in the LER or COAR.

C. The Revised LER.

A revision to the original LER to correct the diesel generator start count was contemplated as early as April 30, 1990, as reflected by the General Manager's memo of that date (Exhibit 7). Due to the several sources of inaccuracy, as identified in GPC's August 30, 1990 letter to the NRC, a consensus on the "correct" count was not reached for some time. In addition, examination and testing of diesel engine sensors was being pursued (representatives of the IIT readily can verify the extensive, almost daily discussions with GPC representatives concerning these efforts). A draft of Revision 1 of LER 1-90-6 was approved by the PRB in PRB meeting No. 90-66 on May 8, 1990 and by the Vogtle General Manager on May 14, 1990. Mr. Mosbaugh, as Acting PRB Chairman, signed the approval sheet on this revision draft on or about May 8. Exhibit 9 reflects this approval, as well as the Technical Support Manager's May 4th approval of this draft revision, which was telecopied to the Corporate Office on May 14, 1990.

A comparison of this site-approved draft revision and the prior draft prepared just a few days before (Exhibit 10) reveals changes in diesel generator starts. As stated previously, on May 24 the Senior Vice President called Mr. Stewart Ebneter, the Region II Administrator. The Senior Vice President recalls that he supplied the Regional Administrator the then-current "correct" numbers which were "14" and "15." This recollection is confirmed by the May 14, 1990 draft of Revision 1. He also recalls informing the Regional Administrator that two revisions to the LER were then contemplated, one to correct the diesel generator start data and the other to document the results of the sensor test program.

The draft revised LER was further modified over time. Exhibit 11 is a June 11, 1990 corporate edition of the revision draft which reflects "15" and "14" starts. Exhibit 12, a site version of the revision updated to include starts through June 11, shows "14" and "11." The Senior Vice President noted about this time that the diesel generator start count data was different than previous data. Irritated at the data variation and without a satisfactory explanation of why the data was different, the Senior Vice President tasked the Safety Analysis and Engineering Review ("SAER") quality assurance group at the plant with the verification of the "correct" numbers for the LER revision.

As of June 11, 1990 the then-current draft revision of the LER identified "14 valid tests of DG1 (sic) with no valid failures" and "11 valid tests of DG1B with one valid failure" (Exhibit 12). On June 14 the Senior Vice President called the Regional Administrator again. The Senior Vice President informed the Regional Administrator that the "count" data had changed once again and was different than the information previously provided to the Regional Administrator on May 24. He also informed him that the SAER group had been assigned to conduct an audit on the numbers. The conversation reflected upper management's commitment to obtain and supply accurate information. The Senior Vice President also instructed that the NRC's Mr. Brockman or Mr. Herdt be contacted and informed about the change in "count" data; Mr. William Shipman, the General Manager - Support did so, either on June 14 or June 15, based upon telephone billing reports.

By June 15, 1990 information related to the testing of the jacket water temperature sensors had been sufficiently developed for inclusion in the revised LER (Exhibit 13).

By June 23, 1990, the Manager of Technical Support, the PRB, and the General Manager at the plant had approved a draft Revision 1 to LER 1-90-6 (Exhibit 14). Concurrently, the SAER group was conducting its comprehensive review of available diesel generator start data. As of June 28, 1990, the SAER group had reviewed diesel generator start data available and prepared a number of spread sheets comparing various data sources. These spread sheets eventually were attached to the group's report (Exhibit 15). Again, this report was developed at the request of the Senior Vice President, who instructed that a copy of the report be provided to the Resident Inspector at Vogtle.

As demonstrated by the foregoing, the delay in submittal of a revised LER was principally a function of assuring an accurate document and providing additional information concerning the temperature sensor testing. Numerous informal notifications to the NRC preceded the formal revision as well as the IIT presentation to the Commission. The various draft revisions unequivocally demonstrate an on-going evolution of LER draft revisions and the significant variation which led, appropriately, to the Senior Vice President's request for independent verification by the SAER group prior to submission of the revised LER.

D. Transmittal Letter for Revision 1 of LER 1-90-006.

Mr. Hairston instructed his staff to prepare a transmittal letter to the NRC for the revised LER which explained the differences in the count numbers between the original and revised LERs. The transmittal letter informs even the most casual reader that the revision was necessary "to correct the LER" and borrows heavily from the SAER report (Exhibit 15). The third sentence of the transmittal letter comes from the "results" section of the SAER report, page 3. The revision's shift to "valid diesel generator tests in accordance with Reg. Guide 1.108 rather than the number of successful starts since the event" is stated clearly. One key phrase is "since the event," which connotes to the knowledgeable reader a shift in the time frame for the counts from (1) after the March 20 event until April 19 (the date of the original LER) to (2) after completion of the test program (as defined in the June 29 letter) through April 19 ("10" and "12" set forth in the transmittal letter) or through June 7, 1990 ("12" and "16" set forth in LER 90-006-1, p. 6 of 9).

The petitioners ascribe nefarious intent to the fact that various drafts of the June 29, 1990 transmittal letter were prepared. The fact that several transmittal letters were prepared merely evidences the difficulty inherent in dealing with the subject matter (i.e., "tests," "valid failures," "valid tests" and "successful starts"). Further, the drafts were just that: preliminary documents which were subject to further verification and approval. None of them were the document forwarded to the NRC. Nonetheless, a review of the drafts in their full text demonstrates the on-going effort of GPC to improve the accuracy of the transmittal letter in spite of the petitioners' selective paraphrasing of their contents on page 12 of their Petition under "Explanation Contained in Draft."

First, the June 28, 1990 draft of 0751 AM Central Time (Exhibit 16) states that "only valid failures were considered in the conclusion that no problems or failures occurred" and that the number of <u>tests</u> was determined by counting <u>tests</u> regardless of whether or not the test constituted a "valid test" under regulatory definition. These would have been inaccurate statements of fact since, as established by interviews of the OSI in August, 1990, "tests" and "valid failures" were not counted by involved personnel.

The June 28, 1990 0855 draft (Exhibit 17) appears identical to the earlier draft of 0751 <u>except</u> that the revised draft appropriately deletes reference to "valid failures" and changes the key word "tests" to "starts": "the number of <u>starts</u> was determined by counting Diesel Generator <u>starts</u>...." These modifications increase the accuracy of the draft by correctly identifying, using a lay term, the things that were, indeed, counted.

The June 29, 1990 0755 (Exhibit 18) draft and 1142 draft (Exhibit 19) of the same date are each longer than the preceding draft, accurately describe the substance of the April 9, 1990 letter and focus on the wording "subsequent to the test program" in the original LER. In both instances, the draft transmittal letter explains that if the report had stated "subsequent to the event," rather than "subsequent to the test program," the LER would have been consistent with the April 9 COAR and the "18" and "19" numbers included in the transparencies provided by GPC to the NRC on April 9. This is a correct statement of fact.

The 1142 draft (Exhibit 19) includes the additional sentence: "The statement made in the LER and in the April 9 letter did not consider troubleshooting problems associated with the restarting of Diesel Generator 1B, which was out of service for maintenance at the time of the event." This, also, is a correct statement -- made with hindsight -- because the SAER report identified "successful" starts associated with non-valid tests where post-maintenance problems were identified (e.g., fuel priming) and these problems were not counted.

With each iteration additional information was added to the prior draft to provide a more complete explanation of the "count" in the original LER and April 9 letter. This is indicative of the Company's attempt to assure accurate and useful information to the NRC Staff -- a revision to the original LER, standing alone, would have resulted in a "correct" count and would have satisfied notification requirements but would not have explained why the revision was appropriate.

The June 29, 1990 1311 draft is essentially the same as the transmittal letter forwarded to the NRC, with one exception. The word "discrepancy" in the last sentence of the first paragraph (Exhibit 20) was modified to "difference" in the final version. This final wording more clearly connotes a contrast between the "count" in the transmittal letter and the "count" in the original LER.

The final LER Rev. 1 transmittal letter, then, draws on statements and conclusions made by the SAER group in its report of June 29, 1990. This makes sense, since the Senior Vice President had commissioned this effort by the group and would execute the transmittal letter. And, as can be observed by reviewing prior drafts and comparing them to the final version of the transmittal letter, the final version not only reconciles the original LER and the April 9 letter with the facts known as of June 29, 1990, but also identifies the causes of the error (e.g., recordkeeping practices and the lack of definition of the time frame of the "count" due to the vagaries associated with the original "test program" wording).

The August 30, 1990 letter (Attachment 11 to GPC's September 28, 1990 response to the Petition) from GPC to the NRC further expounded upon the differences in the "counts." The attachments to the August 30, 1990 letter contain tables which list the starts using more extensive information than used as the basis for the April 9 transparency and letter and designates starts considered "successes" under a definition which is spelled out in the text of the August 30 letter. The letter also acknowledges error on the part of the Vogtle SRO who originally compiled the "counts" in his review of operations logs.

In light of the revised LER, the information supplied to the OSI, the independent review of diesel generator start data conducted by the OSI, the August 30, 1990 letter submitted prior to the Petition, and the further information provided in this response, the lack of merit of the allegation that GPC attempted to mislead the NRC has been demonstrated exhaustively.

Relevant and controlling facts, including interviews conducted by the OSI, the text of draft documents provided in this response, and the informal notification of the "count" error in the original LER were either unknown or not provided by the petitioners. On these bases, each standing alone, the allegation is demonstrated to be without merit.

E. Request for NRC Review of Diesel Generator Performance.

The petitioners, on page 12 of the Petition, request a review of the performance records of the diesel generators. Such a review, according to the Petition, will show unreliability based upon (1) the initiation of three different design changes, (2) additional "failures" after the original LER was submitted to the NRC, and (3) the unreliability of the components, apparently the temperature sensors, which are alleged to have been known to be unreliable for years. The review requested by the petitioners has been fulfilled. First, GPC understands that several NRC representatives have reviewed the performance records of the diesel engines. Mr. Allen Chaffee, Mr. Pete Taylor, and Mr. Milt Hunt are believed to have professional opinions as to the reliability of the diesel engines. Second, while GPC cannot divine the "three" specific design changes referred to by the petitioners, the NRC Staff is intimately familiar with the performance of and design change associated with the temperature sensors of the emergency diesel generators (see, for example, NRC Staff comments filed January 11, 1991 in ASLBP No. 90-617-03-OLA). Third, revised LER 90-06 and other Special Reports to the NRC subsequent to the original LER have formally notified the NRC of additional problems, including "invalid" and "valid" "failures." Fourth, the petitioners' allegation that the reliability of "the components" was "known to be unreliable for years" is supported by no articulated fact. A documented fact is that the NRC has examined, in the context of potential

enforcement action resulting from the March 20, 1990 event, whether information available to GPC should have been identified as precursors of that event, including the failures of the temperature sensors (see Confirmation of Meeting letter dated August 22, 1990 (at page 2) and October 1, 1990 Enforcement Conference Summary letter from Mr. Luis A. Reyes (NRC) to Mr. W. G. Hairston, III (GPC)). GPC's knowledge of the components' historic reliability, therefore, has already been considered by the NRC Staff. Further review is simply not appropriate on the basis of a bald, conclusory allegation.

F. Alleged Retaliation.

The retaliation alleged by Mr. Mosbaugh is the subject matter of ongoing Department of Labor proceedings, as explained in the Enclosure to this submittal. By letter dated January 10, 1991, GPC provided the NRC with an explanation of the basis for the employment action taken with regard to Mr. Mosbaugh.

The Petition is not the appropriate vehicle for resolution of Mr. Mosbaugh's private cause of action, if any, and the requested relief is inappropriate for this employment-related matter.

III. Conclusion.

Based on the foregoing, the Company concludes that the petitioners' allegations are without merit.

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Birmingham, Alabama 35201
Telephone 205 877-7122

C. K. McCoy
Vice President, Nuclear
Vogtle Project

Georgia Power
the southern electric system

December 10, 1991
ELV-03293

Docket Nos. 50-424
50-425

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

Attn: Thomas E. Murley, Director
Office of Nuclear Reactor Regulation

Subject: Additional Information Regarding
Amended 10 CFR § 2.206 Petition Filed by
Mr. Marvin Hobby and Mr. Allen Mosbaugh

Gentlemen:

By letter dated October 3, 1991, Georgia Power Company ("GPC") responded to your August 22, 1991 request for additional information. The request for additional information sought information associated with several allegations submitted by the petitioners in a July 8, 1991 supplement to their original petition.

Subsequent to GPC's October 3, 1991 response, we have identified information which is relevant to the preparation of a Director's Decision in this matter. The NRC may be aware of most of the information provided in this letter. Nevertheless, the purpose of this letter is to assure that you have the benefit of all material and relevant facts and circumstances during your deliberations.

I. Recommended Decision and Order in Marvin Hobby v. Georgia Power Company.

By Order dated November 8, 1991 in the U.S. Department of Labor Case No. 90-ERA-30, (the "Order") the Honorable Joel R. Williams recommended to the Secretary of Labor that the complaint of Mr. Marvin Hobby be dismissed with prejudice. GPC previously had called the NRC's attention to this proceeding, as explained in the Company's response, dated September 28, 1990 and April 1, 1991, in this matter.

Georgia Power

A. Alleged False Testimony of Mr. R. Patrick McDonald.

The petitioners have repeatedly alleged that Mr. R. Patrick McDonald, the Executive Vice President of GPC responsible for nuclear operations, submitted perjured testimony during Energy Reorganization Act proceedings before the Department of Labor. Foremost is the claim that Mr. McDonald's testimony regarding the selection of SONOPCO Project staff was false and that a thorough NRC investigation would demonstrate that the staffing was made in a two-day meeting at the 270 Peachtree Street Building in Atlanta, Georgia (see, e.g., pp. 10-15 of the petitioners' July 8, 1991 "Amendments" to the original petition).

Judge Williams' Order addresses Mr. Hobby's assertions as follows:

> The meeting in preparation for the Fuchko and Yunker trial occurred six days after the memo establishing the NOCA [Nuclear Operations Contract Administration group] was issued. I find that Complainant's [Hobby's] testimony, in regard to his having been told by anybody involved in the proceeding that he would have to change any testimony that he would give in that matter to conform to that of Mr. McDonald, to be totally unbelievable. I fail to see where Respondent's [Georgia Power Company's] attorneys would even consider having the Complainant testify about the SONOPCO selection process as he was not involved in the same and any testimony he would have given relating thereto would have been nothing more than hearsay. The Complainant is unable to identify the attorney who purportedly approached him with such an incredible request. The two partner attorneys, who conducted the two sessions which the Complainant attended, have denied making such a statement and I consider them to be credible witnesses. There were two other associate attorneys present in the meeting, but the Complainant made no attempt to subpoena them to the hearing. Although he allegedly relayed the purported conversation to Mr. McHenry the next day, Mr. McHenry was not examined at the hearing in regard thereto and I decline to credit his affidavit, prepared with the Complainant's assistance 1 1/2 years after the purported event.

Order at p. 40-41.

B. Alleged Unlawful Management Direction of the Licensee.

The petitioners also alleged that Mr. McDonald received his management direction from Mr. Joseph Farley and, as a result, Georgia Power Company has improperly transferred control over its

Georgia Power

nuclear operating licenses. (See, principally, Section III.1 of the September 11, 1990 original petition.) After Judge Williams outlined the origin of this concern of Mr. Hobby (p. 21), the Judge then found as follows:

> I recognize that in addition to the memorandum, the Complainant did mention a concern, as to Mr. McDonald's receiving his management direction from Mr. Farley instead of Mr. [Dahlberg] to Mr. Evans and perhaps others. Mr. Evans did acknowledge the Complainant's having mentioned such concern 'in passing.' Depending upon the tone of such conversation, Mr. Evans could have taken the concern as the Complainant's personal one. Nevertheless, the time frame for the oral complaints is not established in the record. Mr. Smith [of Oglethorpe Power] laid the matter to rest in May 1989 upon receipt of the organization chart and Mr. Williams' memo [of May 15, 1989]. Although the Complainant continued to be concerned about the reporting relationship in June 1989, when he corresponded with Admiral Wilkinson, there is no evidence of record to establish that he continued to raise the subject with anyone beyond that time. Perhaps he had become as convinced as I am that Mr. McDonald did, in fact, take his management direction from Mr. Dahlberg in regard to the two nuclear plants owned, in part, by Georgia Power. Certainly, any doubts in his mind concerning the same should have been dispelled by the August 1989 meeting in reference to the Public Service Commission case. The evidence referable to what transpired in this meeting clearly established that Mr. Dahlberg exercised control over Mr. McDonald regarding Georgia Power's nuclear operations.

Order at 42.

C. Alleged Retaliation for Raising Concerns.

Finally, the Order is relevant to the petitioners' non-specific allegation that Georgia Power retaliates against managers who raise regulatory concerns (Item III.9(d) of the original petition).

> I find that the decision to eliminate [Mr. Hobby's] position of manager of NOCA was in no way related to the Complainant's participation in the January 2, 1989 meeting [in which he allegedly raised the concern about the accuracy of Mr. McDonald's testimony regarding the selection of employees for the SONOPCO Project] or the concern raised in his April 27, 1989 memorandum as to from whom Mr. McDonald receives his management direction for operation of the

> Georgia Power nuclear plants. I find that, instead, the decision to eliminate the position was fully justified as a measure to operate the Respondent's nuclear program more economically and efficiently.

Order at 44.

II. Vogtle Special Team Inspection Report Nos. 50-424, 425/90-19 Supplement 1, November 1, 1991.

Sections 2.1, 2.2, 2.4, 2.5, and 2.7 of the above-cited Inspection Report address, in whole or in part, allegations in Sections III.8, 6(e)(iii), 5(a), 3 and 9(d) of the original petition of September 11, 1990, respectively. Each provides a factual basis for concluding that allegations of wrongdoing are unsubstantiated, although the alleged events and technical deficiencies may have occurred. It should be noted that the NRC's inspection efforts which form the basis of the Inspection Report were initiated over a month prior to the September 11, 1990 submission of the original petition.

GPC is aware of other NRC Inspection Reports on this docket which are relevant to aspects of the Petition. First, Inspection Report 91-20, dated September 12, 1991, at page 4 of the report "Details," addresses the allegation contained in Section III.6(c) of the original petition. Second, Inspection Report 91-14, dated July 19, 1991, in particular Sections 2.b, c, f, and 3.c provide factual bases demonstrating the falsity of the general allegation contained in Section III.6 of the Petition that GPC "subverts" the requirements of Technical Specifications. Third, the two allegations in Section III.6(e)(i) and (ii) were addressed by the NRC well before the submission of the original petition. Inspection Report 90-10 dated June 14, 1990 sets forth factual conclusions relative to the two events which are the subject matter of LERs 1-90-004 and 2-90-001 and which were previously identified as non-cited violations. See, also, Inspection Report 90-19, Supplement 1, page 2, first paragraph, last sentence.

III. Petitioner's Possession of Draft LER 1-90-004.

The petitioners alleged that GPC personnel purposely violated Technical Specifications in order to keep the Vogtle Electric Generating Plant operating or to hasten the restart of a unit (Section III.6(e)(ii) of the original petition). GPC has recently obtained, in conjunction with discovery in Mr. Mosbaugh's Department of Labor proceeding, a draft (Enclosure 1) of Licensee Event Report 1-90-004, dated March 7, 1990. Notations on the draft, in what appears to be Mr. Mosbaugh's handwriting, indicate that prior to submitting the original

petition Mr. Mosbaugh knew: 1) Operations Department management identified the noncompliance of Technical Specifications on a morning status conference call, and 2) the reason for the noncompliance of Technical Specifications was that information concerning equipment out-of-service had been placed on the back of a Limiting Condition of Operation status sheet. Thus, it appears Mr. Mosbaugh possessed information prior to September 1990 which was contrary to the statements made in his September 11, 1990 petition.

IV. Participants in an April 19, 1990 Conference Call Regarding LER 1-90-006.

GPC's October 3, 1991 supplemental response sets forth the basis of GPC's April 1, 1991 statement regarding the Senior Vice President's lack of participation in a telephone conference call late on April 19, 1990 which finalized LER 1-90-006. In late October, 1991 in conjunction with discovery in Mr. Mosbaugh's Department of Labor proceeding, GPC obtained cassette audio tapes which were surreptitiously made by Mr. Mosbaugh during, approximately, the February-September, 1990 time frame. One of those tapes of April 27, 1990 discussions (identified as Tape No. 71) indicates that Mr. Hairston was not a participant during the April 19, 1990 telephone conference call when language concerning emergency diesel generator start counts was finalized in the LER. The following is a transcript of a portion of this tape which contains a discussion between Mr. Mosbaugh ("ALM") and another participant ("P") on the April 19th conference call.

- ALM: I think there is a high probability that there is a problem with their statement [in LER 90-06 concerning diesel generator start information].
- P: What George told me over the phone--
- ALM: George who?
- P: George Bockhold--
- ALM: When?
- P: Before we issued the LER.
- ALM: Yeah.
- P: We had a big conversation on those numbers with George [Bockhold], uh, [George] Hairston--
- ALM: Yeah.

- **P: --or not Hairston, [Bill] Shipman.**

- ALM: They were all on there.

- P: --and what George [Bockhold] said is they had explained to the Region that they had had--they used--Tom got those numbers from what we presented to the NRC and then he just added the additional starts after that---

- ALM: That's right.

- P: But--and we questioned that with George [Bockhold] and what George [Bockhold] said is, 'Yes, we did have failures; the Region was aware we had failures, but we were in the troubleshooting mode and once we cleared the troubleshooting mode then we had that many successful starts.'

- ALM: That's not true--

- P: That's what I was told.

- ALM: You can interpret--

- ALM: Yeah, but that was a presumption on George's part--

- P: No. George did not presume anything. He made that as a statement of fact and all that information was presented to him before he made the NRC presentation [on April 9, 1990] and that's the way he made the presentation.

- ALM: OK, well that's--

- P: George was aware of the fact that [inaudible]--

- ALM: You've got to establish, you know, you have weasel words in this thing, you've got to establish a criteria, OK, between X and X, how many successful starts do I think I had? What'. my criteria? The words in there say 'failures and problems.' What's a problem?

- P: Well, I think probably the more appropriate way would have been to word it to say, 'We have had--we have had eighteen consecutive starts without a trip from this date going back.'

- ALM: Whatever--the words in there are weasely, OK,--

- P: Well, they weren't--

- ALM: --they say 'failures and problems' and they say 'since the 20th.'

- P: They weren't intended to be weasely. From my standpoint, they weren't intended to be weasely.

- ALM: You can read those words a couple of different ways. All I'm saying is that somebody, you know, we need to decide what w s missed--

- P: You'll probably want to mention that to George [Bockhold]--

- ALM: We need to decide what we missed, then we need to review the data and see if what we meant is true or not, but I have yet to be able to figure out, among the various ways of interpreting it, I find a flaw with each method of interpreting the words.

As can be observed from the highlighted portion of this excerpt, the participant indicated that Mr. Shipman and not Mr. Hairston participated in conversations which finalized the LER. This is consistent with the collective recollection of participants during the August, 1990 special inspection, as reflected in documents enclosed in GPC's October 3, 1991 supplemental response. Moreover, the conversation indicates that Mr. Mosbaugh, as of April 27, 1990, had not reached a personal conclusion of "material false statement" relative to the LER. GPC's prior responses include an April 30, 1990 memo to Mr. Bockhold from Mr. Mosbaugh which indicates to us that Mr. Mosbaugh's conclusion regarding incorrect, material information in the LER crystallized only <u>after</u> the original LER had been forwarded to the NRC.

Georgia Power

I hope this information will be helpful to resolve these matters in an expeditious manner. If I can be of any further assistance, please do not hesitate to contact me.

Very truly yours,

C.K.McCoy

C. K. McCoy

Enclosure

cc: <u>Georgia Power Company</u>
Mr. A. W. Dahlberg
Mr. W. G. Hairston, III
Mr. W. B. Shipman
Mr. P. D. Rushton
Mr. J. T. Beckham
Mr. M. Sheibani
NORMS

<u>U. S. Nuclear Regulatory Commission</u>
Mr. S. D. Ebneter, Regional Administrator
Mr. D. S. Hood, Licensing Project Manager, NRR
Mr. B. R. Bonser, Senior Resident Inspector, Vogtle
Document Control Desk

ENCLOSURE

TITLE

LER 1-90-004

N/A / Mgr. Engr. Supt. Date	N/A / Mgr. HP/Chem. Date
[signature] / 3/7/90 Mgr. Tech. Supt. Date	N/A / Mgr. Maint. Date
N/A / Mgr. Admin. Date	N/A / Mgr. Ops. Date
N/A / Mgr. Trng. Date	N/A / Mgr. Out./Plan Date
N/A / Asst. General Mgr. - Support Date	N/A / Asst. General Mgr. - Operations Date

PRB Mtg. No.

Please review, sign where appropriate and return to Willie Smith by ____.
If there are any questions, please call Willie Smith at extension 3329.

RL90770

3-7-4

LER 1-90-004

"FAILURE TO COMPLY WITH TECHNICAL SPECIFICATION 3.0.4 OCCURS ON ENTRY INTO MODE 6"

EVENT DATE: 3-1-90

ABSTRACT:

On 3-1-90, at 0133 CST, a failure to comply with Technical Specification (T.S.) 3.0.4 occurred when Unit 1 entered Mode 6 (Refueling) from Mode 5 (Hot Shutdown). Prior to entering Mode 6, a Limiting Condition for Operation (LCO) had been initiated for Source Range Channel 1N31 to allow performance of a 18 month channel calibration. Although this LCO remained in effect, the Shift Superintendent signed off on the applicable procedure to indicate he had reviewed the LCO Book for impact on entering Mode 6 and that approval was granted to change status from Mode 5 to Mode 6. After entry into Mode 6, the Shift Superintendent recognized that T.S. 3.9.2 requires two Source Range Monitors to be operable in Mode 6 and that a failure to comply with T.S. 3.0.4 had occurred. No immediate action was required since the action requirements of T.S. 3.9.2 were satisfied.

The root cause for this event is considered as cognitive personnel error by the Shift Superintendent. The Shift Superintendent has been counseled and a copy of this LER will be placed in the Operations Required Reading Book.

RL90771

A. REQUIREMENT FOR REPORT

This report is required per 10CFR50.73(a)(2)(i) because of a failure to comply with Technical Specification (T.S.) 3.0.4.

B. UNIT STATUS AT TIME OF EVENT

Unit 1 had shutdown to commence its second refueling outage. This event occurred when Unit 1 entered Mode 6 (Refueling) from Mode 5 (Cold Shutdown). Reactor coolant temperature and pressure were approximately 110 degrees Fahrenheit and 0 psig respectively. Additionally, the Reactor Coolant System was drained to midloop and nozzle dams had been installed.

C. DESCRIPTION OF EVENT

On 2-28-90, a Limiting Condition for Operation (LCO) was entered to allow performance of 18 month surveillance 24695-1 "Nuclear Instrumentation System (NIS) Source Range Channel 1N31 Channel Calibration". Entry of the LCO for Source Range Channel 1N31 was appropriately recorded in the LCO Book and in the Unit 1 Shift Supervisor Log.

On 3-1-90, procedure 12007-C "Refueling Entry (Mode 5 to Mode 6)" was being performed in preparation for entry into Mode 6. Items (4) and (5) of step 4.3.1.c were completed by the Shift Superintendent and initialed off. Step 4.3.1.c reads: "REVIEW the following for impact on entering Mode 6: (1) Jumper and Lifted Wire Log, (2) Temporary Modification Log, (3) Equipment Clearance Log, (4) LCO Book, (5) Outstanding Work Orders." At 0014 CST, the Shift Superintendent signed off on procedure 12007-C to indicate approval to change status from Mode 5 to Mode 6. At 0133 CST, Mode 6 was entered when Reactor Vessel Head detensioning commenced.

Several hours later, the Shift Superintendent was reviewing the LCO Book in preparation for shift turnover and recognized that a failure to comply with T.S. 3.0.4 had occurred on the entry into Mode 6. At the time of the mode change, the LCO for Source Range Channel 1N31 was still in effect and the channel was still in "test" for performance of surveillance 24695-1. T.S. 3.9.2 requires two Source Range Neutron Flux Monitors to be operable in Mode 6. Therefore, the requirements of T.S. 3.0.4, which state in part "Entry into an OPERATIONAL MODE or other specified condition shall not be made unless the conditions for the Limiting Condition for Operation are met without reliance on provisions contained in the ACTION requirements", had not been fully complied with. No immediate corrective action was required due to this discovery since the action requirements of T.S. 3.9.2 were satisfied.

Not discussed at Log Book review.
But Swartz covered on 6:30 call

D. CAUSE OF EVENT

The root cause for this event is considered as cognitive personnel error on the part of the Shift Superintendent. In reviewing the LCO Book and signing off on procedure 12007-C, the Shift Superintendent should have recognized Source Range Channel 1N31 as being a mode change restraint. There were no unusual characteristics of the work location that contributed to the occurrence of this event.

"On the book"

E. ANALYSIS OF EVENT

The action requirements of T.S. 3.9.2 state that with one Source Range Neutron Flux Monitor inoperable or not operating, to immediately suspend all operations involving CORE ALTERATIONS or positive reactivity changes. These action requirements were complied with. By 1120 CST on 3-1-90, surveillance 24695-1 had been completed and the LCO for Source Range Channel 1N31 was exited at that time. Since the action requirements of T.S. 3.9.2 were complied with, it is concluded that there was no adverse effect on plant safety or on the health and safety of the public.

F. CORRECTIVE ACTIONS

1. The involved Shift Superintendent has been counseled regarding his failure to recognize 1N31 as a mode change restraint.

2. A copy of this LER will be placed in the Operations Required Reading Book to reemphasize the need to be aware of mode change restraints.

G. ADDITIONAL INFORMATION

1. Failed Component Identification

   None.

2. Previous Similar Events

   A failure to fully comply with T.S. 3.0.4 previously occurred for Unit 1 on 10-28-87 (reference LER 424/87-051), when the Unit changed status from Mode 4 (Hot Shutdown) to Mode 3 (Hot Standby) with certain required equipment having not been verified as operable prior to completing the mode change. However, the root cause for these two events differ slightly in that the earlier event resulted from a failure to implement "Information Only LCO's".

3. Energy Industry Identification System Codes

   Incore/Excore Monitoring System - IG

For

Larry Robinson

'93 JUL 14 A11:46

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

Commissioners:

Ivan Selin, Chairman
Kenneth C. Rogers
Forrest J. Remick
E. Gail de Planque

SERVED JUL 14 1993

In the Matter of GEORGIA POWER COMPANY, *et al.* (Hatch Nuclear Plant, Units 1 & 2; Vogtle Electric Generating Plant, Units 1 & 2)	Docket Nos. 50-321 50-366 50-424 50-425 (10 C.F.R. § 2.206)

MEMORANDUM AND ORDER

CLI-93-15

The Nuclear Regulatory Commission (NRC) staff's partial decision under 10 C.F.R. § 2.206, DD-93-08, 37 NRC ___ (Apr. 26, 1993), is pending before the Commission for possible review in accordance with 10 C.F.R. § 2.206(c). For the reasons stated in this order, the Commission is vacating the staff's partial decision and remanding the matters decided therein to the staff for further consideration.

The staff's partial decision responds to a petition filed by Allen L. Mosbaugh and Marvin B. Hobby in September 1990, and further supplemented in October 1990 and July 1991, which asked for initiation of proceedings and other enforcement action against Georgia Power Company (GPC). The petitioners based their

petition on various allegations of false statements, willful violations of NRC requirements, and other misconduct. In DD-93-08, the staff denied the petition with respect to certain of the petitioners' allegations which the staff believed were capable of final resolution. However, the staff declined to reach a determination with respect to allegations of unlawful discrimination against Messrs. Hobby and Mosbaugh which are related to pending proceedings before the United States Department of Labor and to other allegations of wrongdoing which are still under consideration by the NRC staff.

In addition to his filing of the section 2.206 petition, Mr. Mosbaugh has been admitted as an intervenor in a proceeding on the transfer of operating authority over the Vogtle Electric Generating Plant from GPC to Southern Nuclear Operating Company (Southern Nuclear).[1] Among the bases for his admitted consolidated contention in the adjudicatory proceeding are the allegations also contained in the section 2.206 petition that GPC and Southern Nuclear had consummated an unlawful de facto transfer of control to Southern Nuclear of the operating licenses for the Vogtle and Hatch facilities, and that GPC's executive vice president in a meeting with NRC staff on January 11, 1991, made material false statements about the formation of Southern Nuclear. The staff denied the petition on the merits with

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[1] LBP-93-5, 37 NRC 96 (1993)(appeal pending before the Commission). Mr. Hobby also petitioned to intervene in the transfer proceeding, but was denied standing to intervene. He has not appealed the denial of his intervention.

respect to these matters in DD-93-08. See slip op. at 5-17 and 53.

The Commission has generally discouraged use of section 2.206 procedures as an avenue for deciding matters that are under consideration in a pending adjudication. Thus, the Commission ordinarily would expect the staff to deny a section 2.206 petition that raises the same issues that are being considered in a pending adjudication on the basis of the pendency of the identical matters in a proceeding involving the same licensee or facility.[2] This general rule is not intended to bar petitioners from seeking immediate enforcement action from the staff in circumstances in which the presiding officer in a proceeding is not empowered to grant such relief. Moreover, we recognize here that Mr. Mosbaugh has not invoked section 2.206 to avoid a pending adjudication and that his section 2.206 petition seeks relief with respect to issues and facilities that are not before the Licensing Board in the pending transfer proceeding. However, in view of the overlap and similarity of some issues between the section 2.206 petition and the transfer proceeding (particularly those addressed in sections II.A. and II.B. of DD-93-08), the staff's final determi ation of the common issues should take into

---

[2] See General Pub. Utils. Nuclear Corp. (Three Mile Island Nuclear Station, Units 1 & 2; Oyster Creek Nuclear Generating Station), CLI-85-4, 21 NRC 561, 563-65 (1985); Pacific Gas & Elec. Co. (Diablo Canyon Nuclear Power Plant, Units 1 & 2), CLI-81-6, 13 NRC 443 (1981).

account the Licensing Board's findings and the outcome of the transfer proceeding.[3]

Apart from the commonality of some issues decided in DD-93-08 with pending issues in the adjudicatory proceeding, a common thread runs throughout the allegations raised in the section 2.206 petition. The issues raised in the petition generally concern the integrity of GPC or Southern Nuclear officers and t corporate organization responsible for operation of the Hatch and Vogtle plants. Under the particular circumstances of this case, rather than address the issues in the section 2.206 petition in a piecemeal fashion, the staff should reach a determination of all issues in an integrated manner after consideration of the remaining matters raised in the section 2.206 petition and the outcome of the transfer proceeding.

We therefore vacate DD-93-08 and remand to the staff those portions of the section 2.206 petition decided therein for the staff's further evaluation and final decision in conjunction with the staff's resolution of the other remaining matters in the petition and in light of the outcome of the transfer proceeding. In taking these actions, we intimate no view on the soundness of the staff's analysis of the issues in DD-93-08. We also do not

---

[3] We recognize that GPC has appealed the Licensing Board's admission of Mr. Mosbaugh as a party and of the consolidated contention. We expect to render a decision on the appeal in the near future. Nonetheless, at least pending a further Commission order on appeal, Mr. Mosbaugh is entitled to party status and his contention is deemed admitted.

bar the staff from taking prompt enforcement action at any time during its ongoing review of the matters raised in the petition.

It is so ORDERED.[4]

For the Commission

SAMUEL J. CHILK
Secretary of the Commission

UNITED STATES NUCLEAR REGULATORY COMMISSION

Dated at Rockville, Maryland,
this 14 day of July 1993.

[4] Commissioner de Planque did not participate in the Commission's consideration of this order.

SOUTHERN COMPANY SERVICES
NUCLEAR PLANT SUPPORT VOGTLE
INVERNESS BUILDING 42 ROOM 330

TELECOPY FORM

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Calculation Cover Sheet

Southern Company Services

Calculation Number	X4C1901S53
Project	VOGTLE ELECTRIC GENERATING PLANT
Discipline	Nuclear
Objective	DEMONSTRATE ARB LRS RELEASE CONSEQUENCE ACCEPTABILITY
REA Number	VG-9057
Subject/Title	ALTERNATE RADWASTE BUILDING LIQUID RADWASTE SYSTEM FAILURE OFFSITE DOSE ANALYSIS
Design Engineer's Signature	[signature]
Date	3-4-90
Last Page Number	11

Contents

Topic	Page	Attachments (Computer Printouts, Technical Papers, Sketches, Correspondence, etc.)	Number of Pages
Purpose of Calculation/ Summary of Conclusions	1	1. Release to Surface Water	10 ⚠1
Criteria	2		
Major Equation Sources/ Derivation Methods	2		
Assumptions	5		
Listed References	8		
Body of Calculations	10		

Safety Related: ☒ Yes ☐ No

Nonsafety-Related That Could Impact Safety-Related: ☐ Yes ☐ No

Record of Revisions

Rev. No.	Description	Originator	Date	Reviewer	Date	Proj. Eng.	Date
0	ORIGINAL ISSUE	[initials]	3-4-90	[initials]	3/5/90	[initials]	3-5-90
1	ADDED ATTACHMENT 1 - "RELEASE TO SURFACE WATER", REVISED PAGE 9 of 11	[initials]	[illegible]	[initials]	[illegible]	[initials]	[illegible]

NOTES:

Design Calculations

Southern Company Services

Project VEGP	Prepared By [signature]	Date 3-4-90
Subject/Title ARB OFFSITE DOSE ANALYSIS	Reviewed By [signature]	Date 3-5-90
	Calculation Number X4C1901S53	Sheet 1 of 11

PURPOSE: The purpose of this calculation is to demonstrate that a postulated liquid radwaste system failure in the Alternate Radwaste Building (ARB) results in offsite dose consequences which are bounded by the current (FSAR) analyses.

SUMMARY OF CONCLUSIONS:

A postulated release of ARB process water to the groundwater or atmosphere represents less than 25% of the previously analyzed activity release for the Recycle Holdup Tank and is therefore bounded by the current FSAR analyses.

Design Calculations

Southern Company Services

Project	Prepared By	Date
VEGP	[signature]	3-4-90
Subject/Title	**Reviewed By**	**Date**
ARB OFFSITE DOSE ANALYSIS	G McLeod	3-5-90
	Calculation Number	**Sheet**
	X4C 1901S53	2 of 11

CRITERIA: The analyses must be consistent with the guidance of the U.S.NRC Standard Review Plan, NUREG-0800, and be bounded by the current FSAR analyses shown in FSAR sections 15.7.2 and 15.7.3. The analyzed releases for airborne and liquid sources from FSAR Tables 15.7.2-1 and 15.7.3-1 are shown on pages 3 and 4.

METHODS:

The total postulated ARB activity released in liquid or gaseous form is calculated and compared to the previously postulated Recycle Hold-up Tank releases, to demonstrate that offsite doses would be bounded by currently existing analyses. Not demonstrated to be true.

VEGP-FSAR-15

TABLE 15.7.2-1

PARAMETERS USED IN EVALUATING THE RADIOLOGICAL CONSEQUENCES OF A LIQUID RADWASTE TANK FAILURE

Source data	
Core power level (MWt)	3565
Defective fuel (%)	1
Atmospheric dispersion factors	See table 15A-2.
Activity release data	
Noble gas activity (percent of tank contents)	100
Iodine gas activity (percent of tank contents)	1
Tank contents subject to release activity released to the environment	See table 15.7.3-1.

Nuclide	0 to 2 h (Ci)
Kr-87	5.5E+2
Kr-88	1.6E+3
Kr-89	4.6E+1
Xe-133	1.1E+5
Xe-135	3.1E+3
Xe-138	2.7E+2
I-131	1.2E+0
I-132	1.2E+0
I-133	1.8E+0
I-135	1.0E+0

Design Calculations

Southern Company Services

Project	Prepared By	Date
VEGP	[signature]	3-4-90
Subject/Title	**Reviewed By**	**Date**
ARB OFFSITE DOSE ANALYSIS	J. McZmyk	3-5-90
	Calculation Number	**Sheet**
	X4C1901S53	4 of 11

VEGP-FSAR-15

TABLE 15.7.3-1

RECYCLE HOLDUP TANK
DATA FOR FAILURE ANALYSIS

Volume of tank (gal)	112,000
Weight of liquid contained (g)	4.22×10^{8}
Radioactive contents	

Nuclide	Activity (Ci)
Kr-87	5.49×10^{2}
Kr-88	1.56×10^{3}
Kr-89	4.64×10^{1}
Xe-133	1.14×10^{5}
Xe-135	3.08×10^{3}
Xe-138	2.7×10^{2}
I-. 1	1.18×10^{2}
I-132	1.18×10^{2}
I-133	1.77×10^{2}
I-135	9.7×10^{1}
Rb-88	2.03×10^{2}
Cs-136	1.22×10^{2}
Cs-138	4.05×10^{1}

Design Calculations

Southern Company Services

Project VEGP	Prepared By [signature]	Date 3-4-90
Subject/Title ARB OFFSITE DOSE ANALYSIS	Reviewed By J. McJost	Date 3-5-90
	Calculation Number X4C1901S53	Sheet 5 of 11

ASSUMPTIONS: 1 There are no large tanks located in the ARB, thus the maximum process flow (Ref. 6) will be assumed spilled. The ARB is continuously manned and surveillance is maintained via TV cameras, hence a 30 minute time for operators to identify and isolate any leaking or failed components is assumed. This is consistent with Design Manual DC-1009 (Ref 5), and suitably conservative as the ARB vault(s) and curb are capable of containing a much larger volume of water (Ref 6) unless it sprays out.

Not true

2. Although the ARB is designed in accordance with USNRC Regulatory Guide 1.143, for this analysis the ARB structure is assumed not to contain or prevent the release of liquids to groundwater nor to prevent or reduce (via filtration) release of gases to the atmosphere. These assumptions are consistent with FSAR sections 15.7.3 and 15.7.2 (Ref. 2) and the STANDARD REVIEW PLAN 15.7.3 (Ref. 1)

3. Radioactive source terms are taken from the Radiation Analysis Manual (Ref 3). The ARB processes water originating from the Waste Monitor Tanks, Waste Holdup Tanks, or Evaporator Condensate Tanks (Ref 4). Based on a comparison of respective filter activities (Tables 6-31 and 6-33 from Ref 3) the Waste Monitor/Holdup Tanks are more limiting and the source terms from Table 6-26 of Ref 3 will be used. Note that these sources are based on 1% defective fuel and are much more conservative than required by Ref 1, but consistent with Ref. 8.

4. For airborne release analysis, a partition fraction of 0.01 will be used. This is consistent with Ref 7 and reasonable because the water processed in the ARB is well below 212°F and thus will not flash.

What is an appropriate partion factor for liquid radwaste on hot asphalt or pavement outside the ARB ?

5. Based on a comparison of Tables 6.26 and 5-15 of Reference 3, no credit has been taken for any processing, cleanup, decay, or dilution between the RCS and Waste Monitor Tanks. This is much more conservative than required by Reference 8.

6. Activity content of demineralizer and/or filter vessels is limited to 10 Ci by Technical Specifications (Ref. 10) thus meeting the acceptance criteria of Reference 1. The likelihood of a demineralizer or filter vessel failure is much less than of a line or hose failure, and if it were to occur, the activity on the contained resin/filter medium would not become airborne or be transported with groundwater (ie activity remains fixed on resin/filters). Thus no offsite doses would result from the vessel failure beyond the releases analyzed herein.

not for a non-code vessel without relief valves.

Not true, activity on powdered & bead resin would flow with surface water.

? basis?

Design Calculations

Southern Company Services

Project VEGP	Prepared By [signature]	Date 3-4-90
Subject/Title ARB OFFSITE DOSE ANALYSIS	Reviewed By J. M. McZork	Date 3-5-90
	Calculation Number X4C1901S53	Sheet 8 of 11

REFERENCES

1. USNRC STANDARD REVIEW PLAN, NUREG 0800, SECTION 15.7.3, REV 2, JULY 1981.

2. VEGP FSAR, SECTIONS 2.4.13, 15.7.2 AND 15.7.3, THROUGH AMENDMENT 39.

3. WESTINGHOUSE RADIATION ANALYSIS MANUAL, 1/2 X6AA10-95-3, DATED 6-2-83.

4. VEGP P&ID's 1X4DB124 REV 26
2X4DB124 REV 21
2X4DB125 REV 14

5. VEGP DESIGN MANUAL, DC-1009, SINGLE FAILURE, PARAGRAPH 4.0, REV 2 THRU DMCN 1009-3, 9-10-87.

6. VEGP MECHANICAL CALCULATION, X4C1901S52, REV 0

7. USNRC NUREG-0017, "CALCULATION OF RELEASES OF RADIOACTIVE MATERIALS IN GASEOUS AND LIQUID EFFLUENTS FROM PRESSURIZED WATER REACTORS".

8. USNRC NUREG 0133, "PREPARATION OF RADIOLOGICAL EFFLUENT TECH SPECS FOR NUCLEAR POWER PLANTS"

Design Calculations

Southern Company Services

Project VEGP	Prepared By [signature]	Date 3-4-90
Subject/Title ARB OFFSITE DOSE ANALYSIS	Reviewed By J. M'Ford	Date 3-5-90
	Calculation Number X4C1901S53	Sheet 9 of 11

9. Code of Federal Regulations, Title 10, Part 20, Appendix B, January 1, 1989.

10. VEGP Technical Specifications 3/4.11.1.4

ATTACHMENTS:

ATTACHMENT 1 — ARB OFFSITE DOSE ANALYSIS "RELEASE TO SURFACE WATER"

3 jw 6-26-90
JMM 6-27-90

Design Calculations

Southern Company Services

Project	Prepared By	Date
VEGP	J. [signature]	3-4-90
Subject/Title: ARB OFFSITE DOSE ANALYSIS	Reviewed By: J. M'[signature]	Date: 3-5-90
	Calculation Number: X4C1901553	Sheet 10 of 11

CALCULATIONS (LIQUID RELEASE):

Waste Monitor Tank specific activity given in Ref. 3, Table 6-26 is:

Nuclide	Activity (μCi/gm)
I-131	2.8
I-132	2.8
I-133	4.2
I-135	2.3
Cs-134	2.3
Cs-136	2.9

A 30 minute release (assumptions) at 125 gpm (Ref. 6) will result in 3750 gal or 1.42×10^7 gm of water spilling on the ARB floor, containing the following activity (Ci):

I-131	40
I-132	40
I-133	60
I-135	33
Cs-134	33
Cs-136	41

These are approximately 34% of the activities postulated to be released due to a Recycle Holdup Tank failure (see page 4) except for Cs-134 which was not considered in FSAR Table 15.7.3-1.

Form No. 8-334A

For Cs-134 the half-life is approximately 2 years, hence the groundwater travel time of 123 years (Ref. 2) will result in a concentration of approximately

$$(2.3\ \mu Ci/gm) \times .5^{(123/2)} \times (1\ ml/gm) \approx 10^{-18}\ \mu Ci/ml$$

This is negligeably small compared to 10 CFR 20 limits (9x10^{-6} μCi/ml, Ref 9), and meets the acceptance criteria of References 1 and 2.

GASEOUS RELEASE :

For the same release to the ARB as above, the airborne release would be 1% of the liquid release (assumption 4):

I-131	0.40
I-132	0.40
I-133	0.60
I-135	0.33

These are approximately 33% of the activities postulated to be released due to a Recycle Holdup Tank failure (see page 3).

Design Calculations | Southern Company Services

Project	Prepared By	Date
VOGTLE ELECTRIC GENERATING PLANT	[signature]	6/26/90
Subject/Title	**Reviewed By**	**Date**
ARB OFFSITE DOSE ANALYSIS	J. N[illegible] - Alternate Calc P.9-10	6/27/90
ATT 6 - Release to Surface Water	**Calculation Number** X4C1901.S053	**Sheet** 1 of 10

PURPOSE: The purpose of this attachment is to demonstrate that postulated release of liquid radwaste in the ARB by spray leakage out of the structure through the storm drain system to the Savannah River is within the limits of 10 CFR 20, Appendix B (Reference 9)..

This calc. addresses only liquid release to unrestricted areas. It does not address gaseous release to unrestricted area via the spraying and exiting the ARB pathway (ie ~~[illegible]~~ evolution of gaseous output from liquid on the ground outside AR.

SUMMARY OF CONCLUSIONS:

The hypothetical releases are demonstrated to be within the limits of 10 CFR 20 and are less limiting than those examined in the body of the calculations (25% vs 35% of previously analyzed limits).

Not Demonstrated to be true.

Design Calculations

Southern Company Services

Project	Prepared By	Date
VOGTLE ELECTRIC GENERATING PLANT	[illegible]	4/26/90
Subject/Title: ARB OFFSITE DOSE ANALYSIS	Reviewed By: [illegible] - AH. Calc p. 9-10	Date: 6.27.90
Att. 2 - Release to Surface Water	Calculation Number: X4C 1901, S053	Sheet 2 of 10

CRITERIA: Per FSAR 2.4.13 (ref 15) the annual average releases must meet 10 CFR 20, Table II column 2 averaged over 1 year. This is true consistent with the NRC SER (NUREG 1137, Ref 17). but other guidelines as in App I and part 21 apply as well.

METHODS: The released activity concentration and dilution volumes are calculated and used to determine hypothetical river concentrations for each isotope of concern. This assumes that Savanna River is the point at which activity is released off GaPc Property. These concentrations are then compared to the individual limits of 10 CFR 20 and are summed to demonstrate that the fraction of Maximum Permissible Concentration ≤ 1.0.

The requirements of Pt 20 Table for "unrestricted areas" applies at the Protected Area Fence if the activity is "exposed" as in an open drainage trench. Where "contained" in a discharge pipe the boundary can be at the ...

Design Calculations

Southern Company Services

Project VOGTLE ELECTRIC GENERATING PLANT	Prepared By J O Wechsemly	Date 6/26/90
Subject/Title ARB OFFSITE DOSE ANALYSIS	Reviewed By Alternate Calc /P. 9	Date 6/28/90
Att 1 - Release to Surface Water	Calculation Number X4C 1901, S053	Sheet 13 of 20

ASSUMPTIONS: In addition to assumptions discussed on pages 5 through 7:

7. Assume dilution factors for the storm drain inc plant discharge and a factor of 10 for near field dilution in the river per reference 15.

For the storm drain discharge, the area bounded by elevation 225 to the North and West of the plant to the Savannah River is assumed to drain through the general area. Per table 2.3.3-1 of reference 15, the annual rainfall is 43 inches and per Figure 1.2.2-1 of reference 15, the area is approximately 4500 ft x 4500 ft or

$$4500 \times 4500 \times 43/12 \times 7.5 = 5.4 \times 10^8 \text{ gal}$$

This ignores the 11000 gpm dilution from the plant liquid waste discharge ($\approx 5.8 \times 10^9$ gal/yr) described in FSAR section 11.3.4 (reference 15).

Invalid as the activity is exposed unrestricted (uncontrolled for the purpose of protection against exposure radiation) areas long before it leaves Georgia Power property, flows in a stream off Georgia Power property & enters the Savannah River.

No

Storm drain flows to east and southeast

wrong Area

Totally invalid as stream does not enter river at the liquid discharge point!

Design Calculations

Southern Company Services

Project	Prepared By	Date
VOGTLE ELECTRIC GENERATING PLANT	[illegible signature]	6/26/90
Subject/Title	**Reviewed By**	**Date**
ARB OFFSITE DOSE ANALYSIS	Alternate Calc (R.W.)	6/27/90
ATT 1 - Release to Surface Water	**Calculation Number**	**Sheet**
	X4C 1901, SA53	4 of 10

Looks like this calc is stretching assumptions to get under the limits.

8. Because this evaluation is to be judged against normal annual plant release limits, the liquid will be assumed released from a "critical crack" in a moderate energy system in accordance with reference 11. The source will be assumed to be at 100 psi (reference 13) and an enveloping pipe size of 3 inches will be assumed (reference 4). ~~As shown on page 7, this is the "break" flow.~~

A hose blow off will release 25 gpm or more and is perhaps more credible than a crack.

9. Based on a review of ARB drawings (reference 16) the sprayed water would be expected to run down the ARB walls to the top of the curb. It is assumed that 1/2 the water flows into the ARB (ie is not released) and 1/2 flows out of the ARB to be collected in the storm drain system and released.

Ok

Design Calculations

Southern Company Services

Project: VOGTLE ELECTRIC GENERATING PLANT	Prepared By: [signature]	Date: 6/26/90
Subject/Title: ARB OFFSITE DOSE ANALYSIS	Reviewed By: J. N. [signature]	Date: 6/27/90
ATT 1 - Release to Surface Water	Calculation Number: X4C 1901. S053	Sheet: 5 of 18

10. Because the evaluation is to be judged against normal annual release limits, the source terms used will be based on normal expected concentrations. As these concentrations are not provided in reference 3, the activities shown on page 10 will be multiplied by the ratio of normal/design basis RCS activities in Tables 5-15, 7-4 and 7-5 of reference 3. For I and Cs isotopes, the ratios are 1/10 and 1/260 respectively.

faident calc should use ailed fuel limit f 1% not hese numbers.

Note: For Cs-134 the ratio of expected to design should be 1/75, however this will not change the conclusions ($2.5\times10^{-4}\times\frac{260}{75}\div$ ... vs 0.25)

[signature] 6-27-90

Design Calculations

Southern Company Services

Project	Prepared By	Date
VOGTLE ELECTRIC GENERATING PLANT	[illegible signature]	6/26/90
Subject/Title: ARB OFFSITE DOSE ANALYSIS	Reviewed By: [illegible signature]	Date: 6/27/90
ATT 1 - Release to Surface Water	Calculation Number: X4C 1901, 5055	Sheet 6 of 10

REFERENCES:

In addition to the references cited on pages 8 and 9, the following were used:

11. VEGP DESIGN MANUAL, DC-1018, PIPE BREAK-INTERDISCIPLINE, REV 26 THROUGH DMCW 1018-16, 8-13-87.

12. VEGP DESIGN MANUAL, DC-2153, STORM DRAIN SYSTEM, REV 6, ~~THRU 10-83~~.

13. Letter SG-BGBW, Dated 1-3-90, Subject: Follow up Response for REA VG-9057, File X7BDH2.

14. CRANE TECHNICAL PAPER NO. 410, FLOW OF FLUIDS THROUGH VALVES FITTINGS AND PIPE, 1988.

15. VEGP FSAR, SECTIONS 2.4.2/3, 2.3.2 AND 11.3.4, UPDATE REVISION N, 3/30/90.

16. PROJECT DRAWINGS (ARB DETAILS & STORM DRAIN LAYOUT)

AX1AR26-41-2
AX2D88-1042 Rev 1
AX2D46-T015 Rev 7

17. NUREG 1137, Safety Evaluation Report related to the operation of Vogtle Electric Generating Plant, Units 1 and 2, June 1985.

Design Calculations Southern Company Services

Project	Prepared By	Date
VOGTLE ELECTRIC GENERATING PLANT	[signature]	6/26/90
Subject/Title	Reviewed By	Date
ARB OFFSITE DOSE ANALYSIS	J. M. [signature]	6/27/90
ATT 1 - Release to Surface Water	Calculation Number X4C 1901. S053	Sheet 7 of 10.

CALCULATIONS:

Per reference 11, the postulated break is a critical crack with an equivalent ~~area~~ of 1/2 internal diameter x 1/2 wall thickness or a diameter of:

$$D = 2\sqrt{1/4 \times 3.068 \times 0.216/\pi} = 0.46 \text{ inches}$$

Why not calculate runout flow for a hose blew off event?

~~Then the spray flow rate~~ is

$$Q(gpm) = 19.65\, d^2 \sqrt{\Delta p/K_i} \quad \text{(Ref 14)}$$

where $d = 0.46$ in

$\Delta p = 100$ psi $= 281$ ft (Ref 13)

$K = 1.0$ (exit only) (Ref 14)

$= 63$ gpm

Credit for operator action

Consistent with operator action, this continues for 30 minutes. The resultant release volume is

$(63 \times 30) = 1890$ gal.

Design Calculations | Southern Company Services

Project	Prepared By	Date
VOGTLE ELECTRIC GENERATING PLANT	[signature]	6/26/90
Subject/Title	**Reviewed By**	**Date**
ARB OFFSITE DOSE ANALYSIS	See Alternate Calc	6/27/90
Att 1 - Release to Surface Water	**Calculation Number** X4C 1901, S053	**Sheet** 8 of 10

For each isotope the concentration in the river is:

$$C(\mu Ci/gm) = \frac{\text{Release volume (gal)} \times \text{Source concentration} (\mu Ci/gm)}{\text{Dilution volume (gal)} \times \text{River dilution}}$$

$$\times \text{Well runoff} \times (\text{Expected / Design Ratio})$$

Invalid dilution

$$= \frac{(6.3 \times 80) \times \text{Source} (\mu Ci/gm) \times \tfrac{1}{2} \times \text{Ratio}}{8.4 \times 10^{8} \times 10}$$

$$= 1.75 \times 10^{-8} \times \text{Source} (\mu Ci/gm) \times \text{Ratio}$$

Then for:

Nuclide	Source ($\mu Ci/gm$)	Ratio	Concentration ($\mu Ci/gm$)	Limit ($\mu Ci/gm$)
I-131	2.8	1/10	4.9×10^{-8}	3×10^{-7}
-132	2.8	1/10	4.9×10^{-8}	8×10^{-6}
-133	4.2	1/10	7.4×10^{-8}	1×10^{-6}
135	2.3	1/10	4.0×10^{-8}	4×10^{-6}
Cs-134	2.3	1/200	2.0×10^{-9}	9×10^{-6}
-136	2.9	1/200	2.5×10^{-9}	9×10^{-6}

where the limit is from reference 9.

The total fraction of Maximum Permissible Concentration

is $\sum \frac{\text{Concentration}}{\text{Limit}} = 0.25$, well within the limit.

Alternate Calculation of Offsite Releases From ARB Pipe Crack

Purpose:

This alternate calculation is performed to verify the conclusion in J. A. Wehrenberg's calculation that the offsite effect of the failure of a pipe in the ARB is within the limits of 10CFR20.

Scope:

This calculation addresses only the offsite releases from the primary calculation. The release rates are the same as those in the primary except as noted in the assumptions.

Conclusion:

Not when you use the right dilution factors.

This alternate calculation clearly demonstrates that a failure of a pipe in the ARB with resultant spray out the building will not create a condition which would cause the plant's annual average release concentrations to exceed the limits of 10CFR20. The computed value is 43% of the 10CFR20 limit. Given the different assumptions this agrees well with the value of 25% in Mr. Wehrenberg's calculation. While slightly higher than the previous FSAR value of 35%, the computed release is well within the regulatory limit.

Assumptions:

The same basic assumptions apply as stated in the principal calculation except that 100% of the water is assumed to be released instead of the 50% assumed by Mr. Wehrenberg. The effect of this assumption is to double the releases.

Method:

Invalid as design basis dilution assumes release flows in "plant discharge" & this does not!

The alternate calc uses FSAR Table 11.2.3-2 as a starting point. That table is a listing of the expected releases of radioactivity per year from the plant for each isotope of concern. The total annual release of I-131 was divided by the projected concentration to obtain a design basis dilution. The activity released from the ARB pipe crack in thirty minutes is then computed for each of the major isotopes by multiplying its concentration (in uCi/ml) by 63 GPM times thirty minutes times the factor to convert gal to ml. That value is added to the FSAR release value to obtain a total release. That release is then multiplied by the dilution factor to obtain a projected annual average concentration. The new average concentrations are summed and the total displayed at the bottom of the page.

NOTE: The following is an alternate calculation for the assessment of the impact of a pipe failure in the ARB on the plant's compliance with 10CFR20. This calculation is based upon the dilution factor derived from FSAR Table 11.2.3-2.

Estimate of Effect of Postulated ARB Radwaste Line Failure on 10CFR20 Compliance

Dilution Factor = 2.5E+13 Spray Rate = 63 GPM

Isotope		I-131	I-132	I-133	I-134	Cs-134	Cs-136
FSAR TBL 11.2.3-2 Release	(uCi)	1.10E+05	5.50E+03	8.50E+04	8.00E+01	2.10E+04	2.60E+03
FSAR TBL 11.2.3-2 Conc	(uCi/ml)	4.46E-09	2.23E-10	3.45E-09	8.11E-13	8.51E-10	1.05E-10
Spray Event							
Release Conc.	(uCi/gm)	2.80E-01	2.80E-01	4.20E-01	2.30E-01	3.07E-02	3.87E-02
Total Release	(uCi)	2.00E+06	2.00E+06	3.00E+06	1.65E+06	2.19E+05	2.77E+05
Combined Release							
Total Release	(uCi)	2.11E+06	2.01E+06	3.09E+06	1.65E+06	2.40E+05	2.79E+05
Release Conc.	(uCi/gm)	8.57E-08	8.14E-08	1.25E-07	6.67E-08	9.75E-09	1.13E-08
Maximum Perm. Conc.	(uCi/ml)	3.00E-07	8.00E-06	1.00E-06	2.00E-05	9.00E-06	9.00E-05
Maximum Perm. Conc.	(ratio)	2.86E-01	1.02E-02	1.25E-01	3.34E-03	1.08E-03	1.26E-04

Sum of MPC ratios = 0.426 Invalid result

LOS ANGELES POWER DIVISION

VOGTLE ELECTRIC GENERATING PLANT

CALCULATION TITLE SHEET

JOB NO. 9510

PAGE 1 OF 23

SUBJECT ALTERNATE RADWASTE BLDG BASEMAT

DISCIPLINE C/S

FILE NO. NA

CALC. NO. X2CD14-093

ORIGINATOR SIG. Amir H Tahbaz DATE 5-20-86

QUALITY CLASSIF. 626

CHECKER SIG. Larry P. Posey DATE 5-27-86

NO. LAST PAGE 23

ORIGINAL ISSUE

	NAME	ACTION REQ'D	DATE	SIGNATURE
GROUP LEADER	S. THOMAS	✓	5/27/86	Stephen K Thomas
GS	R. MALIN	✓	5.30.86	R Malin
SPECIALIST				
CHIEF				
OTHER				

RECORD OF REVISIONS

NO.	DESCRIPTION	ENG.	CKR	GL	GS	SPEC.	CHIEF	DATE
1	Revised SHT. 12, Added shts. 22a, 22b	RW	NSP	SKT	RSP	NA	NA	12/5/86
2	REVISED SHT. 19 ADDED SHTS 19 a, b, c	PC	JT	JK	JTK FOR DDN	NA	NA	5/17/88

THIS CALC IS APPLICABLE TO:
UNIT 1 [ ], 2 [ ] A [✓]
RE J.m. Lovekamp DATE 7-14-88
EG J.m. Lovekamp DATE 7-14-88

TOTAL NO. OF PAGES = 28

SIGNATURE LEGEND

AMIR H. TAHBAZ	Amir H. Tahbaz
LARRY P. POSEY	Larry P. Posey (LPP)
STEPHEN K THOMAS	Stephen K Thomas SKT
R. Malin	R. Malin
N. S. PATEL	N. S. Patel, NSP
R. S. PLATONI	Roberto Platoni RSP
PETER CHANG	P Chang (PC)
Jim Tehranchi	Jim Tehranchi (JT)
JOSEPH T. KASSAS	[illegible] (JK)

CALCULATION CHECKED BY ALTERNATE MEANS

REV. NO.	YES	NO
0		✓
1		✓
2		✓

NOTICE

THESE DESIGN CALCULATIONS ARE ONLY AN ISOLATED PART OF THE COMPLETE DESIGN FOR THE SYSTEM THEY CONCERN, AND ARE SUBJECT TO BEING TAKEN OUT OF CONTEXT, MISINTERPRETED OR MISCONSTRUED IF USED WITHOUT BECHTEL POWER CORPORATION'S DIRECT PARTICIPATION

LAO 0513 8

CALCULATION SHEET

CALC. NO. X2CD14.09-

SIGNATURE Amr H Tabbaz DATE 5-20-86 CHECKED [signature] DATE 5-27-86

PROJECT VNP JOB NO. 7510-008

SUBJECT ALTERNATE RADWASTE BUILDING BASEMAT SHEET 2 OF 23 SHEETS

TABLE OF CONTENTS

LAO 0513 8-7

CALCULATION SHEET

CALC. NO. X2CD14-093

SIGNATURE [illegible] DATE 5-5-86 CHECKED [illegible] DATE 5-27-86

PROJECT VNP JOB NO. 9510-008

SUBJECT ALTERNATE RADWASTE BUILDING BASEMAT SHEET 3 OF 33 SHEETS

GENERAL DESCRIPTION

THE ALTERNATE RADWASTE BUILDING IS A CATEGORY II BUILDING LOCATED BETWEEN THE EXISTING RADWASTE TRANSFER BUILDING AND NSCW VALVE HOUSE 1B. THE BASEMAT WILL BE APPROX 40' x 62'-8" AND WILL SUPPORT A PREFABRICATED BUILDING AND ALSO IT SHOULD BE CAPABLE OF SUPPORTING OTHER EQUIPMENT PLUS THE DEAD WEIGHT OF A TRUCK & TRAILER AND A PEDESTAL CRANE. THERE WILL BE A ROOM WITH 2' THICK WALLS WHICH WILL HOUSE THE DEMINERALIZERS. THERE ALSO WILL BE A FLOOR LEAK DETECTING DRAIN WHICH WILL BE EMBEDED IN THE BASEMAT. SINCE THE DEMINERALIZER ROOM "WILL" BE DESIGNED FOR DBE CONDITION, THE BASEMAT NEEDS TO BE DESIGNED FOR THE SAME BECAUSE THE LEAK DETECTING DRAIN WILL BE EMBEDED IN THE SLAB.

REFERENCES

1. VNP DESIGN MANUAL DC-1000-C REV. 3
2. ACI 318-71
3. UBC 1976 EDITION
4. FOUNDATION ANALYSIS AND DESIGN 2nd EDITION by JOSEPH E. BOWLES
5. TIME SAVER STANDARDS FOR ARCH DESIGN DATA JOHN HANCOCK CALLENDER FIFTH EDITION.
6. BUILDING SYSTEMS & PRODUCTS CATALOG by ARMCO
7. [illegible] CRANE [illegible] FOR 5 TON CAPACITY
8. MEMO FROM ALLEN NAKASHIMA ON EQUIPMENT WEIGHTS. (FILE X6AK07)

[illegible]

1. f'c = 4000 PSI
2. fy = 60,000 [illegible]
3. WIND LOAD = 45 PSF
4. SEISMIC [illegible] SECTION [illegible] SEE [illegible] REF 3

LAO 0513 8

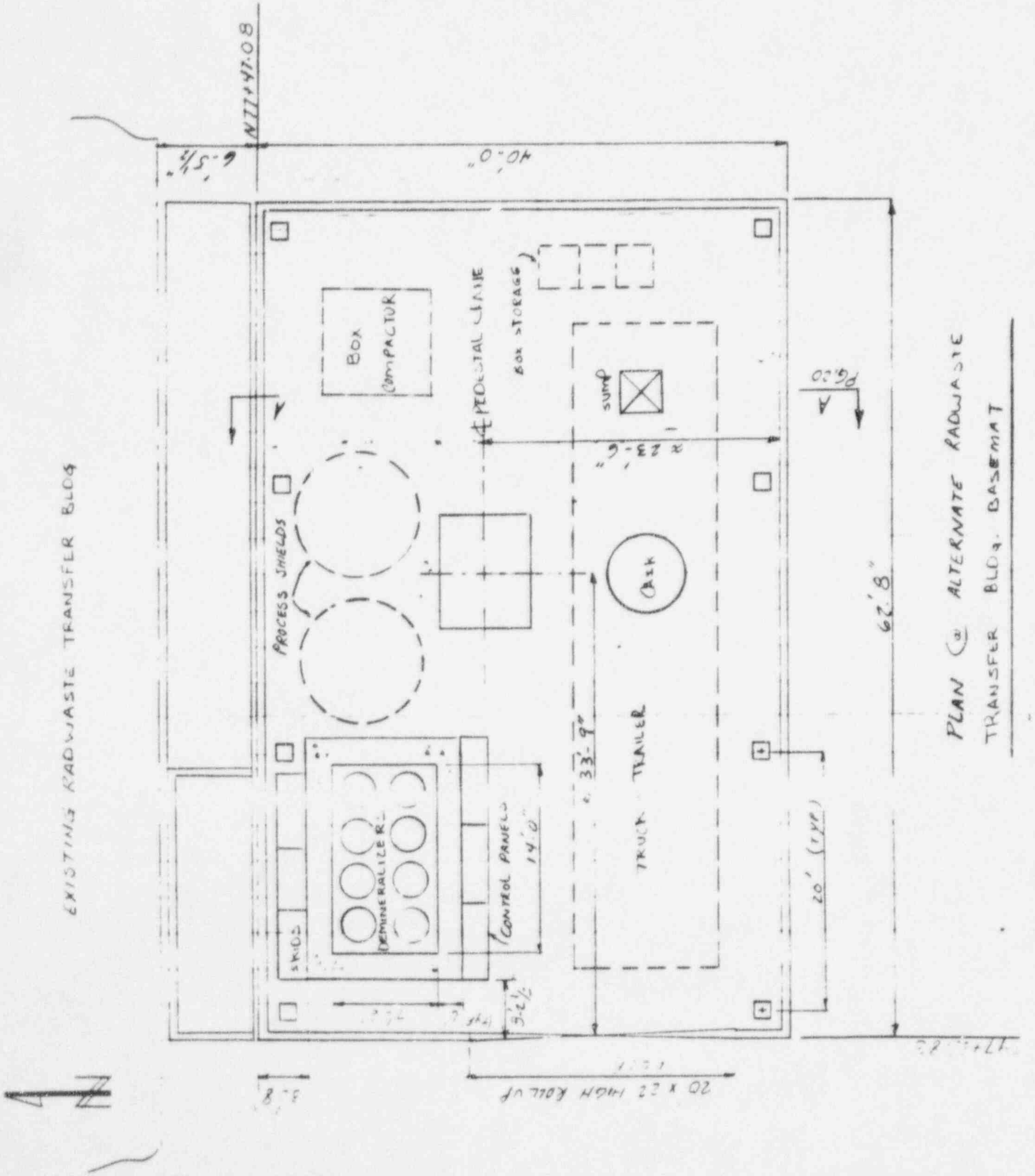
CALCULATION SHEET

CALC. NO. X2C014.093

SIGNATURE Amir H. Tantaz DATE 5-14-86 CHECKED [illegible] DATE 5-27-86

PROJECT VNP JOB NO. 9510-001

SUBJECT ALTERNATE RADWASTE BLDG [illegible] SHEET 4 OF 23 SHEETS



CALCULATION SHEET

LOADS (REF 7 & 8)

DEAD LOADS (EQUIPMENT)

PROCESS SHIELDS 2 @ 80^k	= 160.0^k
PEDESTAL CRANE	= 35.0^k (CONSERVATIVE)
DEMINERELIZERS 8 @ 2^k	= 16.0^k
SKIDS 3 @ 0.6^k	= 1.8^k
CONTROL PANELS	1.0^k
MISC (ASSUME)	2.0^k
	216.0^k

D.L. OF SHIELD "WALLS" & BASEMAT (ASSUME 2'-6" MAT)

$D.L. = [2(2')(7.5) + 2(2')(18')](0.15)(8') = 122.5^k$

$D.L. = [(140')(62.67)(2.5) + \underbrace{(185')(0.67)(0.5')}_{CURB}]\,0.15 = 949.0^k$

BUILDING DEAD LOADS

ROOFING AND SIDING ASSUMED, BASED ON ARMCO BUILDING SYSTEM RIGID FRAME DESIGN RF-80 SYSTEM, 24 GAGE STEEL PANELS

PSF REF 5, PAGE 104 CORRUGATED STEEL	2 PSF
INSULATION [illegible] INTERIOR STEEL LINER	2 PSF
MISC	1 PSF
	5 PSF

ROOFING

[illegible] 5 PSF (46.5)(60) = 13,950 lbs

[illegible] PURLINS [illegible] 4 PSF [illegible] = 11,[illegible] lbs

[illegible] WT = 35 (50 [illegible])(8 [illegible]) = 14,000 lbs

WALLS (PANELS)

(5 PSF)(37.67)(35) + (5 PSF)(46.5)(35) + (5 PSF)(46.5)(35)$\overset{APPROX}{=}$ 57,242.0 lbs

LAO 0513 B

CALCULATION SHEET

BECHTEL

CALC. NO. X2CD14.093

SIGNATURE Amir H. Tabaz DATE 5-5-86 CHECKED [illegible] DATE 5-27-86

PROJECT VNP JOB NO. 9510-208

SUBJECT ALTERNATE RADWASTE G.C. BASEMAT SHEET 6 OF 23 SHEETS

D.L. CALCULATION CONT'D

TOTAL D.L = 216.0 (EQUIP) + 122.5 (3 walls) + 949 (SLAB) + 13.95 (ROOF) + 11.16 (ROOF FR.) + 14.0 (COL.) + 27.2 (WALLS) = 1354.0 KIPS

LIVE LOADS:

FLOOR LIVE LOAD: 250 PSF (REF 1, SECTION 5.3.1 G)(FOR ADDITIONAL EQUIP.

TRUCK + TRAILER + CASK = 77.0 KIPS

SNOW LOAD = 30 PSF (ROOF L.L.) NON CONCURRENT WITH SEISMIC OR WIND

FOR FLOOR L.L ASSUME APPROX ONE HALF THE FLOOR SPACE AVAILABLE

250 PSF x 62.67 x 40 x 0.5 x 10⁻³ = 313. KIPS

SNOW LOAD = 30 PSF x 62.67 x 40 x 10⁻³ = 75. KIPS

WIND LOAD = 45 PSF

SEISMIC

UBC: V = ZIKCSW

Z = 3/8 I = 1.0 CS = 0.14 K = 0.67

V = .375 x 1.0 x 0.14 x 0.67 x W = 0.035 W

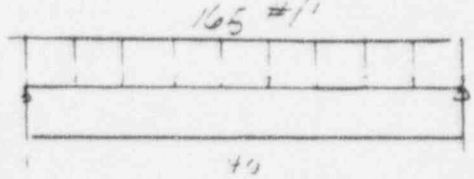
BY INSPECTION UBC VALUE DOES NOT GOVERN OVER OBE VALUES OF REF. 1

IN ORDER TO USE THE PROPER ACCELERATION NATURAL FREQUENCY OF SLAB SHOULD BE CALCULATED.

GROSS SOIL BEARING PRESSURE = $\frac{1354}{62.67 \times 40}$ = 0.54 KSF x 10³ = 540. PSF

NET PRESS = 540 − (2.5')(150) = 165 PSF

$\Delta = \frac{5wL^4}{384EI}$



$I = \frac{bh^3}{12} = \frac{(12)(30)^3}{12}$ = 27000 in⁴

$E_c = 150^{1.5} \cdot 33 \sqrt{4000}$ = 3.83 x 10⁶ PSI

CALCULATION SHEET

BECHTEL

LAO 0513.8

CALC. NO. XCC014-092

SIGNATURE Amr H. Tarboz DATE 5-5-86 CHECKED [illegible] DATE 5-27-86

PROJECT VVP JOB NO. 9510-008

SUBJECT ALTERNATE RADWASTE BLDG BASEMAT SHEET 7 OF 23 SHEETS

FREQUENCY CALC. CONT'D

$$\Delta = \frac{5(165)(45\times12)^4}{384\times3.83\times10^{6}\times27000\times12} = 0.091 \text{ IN}$$

$$T = 2\pi\left(\frac{\Delta}{g}\right)^{1/2}$$

$$T = 2\pi\left(\frac{0.091}{386.4}\right)^{1/2} = 0.0966 \therefore \text{AMPLIFICATION IS REQ'D}$$

FOR [illegible] STRUCT 4% DAMPING MUST BE USED PER REF 1 TABLE. SINCE NO 4% CURVE IS AVAILABLE, LINEAR INTERPOLATION WILL BE USED TO CALCULATE ACCELERATION FOR 4% DAMPING. LINEAR INTERPOLATION IS ALLOWED BY REGULATORY GUIDE 1.60

$\underline{a_{V\,OBE}} = 0.32g$

$\underline{a_{H\,OBE}} = 0.12g$ SINCE MOST OF FLEXIBILITY WILL BE DUE TO THE SUPERSTRUCTURE AND THE BASEMAT IS RIGID IN THE HORIZ. DIRECTION IT IS JUSTIFIED TO USE ZPA VALUE FOR DESIGN.

(NOTE: SUPERSTRUCTURE IS ONLY 4.5% OF TOTAL MASS)

FOUNDATION DESIGN

TRY FOUNDATION MAT [illegible] THICK TO CLEAR ALL PLUMBING AND THE LEAK DETECTING DRAIN. THE TOP ELEV OF SLAB WILL BE [illegible]. THE THICKNESS CHOSEN [illegible] SATISFY REF 1 REQUIREMENT FOR DEPTH OF FROST PENETRATION.

TO DETERMINE IF THE MAT IS RIGID OR FLEXIBLE, THE FOLLOWING FORMUL FROM REF 4 Pg. 272.

$$\lambda L = \sqrt[4]{\frac{[illegible]}{4EI}} \quad \text{WHERE}$$

[illegible]

[illegible] = 4000

[illegible] = [illegible]/12 = [illegible]/12 = [illegible]

LAO 0513 6

CALCULATION SHEET

BECHTEL

CALC. NO. X2CQ14.09

SIGNATURE Amr H Tarcaf DATE 5-6-86 CHECKED [illegible] DATE 5-27-86

PROJECT VIIP JOB NO. 9510-001

SUBJECT [illegible] SHEET 8 OF 23 SHEETS

RIGIDITY CHECK CONT'D

$L = 40'$

K_S = MODULUS OF SUBGRADE REACTION, USING VESIC EQUATION

$K_S = K_S B$

$$K_S = KSB = 0.65 \sqrt[12]{\frac{E_S B^4}{EI}} \cdot \frac{E_S}{1-\mu^2}$$

E_S = ELASTIC MODULUS OF COMPACTED FILL = 1430 KSF } REF. 1

μ POISSON RATIO OF SOIL = .4

$$K_S = 0.65 \sqrt[12]{\frac{(1430)(60.67)^4}{(551,520.)(81.6)}} \cdot \frac{1430}{1-(.4)^2} = 1854.0 \text{ KSF}$$

$$\lambda L = \sqrt[4]{\frac{(1854.0)(40)^4}{(4)(551,520)(81.6)}} = 2.27$$

FROM REF 4 PG. 275 RIGID MEMBERS $\lambda L < \frac{\pi}{4}$ (BENDING NOT INFLUENCED MUCH BY K_S)

FLEXIBLE MEMBERS $\lambda L > \pi$ (BENDING HEAVILY LOCALIZED)

$\frac{\pi}{4} = .79 < 2.27 < \pi = 3.14$

∴ IT CAN BE CONCLUDED THAT THE MAT IS SEMI-RIGID. BUT DESIGN WILL ASSUME A RIGID MAT, WHICH IS CONSIDERED TO BE CONSERVATIVE AS LONG AS THE SAME REINFORCEMENT IS PROVIDED IN TOP & BOTTOM FOR THE HIGHEST DESIGN MOMENT.

$f = \frac{N}{A} + [illegible] \pm \frac{eM_x}{S_x} \pm \frac{[illegible]}{[illegible]}$

DETERMINE MOMENT DUE TO [illegible]

$M = [illegible] h = 18.7 \left(\frac{35}{[illegible]}\right) = 727\ 'K$

LAO 0513 8-73

CALCULATION SHEET

CALC. NO. X2CD14.093

SIGNATURE [illegible] DATE 5-6-86 CHECKED [illegible] DATE 5-27-86

PROJECT VNP JOB NO. 9510.008

SUBJECT ALTERNATE RADWASTE BLDG BASEMAT SHEET 9 OF 23 SHEETS

DETERMINE MOMENT DUE TO SEISMIC:

THE ONLY DEAD LOADS THAT CONTRIBUTE TO OVERTURNING ARE THE PEDESTAL CRANE AND THE SHIELD WALLS. ALL OTHER EQUIPMENT IS NOT ANCHORED TO SLAB AND THEREFORE WILL NOT CONTRIBUTE TO OVERTURNING MOMENT. NO L.L. WILL BE CONSIDERED FOR O.T.

$M_{OT\ TOTAL} = \Sigma M \times L$

DIST. FROM BOTT. OF SLAB TO C.G.

$M_{O.T.\ TOTAL} = [(35^K)(12.33') + (122.5)(6.5') + (94?)(1.25') + (13.75+11.16)(37.5') + (14+27.2)(20')]\ 0.12 = 501.'K$

* PEDESTAL CRANE WILL BE MOUNTED ON FLOOR ON TOP OF 2-W36 BEAMS OR POST ∴ $L \approx 2.5 + 5.48 + 1.33 + 3.0' = 12.33$ (REF. 7)

WF POST

HT. TO PIN

TRANS. BEAM

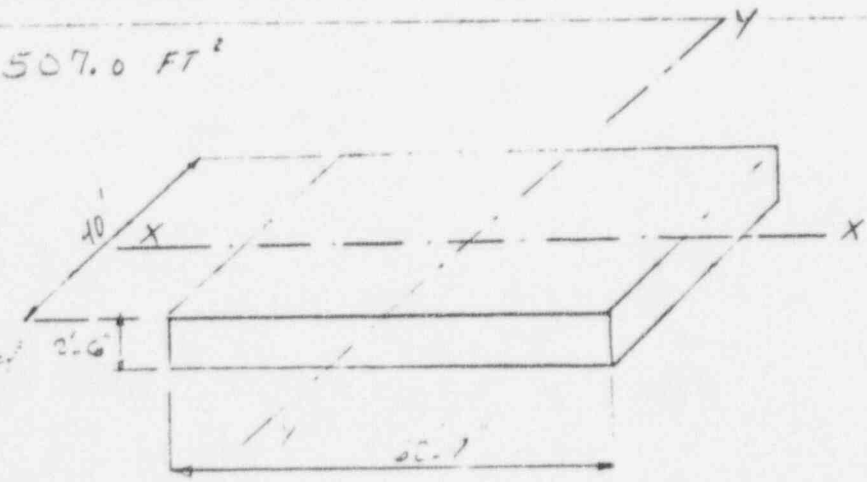
CALCULATE SECTIONAL PROPERTY OF BASEMAT

AREA = $40' \times 62.67 = 2,507.0\ FT^2$

$S_{XX} = \frac{1}{6} bd^2$

$S_{XX} = \frac{1}{6}(62.67)(40)^2 = 16,712\ FT^3$

$S_{YY} = \frac{1}{6}(40)(62.67)^2 = 26,182\ FT^3$



LAO 0513 8

BECHTEL

CALCULATION SHEET

CALC. NO. X2CD14.09.

SIGNATURE Amr H. Ibrahim DATE 5-7-86 CHECKED LGP DATE 5-27-86

PROJECT VNP JOB NO. 9510-008

SUBJECT ALTERNATE RACEWAY ... CASEMAT SHEET 10 OF 23 SHEETS

TO DETERMINE THE MAXIMUM SOIL PRESSURE THE FOLLOWING TWO UNFACTORED LOAD COMBINATIONS WILL BE CONSIDERED.

A. $D + L + W$

B. $D + L + E_{OBE}$

NOTE 1) TO INCLUDE FOR THREE DIRECTIONAL SEISMIC THE FOLLOWING COMBINATION WILL BE USED. BY INSPECTION THIS COMBINATION WILL GOVERN

$$1.0\alpha + 0.4\beta + 0.4\gamma$$

NOTE 2) i) SNOW LOAD AND 45 PSF LOAD SHALL BE NON CONCURRENT.

ii) ONLY 25% OF FLOOR LIVE LOAD WILL BE CONSIDERED WITH SEISMIC.

(iii) TRUCK LOAD WILL ONLY BE CONSIDERED IN THE VERTICAL SEISMIC

SEE THE FOLLOWING PAGE FOR COMBINATIONS.

CALCULATION SHEET

LAO 0513 B

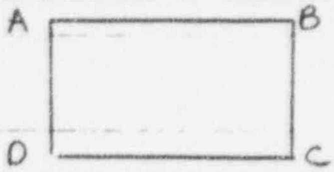
CALC. NO. X2CD14.093

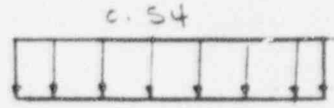
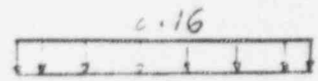
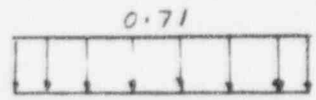
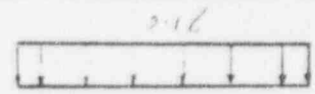
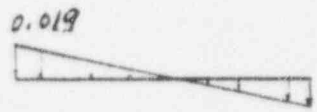
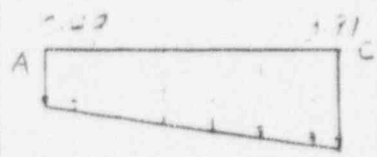
SIGNATURE Amr H. Ibrahim DATE 5-7-86 CHECKED [illegible] DATE 5-27-86

PROJECT WIP JOB NO. 9510-001

SUBJECT ALTERNATE RADWASTE BLDG BASEMAT SHEET 11 OF 13 SHEETS

LOAD COMBINATION CONT'D



	COMBINATION A	SOIL PRESS. (KSF)
①	D.L. PRESSURE = $\frac{1354.}{2,507}$ = 0.54 KSF	
②	L.L. PRESSURE = $\frac{313+97}{2,507}$ = 0.16 KSF	
③	WIND PRESSURE = $\frac{1727.0}{16,712.0}$ = 0.10 KSF	
	TOTAL = [0.54 + 0.16 + 0.10] = 0.80_{max} KSF [0.54 + 0.16 − 0.10] = 0.60_{min} KSF	
	COMBINATION B	SOIL PRESS. (KSF)
①	D.L. PRESS. = 0.54 + $\frac{1354 \times 0.30}{2,507}$ = 0.71 KSF	
②	L.L. PRESS = 0.16 + $\frac{325 \times 0.25 \times 388}{2,507}$ + $\frac{97 \times [illegible]}{2,507}$ = 0.18 KSF (FOR L.L. + SNOW)	
③	SEISMIC [illegible] = 0.4 × $\frac{501}{[illegible]}$ + 0.4 $\frac{501}{20,[illegible]3}$ = 0.019	
	[illegible] 0.019 = 0.91 [illegible] 0.54 + [illegible] 6 − [illegible] − 0.019 = [illegible] 49 [illegible]	

CALCULATION SHEET

SOIL PRESS CONT'D

MAX SOIL PRESS = 0.91 < 8.2 KSF (REF 1 SEC. 4.2.2A) ∴ OK

SINCE THE BASEMAT SHALL BE DESIGNED FOR OBE CONDITION, CAT I LOAD COMBINATIONS OF REF 1 SHALL BE USED. BY INSPECTION, TWO LOAD COMBINATIONS WILL PRODUCE THE MAXIMUM DESIGN PRESS FROM WHICH THE WORST CASE WILL BE USED FOR DESIGN.

① U = 1.4D + 1.7L + 1.7W

② U = 1.4D + 1.7L + 1.9E

FOR DESIGN, MAX PRESS. WILL BE USED AS A UNIFORMLY DISTRIBUTED LOAD OVER THE AREA OF SLAB.

① U = 1.4(0.54) + 1.7(0.16) + 1.7(0.[illegible]) = 1.2 KSF

② U = 1.4(0.54) + 1.7(0.16) + 1.9(0.17 + 0.0[illegible] + 0.019) = 1.42 KSF

0.71 - 0.54

NET DESIGN PRESS = 1.42 - (0.15 × 0.5 × 1.4) - 1.7(0.[illegible]) = 0.92 KSF

DESIGN OF BASEMAT:

[illegible] THE BASEMAT, A [illegible] SUPPORTED BEAM WITH [illegible] OF 36.0 [illegible] WILL BE CONSIDERED WITH [illegible]

[illegible] NEXT PAGE

LAO 0513 B

CALCULATION SHEET

CALC. NO. X2CD14.09.

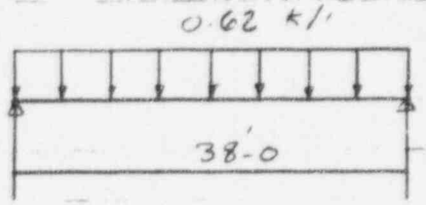
SIGNATURE [illegible] DATE 5-8-86 CHECKED [illegible] DATE 5-27-86

PROJECT VPIP JOB NO. 7512-208

SUBJECT [illegible] SHEET 13 OF 23 SHEETS

SLAB DESIGN CONT'D :

$m = \frac{wl^2}{8} = \frac{0.62(27)^2}{8} = 112$ 'K/'



$F = \frac{bd^2}{12000} = \frac{12(27)^2}{12000} = 0.73$

$Ku = \frac{Mu}{F} = \frac{112}{0.73} = 153$

USE a_u = 4.38

$A_s = \frac{Mu}{a_u \times d} = \frac{112 [illegible]}{4.38 \times 27} = 0.95$ USE #9 @ 12" W/ $A_s = 1.0$ IN²

∴ OK

CHECK MIN STEEL REQUIREMENTS OF REF 2 VS. DESIGN STEEL

FOR FLEXURE : $\rho_{min} = \frac{200}{F_y} \times bd = 0.0033 \times 12 \times 27 = 1.07$ in² ≈ 1.0
#9 @ 12" IS OK T & B.

TEMP STEEL = $0.0018 \times 12 \times 30 = 0.64$ IN² = 0.32 IN²/' /FACE

IN ORDER TO INSURE THAT THE [illegible] IS CAPABLE OF SPANNING THE [illegible] [illegible] SHOWN, [illegible] WILL BE CHECKED SPANNING 10'-0

$M = \frac{wl^2}{8} = \frac{1.42(10)^2}{8} = 17.75$ K/' < 112 ∴ OK

[illegible] Ku [illegible] $\frac{17.75}{[illegible]} = 24.3$ ∴ USE [illegible]

$\frac{17.75}{[illegible] \times 27}$ [illegible]

[illegible] TO SATISFY REF 2 [illegible] 0.15 × 1.33 = 0.2 < 0.32 ∴ OK

CALCULATION SHEET

CALC. NO. X2CD14.093

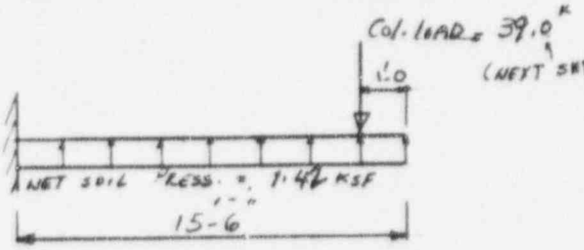
SIGNATURE Amu H. Tohbaz DATE 5-14-86 CHECKED LQP DATE 5-27-86

PROJECT VNP JOB NO. 9510-008

SUBJECT ALTERNATE RADWASTE BLDG BASEMAT SHEET 14 OF 23 SHEETS

CHECK BASEMAT FOR CANTILEVER ACTION:

SINCE THE BASEMAT EXTENDS BEYOND THE SOUTH FACE OF THE EAST-WEST RUN OF TUNNEL BELOW (1T28), AND BASEMAT IS THICKEND OVER THE WALLS OF TUNNELS 1T28 & 1T5B, THE MAT SHOULD BE CHECKED FOR CANTILEVER ACTION SPANNING 15'-6".

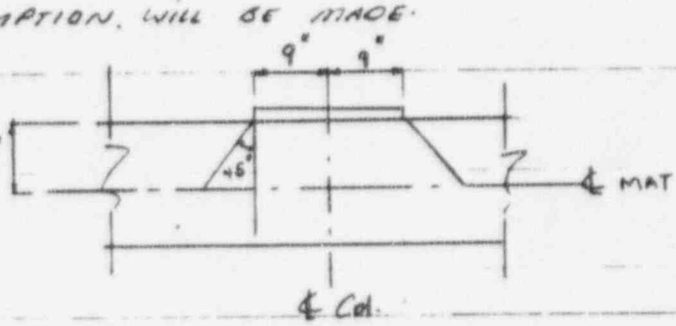


TO CALCULATE THE EFFECTIVE WIDTH OF BASEMAT RESISTING THE BENDING, THE FOLLOWING ASSUMPTION WILL BE MADE.

EFFECTIVE WIDTH IS:

$18 + 2(\tan 45°)(15) = 48" = 4'$

UNIFORM SOIL PRESS. = 4×1.42 = 5.7 K/'



CALCULATE Col. LOADS:

TOTAL BLDG D.L. = 13.95^k + 11.16^k + 14.0^k + 27.2 = 66.^k

L.L. (SNOW) = 75^k

D.L./Col. = $\frac{66}{8}$ = 8.25^k

L.L./Col = $\frac{75}{8}$ = 9.3^k

D.L./Col (OBE) = 8.25 × 0.32 = 2.6^k

L.L./col (OBE) = 9.3 × 0.32 = 3.0^k

TO THE ABOVE NUMBERS SHOULD BE ADDED TO THE ADDITIONAL LOAD DUE TO OVERTURNING OF BUILDING. FOR THIS PURPOSE, ONLY SEISMIC IN TWO DIRECTIONS (1.0V - .4H) WILL BE CONSIDERED

a_v = 0.32g a_H = 0.129 (ASSUME SEISMIC FORCE ACT AT 2/3 × HEIGHT)

LAO 0513 8

CALCULATION SHEET

CALC. NO. X2CD14.09:

SIGNATURE Amir H. Tahbaz DATE 5-14-86 CHECKED [illegible] DATE 5-27-86

PROJECT YNP JOB NO. 9510-008

SUBJECT ALTERNATE RADWASTE BLDG. BASEMAT SHEET 10 OF 23 SHEETS

COL. LOADS CONT'D:

D.L./Col. DUE TO O T = $\frac{0.4 \times 8.25 \times 2 \times 35 \times 0.12}{40 \times 3} = 0.23^k$

L.L/Col. " " " = $\frac{0.4 \times 9.37 \times 2 \times 35 \times 0.12}{40 \times 3} = 0.26^k$

$U = 1.4(8.25) + 1.7(9.37) + 1.9(0.23 + 0.26 + 2.6 + 3.0) = 39.0^k$

$M_{NET} = 39.0^k(14.5) - \frac{5.7(15.5)^2}{2} = -120.0\ 'K$

MOMENT CAPACITY OF SECTION:

$M_d = \phi \rho F_y b d^2 (1 - 0.59 \rho \frac{F_y}{f'_c})$

IN A 4'-0 SECTION THERE WILL BE AT LEAST 4-#9 BARS w/ As = 4.0□"

$\rho = \frac{As}{bd} = \frac{4}{48 \times 27} = 0.0031$

$M_d = \left[(0.90)(0.0031)(60,000)(48)(27)^2\right]\left[1 - 0.59(0.0031)\left(\frac{60,000}{4,000}\right)\right] \times 10^{-3} \times \frac{1}{12}$

$= 474.7\ 'K$ ∴ #9 @ 12" T & B IS O.K

LAO 0513 B

CALCULATION SHEET

CALC. NO. X2CD14.09?

SIGNATURE [illegible] DATE 5-8-86 CHECKED [illegible] DATE 5-27-86

PROJECT VNP JOB NO. 9510-001

SUBJECT [illegible] SHEET 10 OF 23 SHEETS

SHEAR CHECK:

V_A = 0.32 × $\frac{37}{2}$ = 11.8 [illegible]

$V_u = \frac{11.8 \times 1000}{0.85 \times 12 \times 27} = 42.8\,psi < 2\sqrt{f'_c} = 126.5\,psi$ ∴ O.K

[illegible]:

COEFFICIENT OF STATIC FRICTION = 0.45 REF 1 SECTION 4.2.6.F.

(c) [illegible]ECTION, [illegible]

0.4↑ + 1.0H⃗ + 0.4H⃗

FOR THIS CASE [illegible] WILL BE EXCLUDED.

TOTAL D.L = 1354[K] [illegible]

EFFECTIVE D.L = 1354 − 0.32 × 0.4 (1354) = 1180.0[K]

RESISTING FORCE = 1180 × 0.45 = 531[K]

SLIDING FORCE = [illegible] × 1354 + [illegible]354 = [illegible]

$F.S_{OBE} = \frac{531}{[illegible]}$ = [illegible] > 1.5 ∴ O.K

SLIDING (WIND) = 98.7[K]

$F.S = \frac{531}{98.7}$ = [illegible] ∴ O.K

[illegible]

[illegible] 354 ($\frac{40}{2}$) = [illegible][K]

$F.S = \frac{[illegible]}{1707}$ = [illegible] > [illegible] ∴ O.K

BECHTEL

CALCULATION SHEET

LAO 0513 B

CALC. NO. X2CD14.09:

SIGNATURE Amir H. Tanbaz DATE 5-9-86 CHECKED [illegible] DATE 5-27-86

PROJECT VNP JOB NO. 9510-008

SUBJECT ALTERNATE RADWASTE BLDG. BASEMAT SHEET 17 OF 83 SHEETS

CHECK SLAB FOR MOMENT DUE TO PEDESTAL CRANE:

THE PEDESTAL CRANE WILL BE SUPPORTED BY TWO W36 BEAMS OR 4 POSTS WHICH WILL BE ANCHORED TO FLOOR. THE WIDTH OF THE CRANE BASE IS ≈ 7'-0. THEREFORE THIS CONCENTRATED MOMENT DUE TO CRANE WILL BE RESISTED BY A BEAM 2'-6" THICK AND AT LEAST 7'-0 WIDE.

CHECK MOMENT CAPACITY OF BEAM

$$M_d = \phi \rho f_y b d^2 \left(1 - 0.59 \rho \frac{f_y}{f'_c}\right)$$

$\phi = 0.90$ $b = 84"$

$d = 27$ $A_s = 8 \times .44 = 3.52\ in^2$

$$\rho = \frac{A_s}{bd} = \frac{3.52}{84 \times 27} = 0.0016$$

$$M_d = \left[0.9 \times 0.0016 \times 60{,}000 \times 84 \times (27)^2 \left(1 - 0.59 \times 0.0016 \frac{60{,}000}{4000}\right)\right] \times \frac{1}{12} \times (10^{-3})$$

$M_d = 434.6\ 'K$

BASED ON INFORMATION OBTAINED FROM SFC NUCLEAR OPERATIONS* THE MAXIMUM LIFT WILL BE 23000 lbs @ 12' RADIUS.

* JOHANN REITER & JULIAN DANIEL.

MAX. OVERTURNING = 23.0 × 12 = 276 'K × 1.5 = 414.0 'K < 434.6 ∴ o.k. (LOAD FACTOR)

NOTE 1) THIS APPROACH IS VERY CONSERVATIVE BECAUSE IT DOES NOT TAKE IN TO ACCOUNT THE COUNTER ACTING MOMENT [illegible]

2) SINCE THE CRANE IS INSIDE THE BLDG. NO WIND LOAD WILL BE CONSIDERED ON THE CRANE.

LAO 0513 B

CALCULATION SHEET

CALC. NO. X2C014-01

SIGNATURE Amm H Tzoo DATE 5-30-86 CHECKED [illegible] DATE 5-30-86

PROJECT VNP JOB NO. 9510-008

SUBJECT ALTERNATE RADWASTE BLDG BASEMENT SHEET 18 OF 23 SHEETS

DESIGN OF SHIELD WALLS INSIDE THE BLDG.

THE SHIELD WALLS INSIDE THE BUILDING HOUSE THE DEMINERALIZERS. THE WALLS ARE REQUIRED TO BE ABLE TO RETAIN LIQUID FOLLOWING AN OBE EVENT. THE WALL WILL BE 2'-0 THICK AND 8'-0 HIGH. THE 2'-0 THICKNESS IS REQUIRE FOR SHIELDING PURPOSES.

TWO CASES WILL BE CONSIDERED SEPERATELY.

1) SEISMIC EVENT (INERTIA OF WALL)

2) AFTER SEISMIC EVENT WITH THE ROOM FULL OF WATER (CONSERVATIVE)

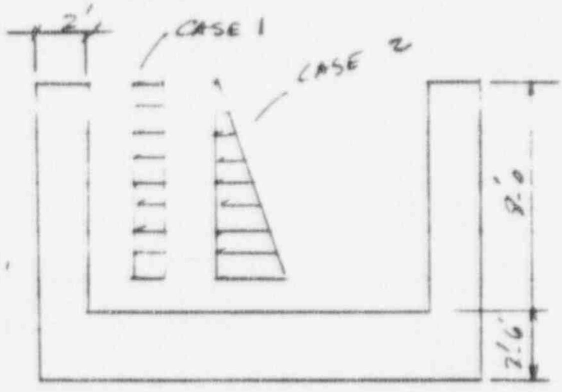
CASE 1 : FOR SEISMIC ACCEL. USE 0.12 (HORIZ)

WALL INERTIA = $0.15 \times 2' \times 0.12 = 0.036$ KSF

$U = 1.4\,D.L + 1.7\,L + 1.9\,E$

$U = 1.9(0.036) = 0.068$ KSF

$M_{max} = \frac{wl^2}{2} = \frac{0.068(8)^2}{2} = 2.2\ 'K/'$



TYP SECTION

CASE 2

$P_{water} = 8 \times 62.4 = 500$ PSF $= 0.5$ KSF

$U = 1.7 \times .5 = 0.85$ KSF

LAO 0513 1

CALCULATION SHEET

1 P. Chang 3/16/88 JT 3-17-88

CALC. NO. 13C014.09

SIGNATURE Amr H. Tanbaz DATE 5-30-86 CHECKED [illegible] DATE 5-30-86

PROJECT VNP JOB NO. 9510-008

SUBJECT ALTERNATE RADWASTE BLDG. BASEMAT SHEET 19 OF 23 SHEETS

SHEET 19a FOLLOWS

SHIELD WALL DESIGN CONT'D:

$M_{max} = \frac{WL}{3} = \frac{0.85 \times 8 \times 8}{2 \times 3} = 9.0$ 'K/' GOVERNS

$F = \frac{(12)(21)^2}{12000} = 0.44$

$K_u = \frac{9}{0.44} = 21$ USE $a_u = 4.45$

$A_s = \frac{9}{4.45 \times 21} = 0.1\ in^2/'$

CHECK MIN. STEEL:

$A_{s(min)} = 0.0033 \times 12 \times 21 = 0.83 > 0.1$ ∴ USE #9 VERT E.F. @ 12".

TEMP STEEL = 0.0018 × 12 × 24 = 0.52 in^2 USE #6 @ 12"

HORIZ. STEEL W/ AS = 0.88 > 0.52 ∴ OK

BECHTEL

CALCULATION SHEET

PROJECT VNP   JOB NO. 9510   CALC. NO. X2CD 14.093

SHEET 19b FOLLOWS

SUBJECT ALTERNATE RADWASTE BLDG BASEMAT   SHEET NO. 19a

REV	ORIGINATOR	DATE	CHECKER	DATE	REV	ORIGINATOR	DATE	CHECKER	DATE	REV. INDI-CATOR
2	P Chang	3/16/88	Jim Tehranchi	3-17-88						

NEW DEMINERALIZER VAULT

(A) BACKGROUND:

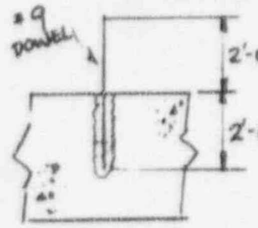
<A.1> PER REF. 1 AND 2, A NEW VAULT IS REQ'D

<A.2> VAULT DESIGN DETAILS SEE REF. 3 & NEXT TWO SHEETS (19b & 19c)

(B) ENGINEERING EVALUATION:

<B.1> SINCE THE NEW VAULT IS IDENTICAL TO THE EXISTING ONE ALL VAULT DIMENSIONS AND REINFORCEMENT MAY BE ADOPTED FROM PREVIOUS DESIGN EXCEPT DOWEL DETAILS AND BASEMAT. NEED TO BE CHECKED

<B.2> CHECK EMBEDMENT LENGTH OF #9 × 4'-0" DOWELS:



- FROM PAGE 19: THE REQ'D FLEXTURE STEEL $A_s = 0.1\ in^2/FT$
- PER REF. 4 SECT. 10.5.1: $A_{s(min)} = 0.1 \times 1.33 = 0.13\ in^2/FT$

  SAY #4 $A_s = 0.2\ in^2 > 0.13\ in^2$
- PER REF. 5, FOR #4 BAR, $f'_c = 4000\ psi$

  MIN. EMBEDMENT LENGTH = 12" < 24" PROVIDED O.K

∴ 24" EMBEDMENT LENGTH FOR DOWELS IS ADEQUATE.

<B.3> BASEMAT IS ADEQUATE BECAUSE REVIEW OF PAGE 5 TO 7 SHOWS THAT THE ADDITIONAL DEAD WEIGHT FROM NEW VAULT WILL HAVE LITTLE EFFECT TO THE BASEMAT FREQUENCY AND THE NEW VAULT ON THE BASEMAT WILL INCREASE THE RIGIDITY OF THE BASEMAT. ∴ BASEMAT IS O.K

(C) REFERENCES:

1 MEMO BB-55187 FROM D CAPITO TO D NIEHOFF DATE 3/8/88 FILE # X4B P26

2 MEMO BB-55192 FROM W HUSSEY TO D NIEHOFF DATE 3/8/88 FILE # X6A K07

3. DCR 88-VCN0061, REV. 0

4 ACI 318-71

5 DWG No. AX2D94V020 REV. 11

6 DWG No. AX2D65A024 REV. 1

CALCULATION SHEET

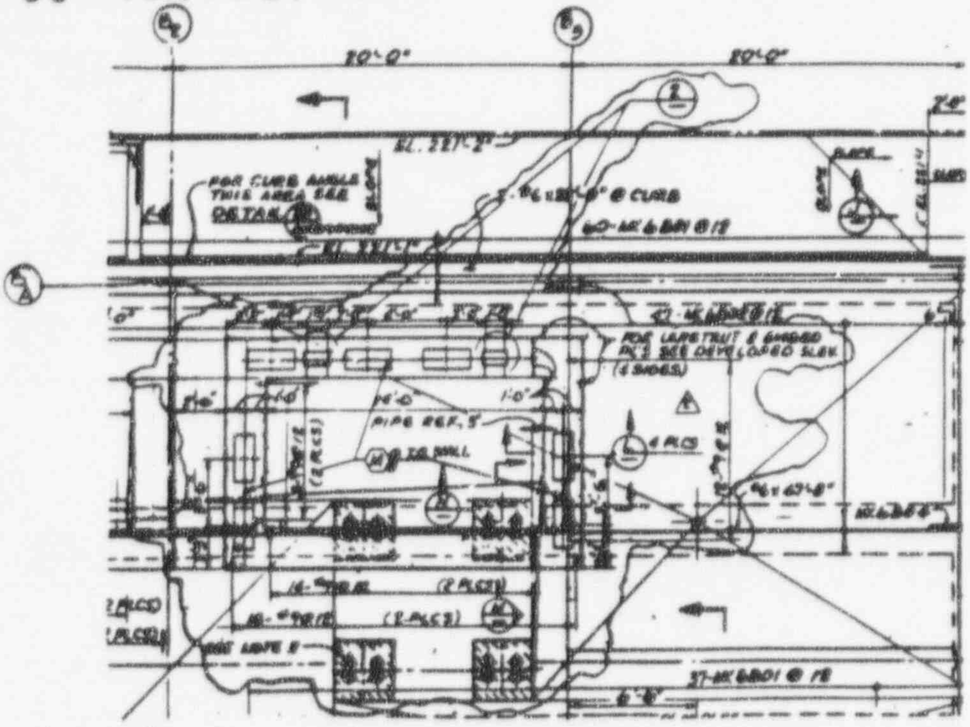
PROJECT VNP JOB NO. 9510 CALC. NO. X2CD14.093

SHEET 19C FOLLOWS

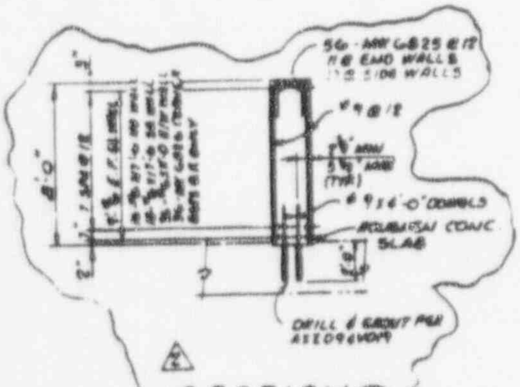
SUBJECT ALTERNATE RADWASTE BLDG BASEMAT SHEET NO. 19b

REV	ORIGINATOR	DATE	CHECKER	DATE	REV	ORIGINATOR	DATE	CHECKER	DATE
2	P Chang	3/16/88	Jim Tehranchi	5-10-88					

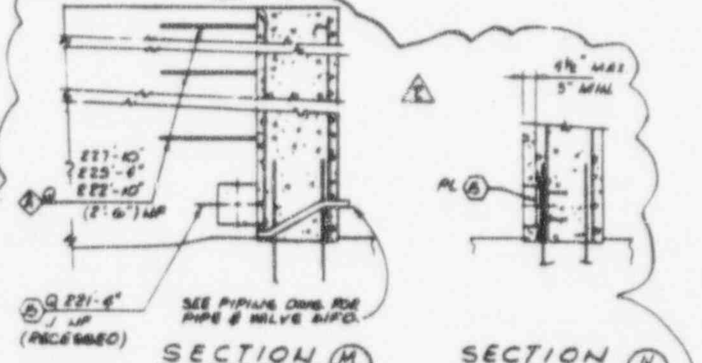
DCR 88-VCN0061 REV. 0



BASE MAT PLAN



SECTION L



SECTION M
1/2"=1'-0"

SECTION N
1/2"=1'-0"

CALCULATION SHEET

PROJECT VNP JOB NO. 9510 CALC. NO. X2CD14.093

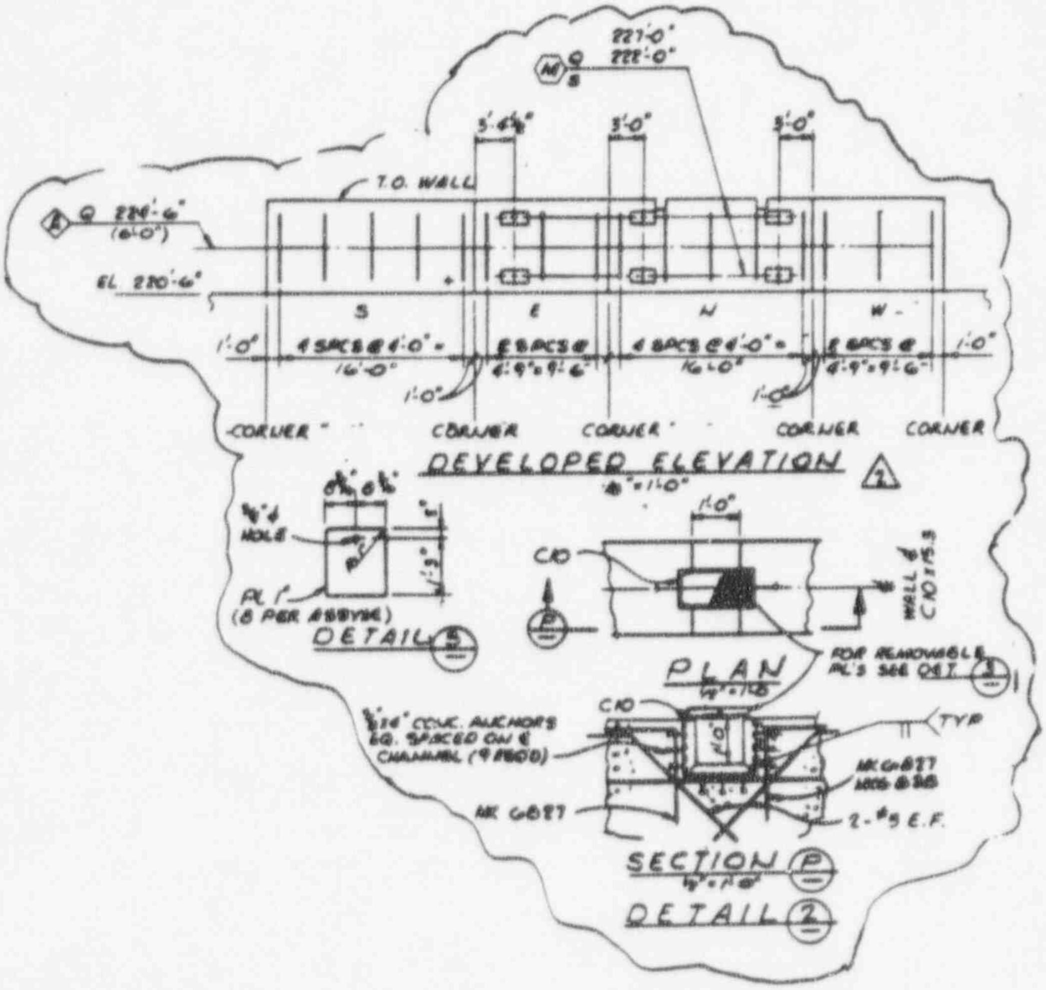
SHEET 20 FOLLOWS

SUBJECT ALTERNATE RADWASTE BLDG BASEMAT SHEET NO. 19C

REV	ORIGINATOR	DATE	CHECKER	DATE	REV	ORIGINATOR	DATE	CHECKER	DATE
2	P. Chang	3/16/88	Jim Jehrand	5-10-88					

REV. INDI-

DCR 88-V''0061 REV.0



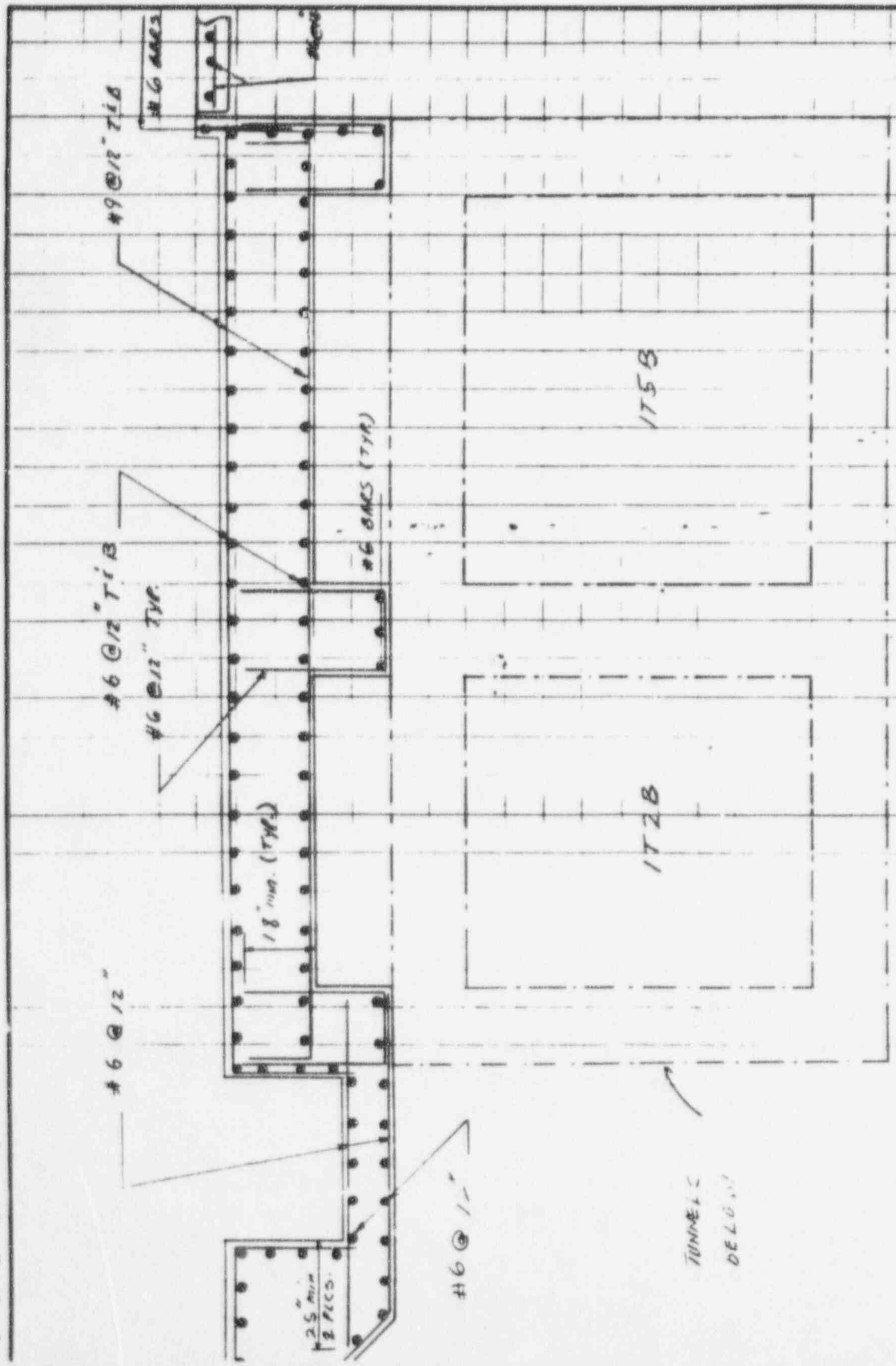
CALCULATION SHEET

CALC. NO. X2CD14.09-

SIGNATURE Amir H. Tabba DATE 5-20-86 CHECKED LOP DATE 5-27-86

PROJECT VNP JOB NO. 9510-008

SUBJECT ALTERNATE RADWASTE BLDG. BASEMAT SHEET 20 OF 23 SHEETS



SEC. A-A

NOTES:

1) MIN. LAP SPLICE FOR #6 BARS TO BE 31" MIN. FOR BOT. & 48" MIN. FOR TOP BARS.
2) " " " #9 " " " 65" " " & 90" " " "
3) ALL REBAR @ 12" UNO.
4) MAXIMUM MUDMAT THICKNESS TO BE PLACED UNDER THE BASEMAT SHALL NOT EXCEED 4" MAX.

LAO 0513 6-7

CALCULATION SHEET

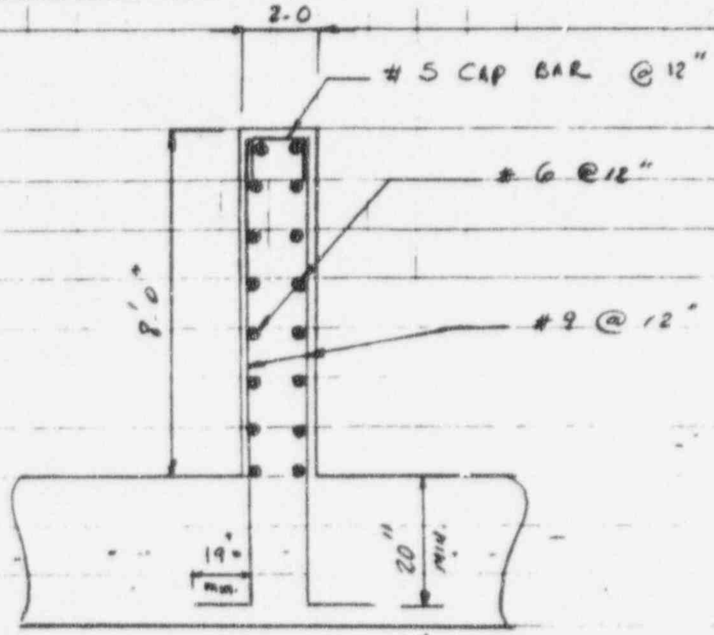
CALC. NO. X2CD14.093

SIGNATURE Amir H. Tahbaz DATE 5-13-86 CHECKED LQP DATE 5-27-86

PROJECT VNP JOB NO. 9510-008

SUBJECT ALTERNATE RADWASTE BLDG BASEMAT SHEET 21 OF 23 SHEETS

SECTIONS & DETAILS CONT'D



TYP. SECTION THRU
DEMIN. ROOM

CALCULATION SHEET

LAO 0513 8

CALC. NO. X2CD14.093

SIGNATURE Amir H Tanbaz DATE 5-20-86 CHECKED [illegible] DATE 5-27-86

PROJECT VNP JOB NO. 9510-008

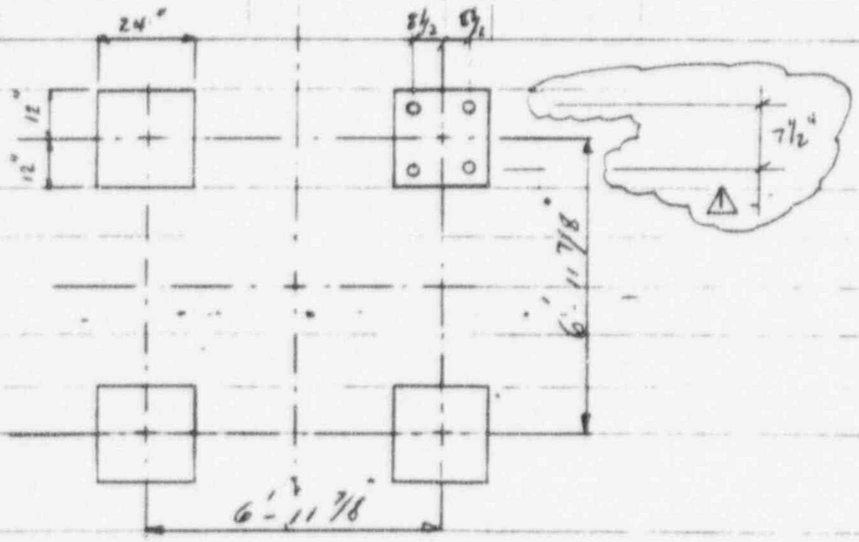
SUBJECT ALTERNATE RADWASTE BUILDING BASEMENT SHEET 22 OF 23 SHEETS

Sht 22a follows

DESIGN PEDESTAL CRANE ANCHOR BOLTS:

MAXIMUM LIFT IS 23000 lbs @ 12' RADIUS (REF NUCLEAR OPERATIONS GROUP)

MAXIMUM O.T = $23.0^{K} \times 12 = 276'K$



FOOT PRINT OF CRANE BASE

THE WORST CASE LIFT IS A LIFT WITH THE BOOM ALONG DIAGONAL.

ASSUME 4 BOLTS/ PLATE.

USE 20% IMPACT FACTOR ==> $276 \times 1.2 = 331$ 'K

$$\text{DIAGONAL} = \sqrt{7^2 + 7^2} = 9.9'$$

$$T = 331/9.9 = 33.4 = 8.36 \text{ KIPS / BOLT}$$

LAO 0513 8-73

CALCULATION SHEET

CALC. NO. X2CD14.09

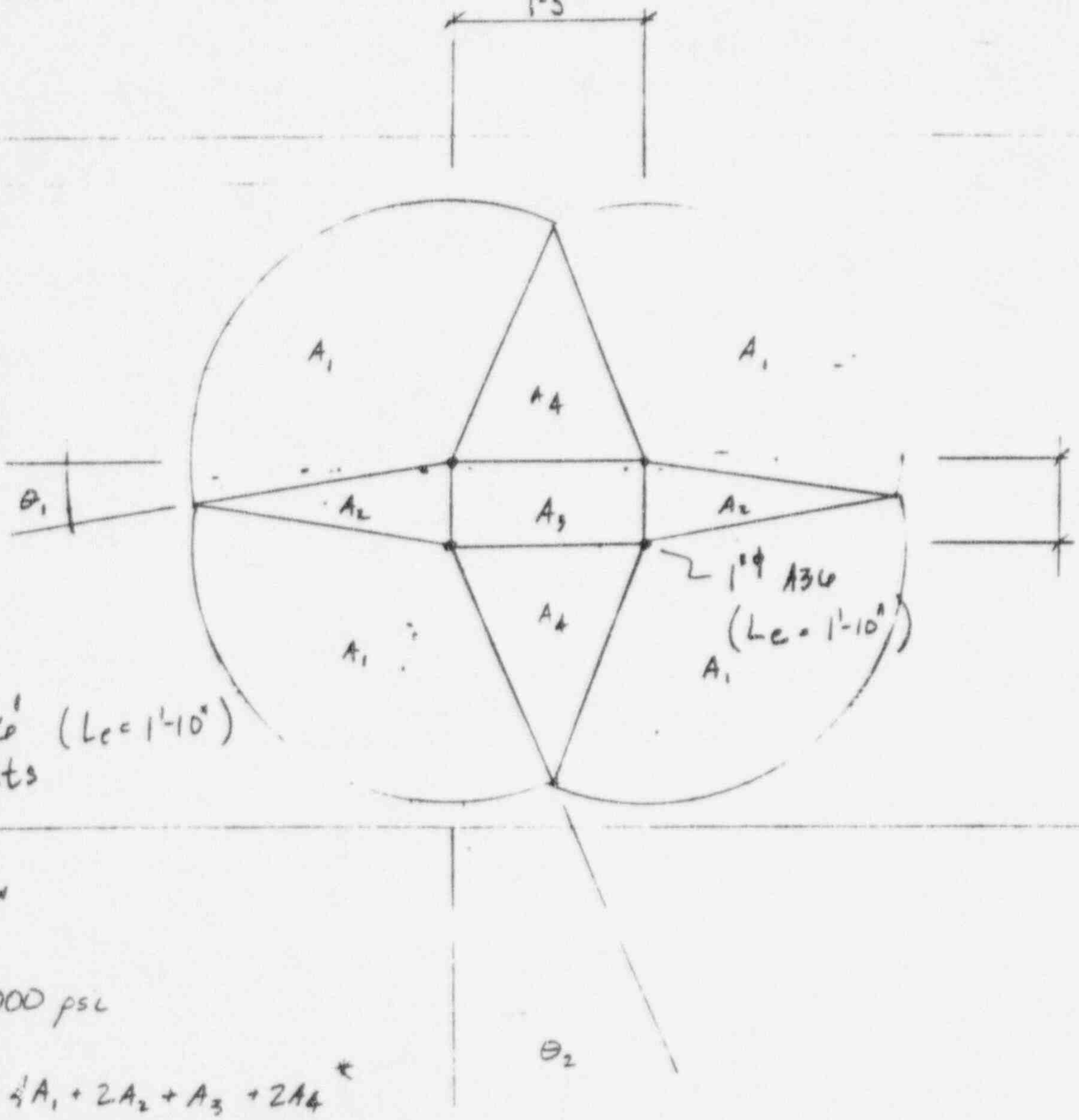
SIGNATURE RWielick DATE 8/5/86 CHECKED NSP DATE 2/5/35

PROJECT VEGP JOB NO. 9510-001

SUBJECT Alternate Radwaste Building Basemat SHEET 22a OF 23 SHEETS

Sht 22b follows

Anchor Bolts for Pedestal Crane :



1½"φ x 2'-6' (Le = 1'-10")
A36 Bolts

$r = 22"$

$f'_c = 4000$ psi

$A_{eff} = 4A_1 + 2A_2 + A_3 + 2A_4$ *

$\theta_1 = \arcsin \frac{1/2(7\frac{1}{2})}{22} = 9.81°$

$\theta_2 = \arcsin \frac{1/2(17)}{22} =$ [illegible]

*Ref Design Guide C2.34)

CALCULATION SHEET

BECHTEL

LAO 8612 6-73

CALC. NO. V2CD14.09

SIGNATURE: R Wielick DATE: 8/5/86 CHECKED: NSP DATE: 8/5/83

PROJECT: VEGP JOB NO. 9510-001

SUBJECT: Alternate Radwaste Building Basemat SHEET 22b OF 23 SHEETS
Sht. 23 follows

$A_1 = \dfrac{3.14(22)^2(90^\circ + 9.81^\circ + 22.73^\circ)}{360} = 517.31 \text{ in}^2$

$A_2 = \tfrac{1}{2}(7\tfrac{1}{2})(22)(\cos 9.81) = 81.29 \text{ in}^2$

$A_3 = 17\ (7\tfrac{1}{2}) = 127.50 \text{ in}^2$

$A_4 = \tfrac{1}{2}(17)(22)(\cos 22.73^\circ) = 172.48 \text{ in}^2$

$A_{tot} = A_{eff} = 4(517.31) + 2(81.29) + 127.50 + 2(172.48) = 2704.3 \text{ in}^2$

Pullout capacity $P_D = 4(\phi)(\sqrt{4000})(2704.3) = 444{,}691$ # $(\phi = .65)$
$= 444.7^k$

To assure ductile design

$n f_{ut} A_t \le P_D$ $(f_{ut} = 58 \text{ ksi},\ n = 4,\ A_t = 1.41 \text{ in}$

$\therefore 4(58)(1.41) = 327.1^k < 444.7^k$ $\therefore$ Ductile design is assured

Allowable $= .9 F_y A_t = .9(36)(1.41) = 45.68^k$ /Bolt

Req'd capacity $= 331^{k\text{-}ft}/9.9' = 33.4^k$/4 Bolts

$= 8.35^k$/Bolt $< 45.7^k$/Bolt (Pullout)

Therefore, Pullout OK

Shear is negligible $\therefore$ OK

CALCULATION SHEET

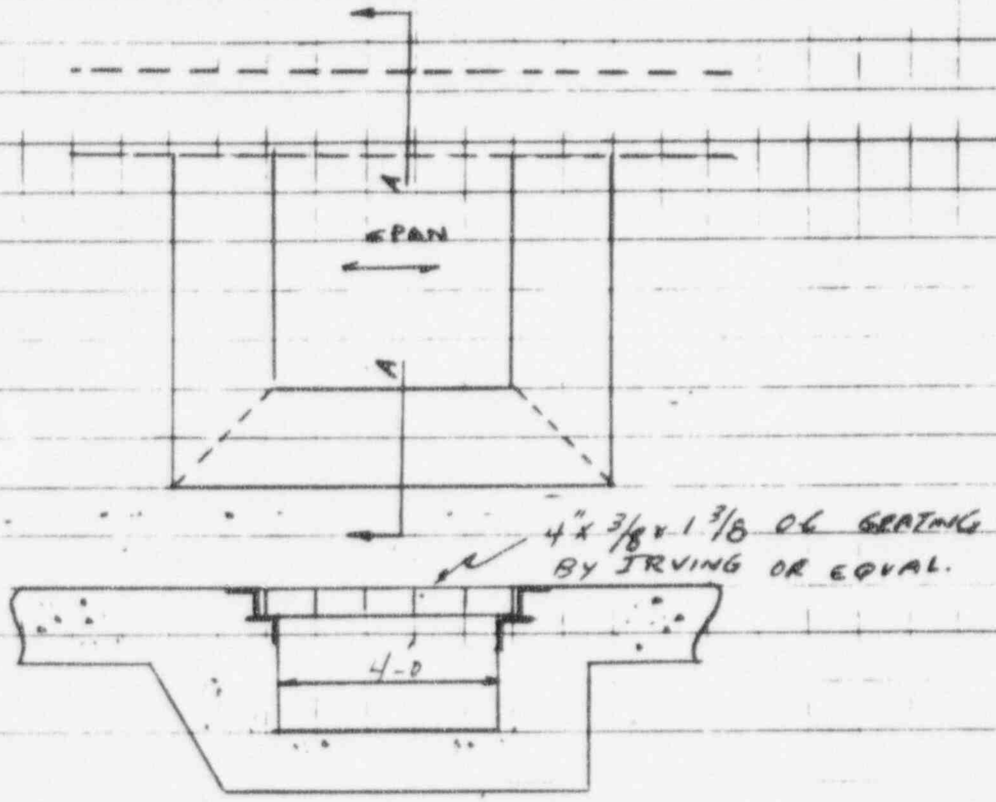
CALC. NO. X2CD14.09:

SIGNATURE Amir H Tabbaz DATE 5-22-86 CHECKED [illegible] DATE 5-27-86

PROJECT VNP JOB NO. 9510-008

SUBJECT ALTERNATE RADWASTE BLDG BASEMAT

DESIGN SUMP GRATING FOR TRUCK LOADING:



SEC. A-A

TRUCK + TRAILER + CASK = 97 KIPS (REF. 8)

ASSUME TRACTOR WEIGHT TO BE 10^K

TRAILER + CASK = $97^K - 10^K = 87^K$

TRUCK HAS 16 WHEELS (4 AXLES)

LOAD / AXLE = 21.75^K

LOAD / 2 WHEEL = 10.87^K

USE 30% IMPACT FACTOR PER MANUFACTURER'S CATALOG RECOMMENDATION. (IRVING GRATING PG 25).

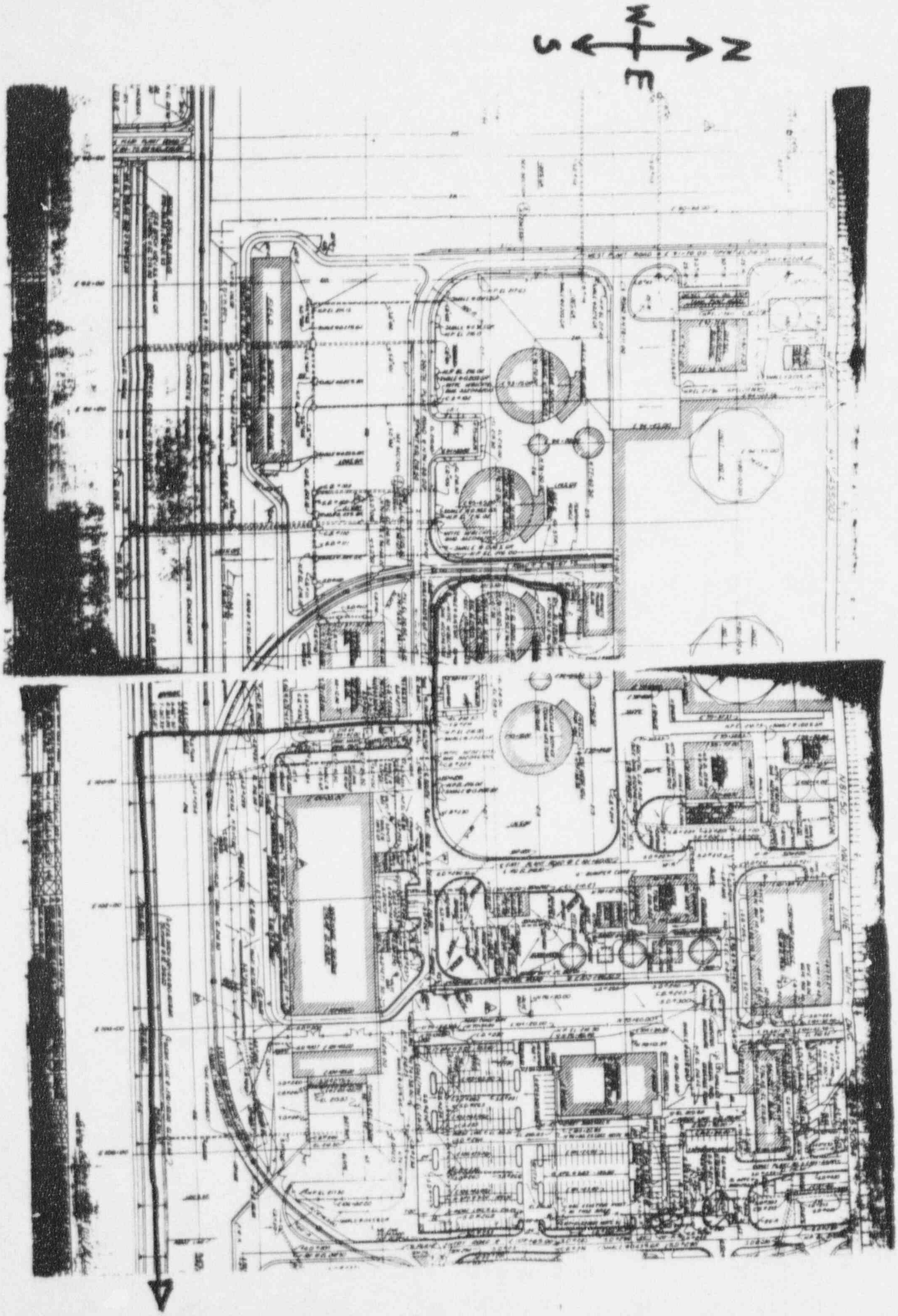
FACTORED LOAD = 10.87 x 1.3 = 14.1^K

ASSUME THIS LOAD TO ACT AS A CONCENTRATED LOAD (CONSERVATIVE)

FROM PG. 28 OF IRVING CATALOG $S = \frac{0.25 C l}{18{,}000}$ S = SEC. MODULUS IN

$S = \frac{0.25(14100)(48)}{18000} = 9.42\ IN^3$

[illegible] O.K



ALTERNATE RADWASTE BUILDING LIQUID RADWASTE SYSTEM FAILURE OFFSITE DOSE ANALYSIS

SCS calculation X4C1901S53 was performed under REA VG-9057 on 3-4-90 to calculate a groundwater and airborne release from a failure in the ARB.It concluded that the release was only about 33% that previously calculated in the FSAR for a failure of the Recycle Holdup Tank.This calculation did not evaluate a pathway by which liquid leaves the ARB as a surface water release.

On 6-27-90 an additional calculation was performed to evaluate the surface water pathway.It wrongly concludes that Regulatory limits are met.Several major errors are made in the calculation as follows:

> The calculation assumes that the radioactive surface water flows west and north as this area is used to compute the drainage area for rainfall dilution.In fact a simple review of storm drain drawings AX2D45S001 and AX2D45S002 shows that the water flows east then southeast.
>
> The calculation assumes that the water is discharged into the Savanna river at the plant discharge point as it assumes "near field dilution" factors of 10 and references the dilution effects of cooling tower blowdown and dilution flow.In fact the radioactive water would leave Georgia Power property in the small stream that flows under the low spot on New River Road (Rt 56 spur) just before the intersection with the Visitor access road near the Training Center (see drawing CX2D45V002).

As a result of these fundamental errors the calculations performed are void.As previously addressed part 21 definitions of" significant safety hazzard" include radioactivity concentrations in unrestricted areas in excess of 500 times the part 20 limits.Previously it was not stated that yearly average part 20 limits would be violated but considering the actual pathway leaving GPC property, near field dilution credit is inappropriate and yearly averages may well be exceeded.

NRC REACTIVE INSPECTION
8-06-90

NRC:

VanDenburgh, Chris A.	-	Team Leader
Wilcox, John D.	-	Assistant Team Leader
Hunemuller, Neal	-	NRR Operator License Branch (sim. cert.)
Branch, Morris	-	Senior Resident Watts Bar
Thomas, McKenzie		Region II Inspector (Q.A.)
Taylor, Pete	-	Region II Inspector (Reactor Safety)
Garner, Larry	-	Senior Resident Robinson
Brockman, Ken	-	Section Chief Reactor Projects
Matthews, Dave	-	Project Director
Reyes, Luis	-	Director Division Reactor Projects
Robinson, Sheila	-	Secretary

CONTACTS

OPERATIONS	-	Swartzwelder, Jim	Extension 3618 Beeper 044
MAINTENANCE	-	Handfinger, Harvey	Extension 4278 Beeper 310
H.P./CHEMISTRY	-	Seepe, Mike	Extension 3380 Beeper 252
Q.A.	-	Frederick, George	Extension 3228 Beeper 170
TECHNICAL SUPPORT	-	Williams, Gus	Extension 4279 Beeper 019
		Aufdenkampe, John	Extension 3600 Beeper 101
SECURITY	-	Dannemiller, Ted	Extension 3637 Beeper 444
ADMINISTRATION	-	Quick, Brent	Extension 3114 Beeper
ENGINEERING SUPPORT	-	Ealick, John	Extension 3545 Beeper 102
		Horton, Mike	Extension 3121 Beeper 107
TRAINING	-	Brown, Bob	Extension 3923
OUTAGES AND PLANNING	-	Beasley, Barnie	Extension 4209 Beeper 194

A1123
~~A1~~152

Meeting Attendance Record

Georgia Power

Meeting Purpose	File
NRC OPERATIONAL ASSESSMENT ENTRANCE	

Date	Conducted By
August 6, 1990	

Name (Print)	Title (Print)	~~Employee~~ PHONE Number	Department/Company
HERBERT BEACHER	SR. PLANT ENGINEER	3769	TECH. SUPPORT / GPC
KEN E. BROCKMAN	CHIEF, RPS 3B		DIV. Rx PROJECTS / NRC
LUIS REYES	DIVISION DIRECTOR		DIV. of Rx. PROJECTS
McKenzie Thomas	Reactor Inspector		NRC/Region II
RON AIELLO	RES INSP	4249	NRC / R II
Pete Taylor	REACTOR INSP.		NRC / R II
Morris Brand	SRI (Watts Bar		NRC / R II
ROBERT E. CARROLL III	PROJECT ENG		NRC / R II
NEAL K. HUNEMULLER	REACTOR ENGINEER		NRC / NRR
John Williams	Plant Engr. Supervisor	4279	Tech Support / GPC
JB Beasley	Mgr. Outages & Planning	4209	Outage & Planning
MC Sepp	RADWASTE SUPERVISOR	4779	HP / CHEM
G. R. FREDERICK	SUPR - SAER	x3228	GPC
E.M. DANNEMILLER II	NUCLEAR SECURITY MGR	3637	SECURITY / GPC
CURTIS STINESPRING	MGR. PLANT ADMIN	3113	GPC
J.E. SWARTZWELDER	OPS. MGR.	3618	OPS / GPC
F.J. EALICK	ENERG SUPPORT SUPV	3545	GPC
G BOCKHOLD	GM	3118	GPC
Tom Green	ADM	3[illegible]	GPC
LEE GLENN	MGR, CORPORATE CONCERNS	3294 8-526-1465	CORP CONCERNS / GPC (ATL)
W. C. LYON	Quality Concerns Coord.	3294 [illegible] #178	Quality Concerns
K.D. STARKEY	RESIDENT INSPECTOR	4249	NRC
B.R. BONSER	SR RES. INS.	4249	NRC
[illegible]	SR RES INSP - ABR		NRC
J. D. Wilcox Jr.	NRC - NRR		NRC

A/154
A/124

Please

Meeting Attendance Record

Georgia Power

Meeting Purpose	File
NRC OPERATIONAL ASSESSMENT ENTRANCE	

Date	Conducted By
August 6, 1990	

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August 7, 1990

the southern electric system

W. G. Hairston, III
Senior Vice President
Nuclear Operations

ELV-01995
0535

Docket No. 50-424
50-425

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D. C. 20555

Gentlemen:

VOGTLE ELECTRIC GENERATING PLANT
SPECIAL REPORT
VALID DIESEL GENERATOR FAILURES

In accordance with the requirements of the Vogtle Electric Generating Plant Technical Specifications, Sections 4.8.1.1.3 and 6.8.2, Georgia Power Company hereby submits the enclosed Special Report concerning three valid diesel generator failures.

Sincerely,

W. G. Hairston, III

W. G. Hairston, III

WGH,III/NJS/gm

Enclosure: Special Report 1-90-05

xc: Georgia Power Company
Mr. C. K. McCoy
Mr. G. Bockhold, Jr.
Mr. P. D. Rushton
Mr. R. M. Odom
NORMS

U. S. Nuclear Regulatory Commission
Mr. S. D. Ebneter, Regional Administrator
Mr. T. A. Reed, Licensing Project Manager, NRR
Mr. B. R. Bonser, Senior Resident Inspector, Vogtle

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ENCLOSURE

VOGTLE ELECTRIC GENERATING PLANT - UNITS 1 AND 2
TECHNICAL SPECIFICATION SPECIAL REPORT 1-90-05
VALID DIESEL GENERATOR FAILURES

A. REQUIREMENT FOR REPORT

This report is required in accordance with Technical Specification (TS) 4.8.1.1.3. This specification requires that all diesel generator (DG) failures, valid or non-valid, be reported to the Commission in a Special Report pursuant to TS 6.8.2.

B. DESCRIPTION OF EVENT

On 7-11-90, Diesel Generator (DG) 2A was being tested during a routine surveillance per procedure 14980-2, "Diesel Generator Operability Test". The right air start bank was isolated to allow testing of the left air start bank. The engine start button was pushed by the control room operator and the engine began to roll with starting air. According to the local operator in the diesel room, the engine rolled twice and stopped. The DG was declared inoperable and the TS action statement was initiated. The DG was unavailable for emergency operation for a period of 67 hours and 49 minutes.

During the review of this event, it was determined that similar events had occurred on 4-12-90 and 7-5-90. These previous similar events had not been recognized as failures and therefore had not been reported as such. These events are described as follows:

On 4-12-90, operators conducted a TS surveillance test of DG 2A per procedure 14980-2. The manual start button was pushed, but no start occurred. Operators decided that the pushbutton had not been depressed long enough and made another attempt which resulted in a successful start. On 7-5-90, a similar incident occurred on DG 1B, and a successful start again resulted on the second attempt. Neither DG was considered to be unavailable for emergency operation as a result of these two events.

C. CAUSE OF EVENT

An investigation into the 7-11-90 event by utility and vendor personnel found that the starting air valve pistons could stick in their cap assemblies due to inadequate manufacturing tolerances. This condition was apparently the result of the initial manufacturing process which left insufficient clearances between some of the pistons and caps. A failure to start would occur only after the engine had been shut down from a previous run and the engine stopped with a particular alignment of faulty air start valves and crankshaft position.

ENCLOSURE

VOGTLE ELECTRIC GENERATING PLANT - UNITS 1 AND 2
TECHNICAL SPECIFICATION SPECIAL REPORT 1-90-05
VALID DIESEL GENERATOR FAILURES

On a non emergency manual start with the air start pilot valves malfunctioning, the initial burst of air was not adequate to start the engine. The burst of air was adequate to change the alignment of the crankshaft with respect to the faulty air start pilot valves so that any subsequent attempt to start the engine could be successful. This problem is now believed to have been the cause of the DG failures on 1-24-90 and 1-25-90, which were reported to the Commission on 2-19-90 as Special Report 2-90-02. On 7-19-90, the manufacturer of the valves submitted a 10 CFR 21 report to the Commission as a result of the above findings.

The failure of the DG operators to recognize the initial start attempts of 4-12-90 and 7-5-90 as DG failures is partially attributed to limitations of the simulator computer. The simulator requires operators to hold the DG manual start pushbutton in order to have the proper control signals annunciate, creating the misconception that the pushbutton must remain depressed for a given period of time in order for a DG start to occur.

D. CORRECTIVE ACTIONS

1. The sixteen starting air valves on each of the four DG's were tested and polished where necessary to provide adequate clearance between the pistons and caps.

2. The appropriate maintenance procedures will be revised by the next refueling outages to require testing of the starting air valves to demonstrate freedom of movement following DG overhaul.

3. During shift briefings, operators were advised that the DG should start when the manual pushbutton is depressed, any failure to manually start is a reportable event, and such information should be relayed to the appropriate personnel so that a report can be initiated.

4. Operator training will be enhanced during the next training cycle to advise personnel that a DG start should occur without having to continue depressing the manual start pushbutton.

5. The DG 1B and 2A test frequency is currently once per 7 days in accordance with TS Table 4.8-1. This frequency will be continued until 7 consecutive valid tests are completed with no more than one valid failure in the last 20 valid tests and/or no more than 4 valid failures in the last 100 valid tests. Up to and including the 7-5-90 valid failure, there have been a total of 6 valid failures in 79 valid tests of DG1B. Up to and including the 7-11-90 valid failure, there have been a total of 5 valid failures in 43 valid tests of DG2A.

Interoffice Correspondence

Georgia Power

August 3, 1990

Memo To: George Bockhold, Jr.
General Manager Nuclear Plant-Vogtle

Subject: Vogtle Electric Generating Plant - Units 1 & 2
QA Audit of Surveillance Program/Technical Specification Compliance
OP09-90/31

File: X7BG17-P-OP09

Log No: VSAER-90-186

Audit Scope: The purpose of this audit was to verify compliance with, and the effectiveness of, the Plant Vogtle Quality Assurance Program as applied to Surveillance Program/Technical Specification Compliance. The scope of this audit included a review of Technical Specification surveillances covering power distribution limits, electrical power systems, limiting safety system settings, reactivity control systems, and instrumentation.

Summary of Problems Found:

- Procedure 88075-C inadequately implemented Technical Specification surveillance 4.2.5.3 requirements that instrumentation used for the precision heat balance calculation must be calibrated within seven days prior to the surveillance. Therefore, the Unit 1 surveillance was improperly performed prior to operation above 75% rated thermal power following the Unit 1 second refueling outage. (AFR #432)

- Some corrective actions for the event of March 20, 1990, in which loss of offsite power lead to a site area emergency, were not implemented in an effective and timely manner. (AFR #433)

Evaluation: With the exception of the Unit 1 precision heat balance calculation, surveillance tasks reviewed had been performed within the specified time periods, had received required reviews and approvals, and had met appropriate acceptance criteria. The Technical Specification surveillance which was improperly performed was caused by procedural inadequacy. However, the results of this audit, based on sample size, do not indicate a programmatic weakness concerning the technical adequacy of implementing procedures for surveillances.

During this audit, indication of inadequate corrective action was also identified. The diesel generator 1B (7/5/90) and 2A (4/12/90) failures are examples that effective and timely corrective actions from the event of March 20, 1990 were not taken. These examples indicate that additional management attention may be required and that other corrective actions from the event reviewed for adequacy of implementation.

Action: In accordance with ANSI N45.2.12, you are requested to respond to the attached audit report no later than September 2, 1990.

G. R. Frederick

G. R. Frederick
Supervisor - SAER

CMB/GRF/btp

Attachment

xc: R. P. McDonald
W. G. Hairston, III
C. K. McCoy
O. M. Fraser
H. M. Handfinger
W. F. Kitchens
M. J. Ajluni
W. E. Mundy
J. G. Aufdenkampe, Jr. (Org. AFR's #432 and #433)
J. W. McGowan
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Q. A. File
J. E. Swartzwelder
M. W. Horton
T. V. Greene
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NORMS

PLANT

Plant Vogtle - Units 1 & 2

ACTIVITY

QA Audit of Surveillance Program/Technical Specification Compliance

AUDIT NO.

OP09-90/31

DATES AUDITED

June 28 through July 12, 1990

AUDITOR

C. M. Burke, Senior QA Field Representative (Audit Team Leader)

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M. M. Handfinger	x		x
J. C. Williams	x		
J. E. Swartzwelder	x		x
R. L. LeGrand	x		
D. R. Lee	x		
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M. E. McGrath		x	
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M. C. Henry		x	
T. L. Morris		x	
R. K. Pope		x	
G. A. Ovellette		x	
D. N. Sowsawat		x	

R. M. Odom	x
A. Rickman	x
R. P. Farrow (Wisco)	x
R. M. Smith (Wisco)	x
T. L. Willis	x
J. J. Godbee	x
J. Redding	x

REFERENCES

Vogtle Electric Generating Plant (VEGP) Units 1 and 2 Technical Specifications

PROCEDURE	REVISION	DESCRIPTION
11885-C	13	Diesel Generator Operating Log
13145-1	22	Diesel Generators
14230-1	4	AC Source Verification
14925-1	3	Power Range Reactor Trip Interlocks 18 Month ACOT [Analog Channel Operational Test]
14940-1	9	Estimated Critical Condition Calculation
14940-2	3	Estimated Critical Condition Calculation
14980-1	20	Diesel Generator Operability Test
24525-1	7	Pressurizer Pressure Protection Channel I 1P-455 Analog Channel Operational Test and Channel Calibration
24553-1	3	Turbine Trip - Reactor Trip Hydraulic Pressure 1P-6161 Analog Channel Operational Test and Channel Calibration
24700-1	22	Nuclear Instrumentation System Power Range Channel 1N41 Channel Calibration
24782-1	9	Reactor Coolant Flow Loop 1 Protection Channel I 1F-414 Analog Channel Operational Test and Channel Calibration
24783-1	8	Reactor Coolant Flow Loop 2 Protection Channel I 1F-424 Analog Channel Operational Test and Channel Calibration
24784-1	8	Reactor Coolant Flow Loop 3 Protection Channel I 1F-434 Analog Channel Operational Test and Channel Calibration
24810-1	13	Delta T/T AVG Loop 1 Protection Channel I 1T-411 Analog Channel Operational Test and Channel Calibration
88003-C	0	Shutdown Margin by Minimum Bank Height
88013-C	1	Overall Core Reactivity Balance
~~88014-C~~	~~0~~	~~Reactor Coolant System Flow Measurement~~
88023-C	3	One Point Incore/Excore Detector Calibration
88025-2	0	Determination of Movable Incore Detector Operating Voltages
88075-C	1	Precision Heat Balance

PURPOSE/SCOPE

The purpose of this audit was to verify compliance with, and the effectiveness of, the Plant Vogtle Quality Assurance Program as applied to Surveillance ~~Program/Technical~~ Specification Compliance. The scope of this audit ~~includes~~ a review of power distribution limits, electrical power systems, limiting safety system settings, reactivity control systems, and instrumentation.

EVALUATION

Observation of Technical Specification surveillances being performed noted compliance with procedures and that personnel were knowledgeable of the surveillance requirements. With the exception of the Unit 1 precision heat balance calculation, surveillance tasks reviewed had been performed within the specified time periods, had received required reviews and approvals, and had met appropriate acceptance criteria. The Technical Specification surveillance which was improperly performed was caused by procedural inadequacy. However, the results of this audit, based on sample size, do not indicate a programmatic weakness concerning the technical adequacy of implementing procedures for surveillances. The diesel generator 1B and 2A failures are examples of inadequate corrective actions from the event of March 20, 1990. These examples indicated that attention to detail is required in the area of corrective actions and that other corrective actions from the event should be reviewed for adequacy and implementation.

AUDIT DETAILS

I Power Distribution Limits

A. Requirement

VEGP Units 1 and 2 Technical Specifications, section 4.2.5.3 states that after each fuel loading, the Reactor Coolant System (RCS) flow rate shall be determined by precision heat balance prior to operation above 75% rated thermal power (RTP). The RCS flow rate shall also be determined by precision heat balance at least once per 18 months. Within seven days prior to performing the precision heat balance flow measurement, the instrumentation used for performing the precision heat balance shall be calibrated.

B. Results

This surveillance requirement is satisfied by performance of procedures 88014-C and 88075-C. Review of completed documentation for tasks 88014-102, 88014-103, and 88075-101 noted that the surveillances had been performed following the Unit 1 second refueling outage. However, review of procedure 88075-C noted that the procedure addressed calibration of special test instrumentation but did not address plant instrumentation. Discussions with the Reactor Engineering Supervisor and review of completed procedure 88075-C noted that plant instruments 1-TE-15200, 1-TE-15201, 1-TE-15202, and 1-TE-15203 were used for feedwater temperature input for the precision heat balance calculation.

Discussions with the Instruments and Controls Superintendent noted that the instruments are calibrated on a six months basis and were calibrated on 1/23/90. Additional review noted that Maintenance Work Order (MWO) 1-90-02215 calibrated the instruments on 4/28/90. However, the precision heat balance calculation and the RCS flow measurement were performed on 4/23/90 at 2330 CDT. Also, 75% RTP

was achieved at approximately 0130 CDT on 4/24/90. Therefore, surveillance 4.2.5.3 was incorrectly performed because procedure 88075-C did not require calibration of plant instrumentation used for feedwater temperature input for the precision heat balance calculation. This discrepancy will be included in Audit Finding Report OP09-90/31 #432. After the post-audit conference, Deficiency Card #1-90-293 was initiated by Reactor Engineering personnel to facilitate the reportability review of this item.

C. Requirement

VEGP Units 1 and 2 Technical Specifications, section 4.2.5.2 requires the RCS flow rate indicators to be subjected to channel calibration at each fuel loading and at least once per 18 months.

D. Results

Review of completed documents for surveillance tasks 24782-101, 24783-101, 24784-101, and 24785-101 noted that the 18 months surveillances were performed with acceptable results and within the required time limits during the Unit 1 second refueling outage. No problems were noted.

II. Electrical Power Systems

A. Requirement

VEGP Units 1 & 2 Technical Specifications, section 4.8.1.1.2.a requires each diesel generator to be demonstrated operable in accordance with the frequency specified in Table 4.8-1 on a staggered test basis by verifying seven specific items.

B. Results

The auditor observed performance of surveillance procedure 14980-1 for the Unit 1 Train B diesel generator operability test. This test is normally performed monthly, but is being performed weekly as required by Table 4.8-1 due to the number of recent test failures. In the Unit 1 Train B Diesel Generator Building, the ~~auditor verified that the following procedures being used were~~ the current revision: 11885-C, 13145-1, and 14980-1. Additionally, the auditor observed performance of the cylinder moisture check, an independent verification of the cylinder moisture check, the diesel generator air start compressor test, and completion of the Diesel Generator Operating Log.

In the Unit 1 Control Room, the auditor observed loading of the diesel generator and the Diesel Generator Fuel Oil Transfer System test. Additionally, the auditor verified that the stop watches, required by step 4.2.a, were currently calibrated. Upon completion of the test, the auditor verified that the acceptance criteria of section 6.0 had been met.

During performance of the test, the diesel generator failed to start on the first attempt. Discussions with an Operations Shift Superintendent and a Shift Supervisor noted that on the first attempt a Reactor Operator trainee had not held the diesel generator start push button for sufficient time to start the diesel. The failure was attributed to operating error and was not considered to be a valid test failure as allowed by Table 1 of procedure 14980-1. Alignments were reverified by Operations personnel, and on the second attempt the diesel generator started. No problems were noted with the diesel generator operability test.

NOTE: Because of a similar problem identified with the Unit 2 Train A diesel generator on 7/11/90 (VEGP Event Report #2-90-005), this audit detail was brought to the attention of plant management during the write-up of this report on 7/17/90. Immediate corrective action was initiated by the General Manager Nuclear - Plant Vogtle and a special test (MWO 1-90-03340) was conducted on 7/18/90 for the Unit 1 Train B diesel generator. This test, witnessed by the auditor, identified that ten of sixteen air pilot valves on the cylinders failed when subjected to 100 psig air pressure.

Additional review by the auditor noted that the 7/5/90 1B diesel generator failure and associated pertinent alarms and indications were not documented in the Unit 1 Shift Supervisor Log, the Unit 1 Control Log, or in the technical specification surveillance package for task #14980-102-25185. This event should have been identified as a failure to start and should have been logged accordingly. Also, a 4/12/90 2A diesel generator failure - for the same reasons as the 1B failure - was not critiqued until after the similar failure on 7/11/90 for diesel generator 2A. These discrepancies indicate that:

- o The requirement to conduct an event critique for each diesel failure until plant management decides critiques are no longer required (commitment 18758), and
- o The requirement to revise or develop a procedure/policy outlining guidelines for logging pertinent alarms and indications to assist in evaluation of equipment or system malfunctions (commitment 18756)

were not implemented in an effective and timely manner for the event of 3/20/90, in which loss of offsite power lead to a site area emergency. Audit Finding Report OP09-90/31, #433 will be issued to track corrective actions for this item.

C. Requirement

VEGP Units 1 and 2 Technical Specifications, section 4.8.1.2 requires the following AC electrical power sources to be demonstrated operable during modes 5 and 6 by the performance of each of the requirements of Specifications 4.8.1.1.1, 4.8.1.1.2 (except for specification 4.8.1.1.2a.5), and 4.8.1.1.3:

1. One circuit between the offsite transmission network and the Onsite Class 1E Distribution System, and

2. One diesel generator with:

   o a day tank containing at least 650 gallons of fuel,

   o a fuel storage system containing at least 68,000 gallons of fuel, and

   o a fuel transfer pump.

D. Results

Specification 4.8.1.1.1 is met by the weekly performance of procedure 14230-1. Specification 4.8.1.1.3 is satisfied by reports as applicable. For Specification 4.8.1.1.2, the auditor verified performance of surveillance task 14980-103 for the Unit 1 Train A diesel generator. In Document Control, the auditor reviewed completed procedure 14230-1 data sheets for the period of 3/13/90 through 6/20/90 and noted that acceptance criteria had been met. Also, the surveillances were performed within the required time limit (weekly since 4/9/90) and had been properly approved. No problems were noted.

III. Limiting Safety System Settings

A. Requirement

VEGP Units 1 and 2 Technical Specifications, section 2.2.1 requires the Reactor Trip System Instrumentation and Interlock Setpoints to be set consistent with the trip setpoint values shown in Table 2.2-1.

B. Results

The auditor verified by calculation that the setpoint and allowable values for eleven of twenty-eight functional units (overtemperature delta T, overpower delta T, power range neutron flux, etc.) from Table 2.2-1 were accurately incorporated into Instruments and Controls (I&C) procedures 24700-1, 24525-1, and 24553-1 for channel calibration and analog channel operational tests (ACOT's). Additionally, the auditor verified that setpoint and allowable values for four functional units were properly incorporated into Operations procedure 14925-1 for the eighteen months ACOT of the power range reactor trip interlocks. No problems were noted.

IV. Reactivity Control Systems

A. Requirement

VEGP Units 1 and 2 Technical Specifications, section 4.1.1.1.1.c requires the shutdown margin to be determined to be greater than or equal to 1.3% delta k/k with K_{eff} less than 1, within 4 hours prior to achieving reactor criticality by verifying that the predicted critical control rod position is within the limits of Specification 3.1.3.6.

B. Results

The auditor determined the dates of the Units 1 and 2 reactor trips and reviewed completed 14940-1 and 14940-2 procedures in Document Control. The review noted that an estimated critical condition calculation utilizing either an estimated critical boron concentration (ECC) calculation or an estimated critical position (ECP) calculation had been performed prior to the reactors achieving criticality after the trips. No problems were noted.

C. Requirement

VEGP Units 1 and 2 Technical Specifications, section 4.1.1.1.1.d requires the shutdown margin to be determined to be greater than or equal to 1.3% delta k/k prior to initial operation above 5% rated thermal power (RTP) after each fuel loading, by consideration of six factors, with the control banks at the maximum insertion limit of Specification 3.1.3.6.

D. Results

Review of completed documents for surveillance task 88003-101 noted that the surveillance had been performed prior to operation above 5% RTP after the second refueling of Unit 1. No problems were noted.

E. Requirement

VEGP Units 1 and 2 Technical Specifications, section 4.1.1.1.2 requires the overall core reactivity balance to be compared to predicted values to demonstrate agreement within ± 1% delta k/k at least once per 31 effective full power days (EFPD). The predicted reactivity values shall be adjusted (normalized) to correspond to the actual core conditions prior to exceeding a fuel burnup of 60 EFPD after each fuel loading.

F. Results

The auditor reviewed completed documentation for surveillance tasks #88013-201 performed on Unit 2 since January 1, 1990. The review verified that the seven calculations had been performed at least once per 31 EFPD and that appropriate acceptance criteria had been met. No problems were noted.

Concerning normalization of predicted reactivity values after fuel loading, the auditor verified that procedure 88013-C had been performed satisfactorily for Unit 1 prior to 60 EFPD after the second refueling. No problems were noted.

V. Instrumentation

A. Requirement

VEGP Units 1 and 2 Technical Specifications, section 4.3.1.1 requires each Reactor Trip System instrumentation channel and interlock and the automatic trip logic to be demonstrated operable by the performance of the Reactor Trip System instrumentation surveillance requirements specified in Table 4.3-1.

B. Results

The auditor reviewed completed documentation for twelve surveillance tasks and noted that the surveillances for the applicable functional items in Table 4.3-1 had been performed within the required time limits and with acceptable results. No problems were noted.

Additionally, the auditor observed performance of surveillance task 24810-102 for the 31 days Analog Channel Operational Test surveillance requirement for Table 4.3-1 item 7 (overtemperature delta T) and item 8 (overpower delta T). Test equipment was currently calibrated and documents used were noted to be current. Quality Control hold points were noted to be honored and performed. Also, the surveillance was performed in a professional manner by all involved personnel. No problems were noted.

Also, the auditor accompanied Reactor Engineering personnel to the Unit 2 Control Room to observe performance of surveillance tasks 88023-201 and 88023-202. The auditor observed the Reactor Engineer obtain clearance 2-90-10046 and perform procedure 88025-2 for Westinghouse incore instrumentation detectors A, B, C, D, & E as a prerequisite to performance of procedure 88023-C. However, the surveillance was postponed after a Reactor Engineer noted that the reactor was not at a constant power due to a reactor trip several days earlier. For the activities observed, no problems were noted.

OPEN ITEMS

From previous audits: None were reviewed.

From this audit:

OP09-90/31
432

Plant instrumentation used for feedwater temperature input in the Unit 1 precision heat balance calculation was not calibrated within seven days prior to the surveillance as required by Technical Specification 4.2.5.3.

OP09-90/31 433	Some corrective actions for the event of March 20, 1990, in which loss of offsite power lead to a site area emergency, were not implemented in an effective and timely manner.

POST-AUDIT CONFERENCE

A post-audit conference was held on July 12, 1990. The audit results were presented, discussed, acknowledged, and agreed upon by those attending the meeting.

Charles M. Burke

C. M. Burke
Senior QA Field Representative

G. R. Frederick

G. R. Frederick
Supervisor - SAER

ORIGINAL

Audit Finding Report
Safety Audit and Engineering Review

Trend Code: P05D
Responsible Party : J. G. Aufdenkampe

AFR No. OP09-90/31 #432	☐ Reconfirmed Previous Finding No.	Tracer No. VSAER-90-186
Company/Organization Georgia Power Company QA Audit of Surveillance		
Project/Activity Plant Vogtle - Units 1 & 2, Program/Technical Specification Compliance		
Auditor(s) C. M. Burke, Senior QA Field Representative (ATL) June 28-July 12, 1990		

Signature: Charles M. Burke Date: 8-3-90

REFERENCE/REQUIREMENTS

VEGP Units 1 and 2 Technical Specification surveillance requirement 4.2.5.3, "After each fuel loading, the RCS [Reactor Coolant System] flow rate shall be determined by precision heat balance prior to operation above 75% RATED THERMAL POWER. The RCS flow rate shall also be determined by precision heat balance at least once per 18 months. Within 7 days prior to performing the precision heat balance flow measurement, the instrumentation used for performing the precision heat balance shall be calibrated. The provisions of 4.0.4 are not applicable for performing the precision heat balance flow measurement."

FINDING

Review of completed Technical Specification surveillance procedure 88075-C, "Precision Heat Balance," and discussions with plant personnel noted that plant instruments 1-TE-15200, 1-TE-15201, 1-TE-15202, and 1-TE-15203 used for feedwater temperature input for the Unit 1 precision heat balance were not calibrated within seven days prior to performance of the surveillance. Therefore, procedure 88075-C inadequately implements Technical Specification 4.2.5.3 requirements and the surveillance was improperly performed.

RECOMMENDED ITEMS TO BE CONSIDERED IN CORRECTIVE ACTION

1. Investigate the problem and any potential similar conditions.
2. State actions taken to resolve the specific and similar problems.
3. Determine root cause.
4. State actions taken to prevent recurrence.

CORRECTIVE ACTION/REVIEW VERIFICATION

Signature: Date:

ORIGINAL

Audit Finding Report
Safety Audit and Engineering Review

Responsible Party : G. Bockhold, Jr.

Trend Code: P16B

FR No.
OP09-90/31 #433

☐ Reconfirmed Previous Finding No.

Tracer No.
VSAER-90-186

Company/Organization
Georgia Power Company QA Audit of Surveillance

Project/Activity
Plant Vogtle - Units 1 & 2, Program/Technical Specification Compliance

Auditor(s)
C. M. Burke, Senior QA Field Representative (ATL) June 28-July 12, 1990

Signature: Charles M. Burke Date: 8-3-90

REFERENCE/REQUIREMENTS

1. Title 10 Code of Federal Regulations (CFR) 50, Appendix B, Criterion XVI/"Measures shall be established to assure that conditions adverse to quality, such as failures, malfunctions, deficiencies, deviations, defective material and equipment, and nonconformances are promptly identified and corrected. In the case of significant conditions adverse to quality, the measures shall assure that the cause of the condition is determined and corrective action taken to preclude repetition."

2. Open Item Tracking System #18756 indicated that a procedure/policy would be revised or developed outlining guidelines for logging pertinent alarms and indications to assist in evaluation of equipment or system malfunctions. It also indciated that this data would be emphasized in training and by appropriate management attention.

3. Open Item Tracking System #18758 indicated that the trend program would be reviewed for adequacy of coverage of diesel generator component failures and that event critiques would be required for diesel generator failures.

FINDING

Effective and timely corrective actions for the event of 3/20/90, in which loss of offsite power lead to a site area emergency, were not adequately implemented. The following are examples:

o Finding #1 (corresponds to requirements #1 & #2)
On 7/5/90, the Unit 1 Train B diesel failed to start. However, this failure was not documented in the Unit 1 Shift Supervisor Log, the Unit 1 Control Log, or in the associated Technical Specification surveillance package for surveillance task #14980-102-25185.

o Finding #2 (corresponds to requirements #1 & #3)
The 4/12/90 2A Diesel Generator failure was not critiqued until after the 7/11/90 failure of diesel generator 2A. The 1B diesel generator failure of 7/5/90 was not critiqued until after management was made aware of the failure on 7/17/90.

RECOMMENDED ITEMS TO BE CONSIDERED IN CORRECTIVE ACTION

1. Investigate the problem and any potential similar conditions.
2. State actions taken to resolve the specific and similar problems.
3. Determine root cause.
4. State actions taken to prevent recurrence.

AFRRO

ORIGINAL

Audit Finding Report Continuation

AFR No. OP09-90/31 #433 VSAER-90-186

CORRECTIVE ACTION/REVIEW VERIFICATION

Signature: Date:

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W. G. Hairston, III
Senior Vice President
Nuclear Operations

July 9, 1990

the southern electric system

ELV-01881
0233

Docket Nos. 50-424
50-425

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

ATTN: James M. Taylor
Executive Director for Operations

Gentlemen:

VOGTLE ELECTRIC GENERATING PLANT
COMMENTS ON NUREG-1410

Your letter of June 8, 1990 transmitted a copy of the NRC Incident Investigating Team (IIT) report (NUREG-1410) and stated that any comments should be provided by July 9, 1990. We believe that the IIT performed a thorough investigation of the event and that the report in general represents an accurate description of the circumstances surrounding the Site Area Emergency on March 20, 1990. We have reviewed the document and specific comments are attached.

Sincerely,

W. G. Hairston III

W. G. Hairston, III

WGH,III/RWM/gm

Attachment

xc: Georgia Power Company
Mr. C. K. McCoy
Mr. G. Bockhold, Jr.
Mr. R. M. Odom
Mr. P. D. Rushton
NORMS

U. S. Nuclear Regulatory Commission
Mr. S. D. Ebneter, Regional Administrator
Mr. T. A. Reed, Licensing Project Manager, NRR
Mr. B. R. Bonser, Senior Resident Inspector, Vogtle

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ENCLOSURE

COMMENTS ON NUREG-1410

1. P. 1-5 states that the risk of core damage is greater from mid-loop operation than due to a fuel handling accident; no basis to support this assumption is provided.

2. P. 2-3 states erroneously that "special containment building penetrations had been constructed in anticipation of the need to quickly close the containment building during an outage." They were in fact constructed to permit SG sludge lancing with the equipment hatch closed.

3. P. 3-11 (3.2.5) needs to recognize that an emergency start of the diesel generators at VEGP has been changed to include starting on loss of offsite power.

4. P. 5-22 (5.5.2.2) states that maintenance personnel should be trained in mid-loop operations. This has not been determined to be appropriate for required maintenance training through systematic analysis of job tasks.

5. P. 7-5 and Appendix K (p. k-8) (Also, on page 7-7 for the first refueling outage) list several "nonconservative conditions" that existed in the plant, with the implication that, in retrospect, these conditions should not have existed. GPC contends that sufficient risk analysis does not exist to show that the list is non-conservative.

6. P. 10-3 (Conclusion 10.4.1) and J-28 state that foreign material in the DG jacket water temperature switches is considered the most likely cause of the DG trips during the event. GPC concurs that internal contamination is the most likely cause of one switch tripping. The second switch, although it had internal foreign material, was also set low as a result of inadequate calibration procedures. Subsequent additional testing has shown that these switches are temperature sensitive, requiring that consistent calibration techniques be used to achieve the desired setting (reference LER 50-424/1990-006, revision 1). Therefore, GPC has concluded that the primary cause of the intermittent actuation of the jacket water temperature switches was inadequate calibration in conjunction with foreign material.

7. P. 10-8 (Conclusion 10.7.3) states that Vogtle had a high number of failures of DG sensors compared to the rest of the industry. From Appendix I (p. I-8) of the report which itemizes these failures, a large percentage of the problems were calibration setpoint out of specification during construction acceptance testing. Out of calibration in this application is not typically counted as a failure by either Vogtle or other plants in accordance with NPRDS reporting criteria; hence the NRC conclusion that Vogtle has a higher number of switch failures is not based on comparable data.

8. P. K-2, states erroneously that future SG eddy current testing will be conducted on only 20 percent of the tubes. VEGP currently plans to follow the EPRI guidelines which require a minimum of 20 percent of the tubes to be tested as a base scope, plus an augmented sample of additional tubes for any active damage mechanism.

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LER 1-90-004

Original NRC Concern

To determine if Technical Specification 3.0.4. was violated, when Unit 1 entered Mode 6 from Mode 5 while Source Range 1N31 was out of service for an 18 month calibration, solely for the purpose of progress on the critical path schedule; if the shift was subsequently congratulated for making that progress; and if the Shift Superintendent demonstrated a willingness to violate Technical Specifications for the sake of schedule.

Found to be unsubstantiated

NRC Concern

The inadequacy of the root cause determination and corrective actions of LER 1-90-004, in that, human factor problems involving the LCO sheet may have contributed to the Shift Superintendent's failure to note the LCO mode change restriction.

INADEQUATE CORRECTIVE ACTION for not changing the LCO sheet.

NRC Documentation

Technical Specification 3.9.2
Deficiency Card 1-90-0050
LER 1-90-004
12007-C, **Refueling Entry** (Mode 5 to Mode 6)
Unit 1 Control Log
Unit 1 Shift Supervisor Log (2/28/90 and 3/1/90)
LCO Status Sheet 1-90-152
LCO Log (10008-C, P.8 of 11, dated 2/28/90)
14000-1, **Operations Shift and Daily Surveillance Logs,** dated 2/28/90
1R2 Outage Schedule (actual vs. schedule)
Turnover Checklist (11870-C, dated 2/28/90
Completed Procedures, dated 2/28/90 - 12007-C, 14000-1, 11871-C and 11872-C

VEGP Position

VEGP's position is that human factor problems with the LCO sheet was not a significant contributing causal factor in this event. However, due to a number of human factor concerns noted during the 1R2 refueling outage, VEGP has revised procedure 10008-C twice, to enhance usability and human factoring. Furthermore, VEGP will review Procedure 10008-C to determine if further enhancements are warranted.

A/11
A/12

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LER 1-90-004

VEGP Documentation

LER 1-90-004
12007-C, Refueling Entry (Mode 5 to Mode 6)
Unit 1 Control Log, 2/27/90 to 3/2/90
Unit 1 Shift Supervisor Log, 2/27/90 to 3/2/90
Unit SS Relief Checklists, 2/28/90 and 3/1/90
Support SS Relief Checklists, 2/28/90 and 3/1/90
Operations Supervisor Relief Checklist, 2/28/90 and 3/1/90
RO Relief Checklists, 2/28/90 and 3/1/90
BOP Relief Checklists, 2/28/90 and 3/1/90
14000-1, **Operations Shift and Daily Surveillance Logs**, 2/28/90
1R2 Outage Schedule, 2/23/90 thru 3/3/90
10008-C, **Recording Limiting Conditions for Operation**, Rev. 12

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MISSED SURVEILLANCE ON
"CONTAINMENT INTEGRITY VERIFICATION
VALVES OUTSIDE CONTAINMENT"

NRC CONCERN: Required Tech Spec actions may have been delayed by initiating an investigation. Was management pressure a contributing factor?

Concern: Potential concealment of correct Tech Spec LCO entry time to prevent a forced shutdown and immediate notification of the NRC.

Finding: The correct T.S. LCO entry was not concealed.

Concern: Cause for confusion over the Surveillance Task Sheet.

Finding: The cause for the confusion was an inconsistent use of equipment identification numbers on these sheets. Although corrective actions have been taken to eliminate this inconsistency, potential for confusion still exists.

Concern: Extent of emphasis on keeping the plant in operation and limiting NRC notifications.

Finding: There was no indication of unreasonable emphasis on keeping the plant in operation or limiting NRC notifications.

NRC DOCUMENTATION: D.C. 2-90-022

Surveillance 14475-201 Jan 3, 1990, Feb 1, 1990, Feb 28, 1990

Unit II LER 90-001

Control Room Logs from Feb 27 and Feb 28, 1990

NRC Inspection Report 90-10

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VEGP POSITION:

Management does not apply pressure to delay action statement entry. Investigations are only for the purpose of determining if a problem exists.

Timely resolution of problems is required. Suspected problems are promptly reviewed to determine if a problem exists.

LER 90-001 gives details of event, the cause and corrective actions.

The SS acted to instruct the personnel to complete all valves on the procedure and concurrently notified Surveillance Tracking. Surveillance Tracking went to Document Control and concluded we had a missed Surveillance. Surveillance Tracking then contacted the SS and notified him of the discovery and initiated D.C. 2-90-022 in accordance with Procedure 00150-C. The personnel involved acted properly.

VEGP DOCUMENTATION:

Same as NRC Documentation

Task 14475-201 Verification Sheets

DRAFT

RHR PUMP 1B VIBRATION

Original NRC Concern

A non-conservative decision was made concerning the operability of the 1B RHR pump in order to avoid substantial impact to the outage critical path schedule

Found to be unsubstantiated

NRC Concern

A Deficiency Card was not generated in a timely fashion concerning the 1B RHR pump cooler leak and elevated vibration levels.

VIOLATION FAILURE TO WRITE DC.

NRC Documentation

Unknown

VEGP Position

VEGP concurs that a Deficiency Card was not generated in a timely fashion. Since the occurrence of this event, VEGP management has taken positive action to improve the effectiveness of the Deficiency Card Program. These improvements include:

1. Revision of Reactor Trip Review Procedure, 10006-C to specifically require a sign-off indicating a Deficiency Card has been written.
2. Address by General Manager to the PRB stressing the necessity for timely Deficiency Card generation, and memo to all managers from the Technical Support Manager stressing the requirements for timely submittal of Deficiency Cards.

In addition, the Deficiency Card Program has received increased management attention and oversight to ensure Deficiency Cards are generated in a timely fashion. This will ensure that operability and reportability determinations and appropriate engineering evaluations are performed.

VEGP Documentation

Letter, Manager, Technical Support to Department Managers dated 6/22/90.

DRAFT

IMPROPER TCP PROCESSING

NRC Concern

TCP 18028-C-7-90-1 was "back-dated" to avoid violating section 6.7.3.c of Technical Specifications.

NRC Documentation

TCP 18028-C-7-90-1
DC 1-90-282
DC 1-90-283
PRB Minutes 90-81 and 90-82
Procedure 00052-C

VEGP Position

A violation of Section 6.7.3.c of the VEGP Technical Specifications occurred. However, the cover sheet of TCP 18028-C-7-90-1 was not dated 6-12-90 to avoid this violation.

TCP 18028-C-7-90-1 (written against Rev. 7 of the permanent plant procedure), Loss of Instrument Air, was approved by the Operations Manager on 5-31-90. On 6-8-90 the PRB tabled this TCP to allow the Operations Department to determine if additional instructions for Modes 3, 4, 5, and 6 should be added to the revision to strengthen the AOP. Revision 8 of the permanent procedure was prepared by Operations and approved by the PRB on 6-12-90. This revision addressed both TCP 18028-C-7-90-1 and additional instructions for a Loss of Instrument Air in Modes 3, 4, 5, and 6. The Acting Operations Manager understood that the TCP would not be used in the field once Rev. 8 was issued. Upon approval of Rev. 8 of the permanent procedure by the PRB, verbal instructions were given by the Acting Operations Manager to the procedure coordinator to void TCP 18028-C-7-90-1. The TCP was next in the procedure coordinator's possession on 6-15-90. On that date the acting Operations Manager signed the TCP cover sheet and dated it 6-12-90 to reflect his understanding, based on discussions with the procedure coordinator and his verbal instructions of 6-12-90, that the TCP was voided on 6-12-90.

The Acting Operations Manager assumed that the approval of Rev. 8 of the permanent procedure (which he assumed occurred on 6-12-90) resulted in the voiding of the TCP, and that his verbal instruction to the Operations staff was adequate to close-out required paperwork. This was an error and resulted in a failure to comply with Procedure 00052-C, Section 4.6.2 in a timely manner. GPC notes that the minutes of PRB meeting 90-82 indicate the TCP "was voided" on 6-12-90 which reflected the understanding of the Acting Operations Manager.

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IMPROPER TCP PROCESSING

On 6-18-90, the Operations Manager instructed the Acting Operations Manager to write a DC on the inaccurate dating of the TCP close-out sheet and a failure to meet the 14 day period set under Procedure 00052-C, Section 3.2.4. This instruction was a result of normal Technical Support Group review and verification close-out of TCPs.

On 6-22-90 two DCs were written by Technical Support. On the same date the (former) Acting Operations Manager, in preparing a DC on the inaccurate dating of the TCP cover sheet, determined that the permanent procedure, Rev. 8, was not issued until 6-13-90, that the TCP was pulled from the Control Room on 6-13-90 and that the 14 day limit under Procedure 00052-C had been exceeded.

8-16-90
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SEQUENCER INOPERABILITY

NRC Concern

Inadequate information exists for shift personnel to determine which Technical Specification to apply if the sequencer is inoperable. In addition, previous verbal guidance was inadequate.

NRC Documentation

1. Sequencer related work orders
2. Previous sequencer LCO sheets
3. Control room narrative logs
4. Sequencer related surveillances

VEGP Position

The NRC position is accurate in that no Tech. Spec. interpretation exists. Previous guidance connected sequencer inoperability to diesel generator inoperability. Recent information has demonstrated that sequencer inoperability should also be tied to "actuation logic and actuation relays", as found in the instrumentation specifications. VEGP will further review and evaluate this issue to ensure an adequate interpretation exists for the shift personnel.

VEGP Documentation

As above

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8/16/90
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ALTERNATE RADWASTE BUILDING

NRC Concern

Concerned that the FAVA system was installed without performing adequate engineering and safety evaluation (50.59), because the fabrication and quality of the system did not meet the RG-1.143 and ASME code requirements.

Concluded that the FAVA system was originally installed without an adequate safety evaluation. As a result of a VEGP QA finding in early 1989 involving a breakdown in the procurement and failure to meet FSAR commitments, the system was removed from service. Subsequently the system was returned to service following two SEs (dated 11/89 and 2/90) which adequately addressed the use of PVC piping with respect to radiation degradation and pipe rupture. Although these SEs did not address the effects of a break in the hoses (which could result in wall spray down or leakage), the use of hoses and effects of hose breaks (i.e., airborne activity and puddling) were addressed in SER Supplements 3 and 4. Although these SEs did not address high temperature effects our interview indicated that these effects were considered in performance of the SE.

Concluded that the SE performed on 6/90 at the request of RII to evaluate the effects of a FAVA system wall spray down and wall leakage to an unrestricted area have been adequately addressed for the use of the FAVA system, because the FAVA System has a protective cover. However, the June 90 safety evaluation inadequately addresses the potential effects of wall spray down from any other source in the ARB due to erroneous assumptions concerning the release path and the dilution volumes. This is a potentially unreviewed safety question concerning the use of the alternate radwaste building. And as such will be followed as an unresolved item pending further review and evaluation. (Unresolved item concerning unreviewed safety question).

NRC Documentation

IEC 80-18
SSER 3 Section 11.4
SSER 4 Section 11.4
SSER 8 Section 11.4

VEGP Position

The safety evaluation for the FAVA microfiltration system was adequate for use of the system. The calculation performed to evaluate the "spray accident" in the ARB was flawed due to erroneous assumptions regarding the release path. These flawed assumptions do not affect the 50.59 evaluation made for the FAVA unit. The analysis of the "spray accident" in the ARB should not have been included as a revision to the safety evaluation in the FAVA unit. Doing so confused these two separate issues and was not appropriate.

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8/16/90
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During the plant licensing process, details of the construction of the ARB were provided to the NRC. NRC personnel visually inspected the as-built condition of the ARB and associated solid waste processing steams and interface connections that tie other support systems to the equipment in the ARB. Flexible hoses and couplings were in use in the facility at the time of this inspection. The NRC found the facility acceptable for use based on our submittal and their visual inspection with one exception which was subsequently corrected and had to do with exhaust air filtration. This information is well documented in the SER Supplements 3, 4, and 8.

In June, 1990, NRC requested VEGP to perform an analysis of the "spray accident" in the ARB. This analysis was performed but was based on erroneous assumptions as noted above. VEGP can reperform an analysis, but since we know of no one who has performed any similar analysis, a methodology will have to be developed first.

<u>VEGP Documentation</u>

Safety Evaluation dated 2/26/90
Other documentation same as NRC

DRAFT

Date: August 16, 1990
Time: 1600

Snubber Reduction and Use of the Appropriate Technical Specification

NRC Concern

Voluntary entry into a LCO action statement is acceptable for the purpose of surveillance testing but is discouraged for modification work. (See NRC internal correspondence Murley to Martin dated May 18, 1990). NRC has determined that applying the action statement associated with Specification 3.7.8 and then applying the action statement of the applicable system is a correct interpretation of Technical Specification requirements. Specification 3.7.8 is intended for broken snubbers or functional testing and not for other purposes. With respect to snubber reduction you must have a valid safety evaluation which considers the ramifications of performing the modification at power. If the modification renders the system inoperable during the installation, as determined by analysis, then the applicable system action statement must be applied.

Entering a LCO action statement should represent a net safety benefit and be warranted by operational necessity, not just by convenience. The practice should not be abused by repeated entry into and exit from the LCO. Implementation of snubber reduction during power operation is non-conservative.

NRC Documentation

LCO's
Safety Evaluations for DCP's 88-V1N0114, 89-V1N0047
Copies of MWO's for NSCW "A" Train
Letter from Murley to Martin, dated 5/18/90
Letter from W. C. Ramsey to C. C. Miller, dated 8/15/90
Letter from Denton to Norelius, dated 5/27/90

VEGP Position

The position associated with voluntary entry into a LCO action statement is discussed in a separate position paper.

VEGP agrees with the NRC position that applying the action statement associated with specification 3.7.8 then applying the associated system action statement is the appropriate way to implement Technical Specification requirements. In addition, VEGP agrees that when performing a modification a Safety Evaluation must be performed. If the evaluation determines the system is rendered inoperable during the installation process then the action statement associated with the system must be followed and 3.7.8 cannot be applied.

Since snubber reduction increases system reliability by eliminating potential failure modes, implementation of snubber reduction during power operation is conservative.

VEGP Documentation

None

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Date: August 16, 1990
Time: 1600

LCO Action Statement Entry to Implement Design Change

NRC Concern

Entering a LCO action statement should represent a net safety benefit and be warranted by operational necessity, not just by convenience. The practice should not be abused by repeated entry into and exit from the LCO.

NRC Documentation

Letter from Murley to Martin, dated 5/18/90

VEGP Position

Voluntary entry into a LCO action statement for the purpose of implementing a design change is acceptable provided the activity is accomplished within the provisions of the Technical Specification and proper consideration has been given to the impact on plant safety. This position is supported by NRC Standard Technical Specification interpretation which actually endorses voluntary entry into an action statement condition on the basis that the NRC "has structured the Technical Specifications to permit the licensee to exercise judgement within the latitude permitted by the Action Statement language in the Technical Specification".

VEGP implements design changes on safety related systems for the purpose of improving system reliability and thereby enhancing plant safety. VEGP maintains that voluntary entry into a LCO action statement to implement a design change is acceptable and desirable in specific cases. VEGP considers this consistent with industry practice.

VEGP Documentation

STS, Section 3.0 Voluntary Entry Into Action Statements
Issue Date 1/1/82

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8/16/90
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CONTAINMENT HYDROGEN MONITORS

CONCERNS:

I. OPENING VALVES AT POWER
II. ANALYZER OPERATION FOLLOWING SI

I. OPENING H2 MONITOR VALVES AT POWER

Violation of T/S

Will deny

NRC QUESTION
Are the following valves considered containment isolation valves?
HV-2792A
HV-2792B
HV-2793B
HV-2791B

NRC CONCERN
NRC feels they are based on:
FSAR Table 6.2.4-1
FSAR Table 16.3-4

VEGP POSITION
Yes, the above identified valves are containment isolation valves for the A-Train Containment H2 Monitor.

NRC QUESTION
If they are containment isolation valves does Tech Spec 3.6.3 apply to the operation of these valves?

NRC CONCERN
NRC feels Tech Spec 3.6.3 applies for the following reasons:
- Tech Spec 3.6.3 applies to containment isolation valves.
- We say they are containment isolation valves in Tech Spec interpretation to 3.6.3 (1-18-90).
- Operations Procedure 13130-2 page 4 "Caution" statement.
- Maintenance Procedure 24932-2 step 3.2 "Prerequisites or Initial Conditions"
- FSAR 6.2.4.2.3

VEGP POSITION
Tech Spec 3.6.3 applies to containment isolation valve operability. Opening these valves to perform channel calibration of the H2 analyzer does not render the valve inoperable and therefore Tech Spec 3.6.3 is not entered. If one of these valves became inoperable (e.g. would not close, leaked excessively, etc.), then Tech Spec 3.6.3 would apply to that valve and the associated containment penetration.

Tech Spec 3.6.3 interpretation (1-18-90) applies to hydrogen monitor valves as described above.

Procedure 13130-2 confirms the way we want to operate these valves.

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8/16/90
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Procedure 24932-2 was only recently revised to include the reference to the LCO condition. The LCO condition was with reference to breaching the piping boundary outside containment.

FSAR 6.2.4.2.3 states these essential lines are normally closed and remain closed during power operation. The configuration of these valves are normally closed during power operation. Opening these valves to perform calibration does not conflict with the system description in the FSAR.

CALIBRATION PROCEDURE 24551-2

NRC QUESTION
Do we feel step 2.9 still valid, precautions/limitations "may be performed in any plant mode"?

VEGP POSITION
Yes, precaution 2.9 is valid. The procedure for H2 monitor calibration is required to be performed every 92 days on a staggered test basis (i.e. one channel must be tested approximately every 46 days), per Tech Spec 4.6.4.1. Thus Tech Specs recognize this surveillance as one that can be and should be performed at power.

NRC QUESTION
Do we feel it is necessary to open the isolation valves to perform calibration?

VEGP POSITION
Yes, for the following reasons:
1) By establishing a flow path to and from containment we are verifying an open flow path exists.
2) Verification that pump will operate in the normal flow path configuration is confirmed.
3) ALARA concerns associated with positioning the vent valves.
4) Risk associated with vent valve manipulations.

NRC QUESTION
Was operations involved in review and approval of the procedure?

VEGP POSITION
No

NRC QUESTION
Was the 50.59 safety evaluation performed adequately?

VEGP POSITION
Yes

8/16/90
1600

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NRC QUESTION
When this issue was pointed out 8/7/90 on Unit 2, why was test performed on Unit 1 the next day?

VEGP POSITION
When the issue was brought up with the Operations Manager, he began gathering information on the issue. At the time he felt our procedures were correct and did not review the next day's activities. When shift personnel were made aware of the NRC concern the test was terminated immediately.

NRC QUESTION
What Tech Spec requires LLRT testing of the system?

24910
24930
24931
24932
24933

VEGP POSITION
Procedures 24930, 24931, 24932, 24933 satisfy the requirements of Tech Spec 4.6.1.2.d for components defined in FSAR Table 6.2.4-1. Procedures 24910 and 24932 satisfy the requirements of leakage assessment of Tech Spec 6.7.4.A.

NRC QUESTION
Is leak rate testing performed on these containment isolation valves added to overall containment leak rate?

VEGP POSITION
Leak rate for these isolation valves is added to the total type B and C leakage. It is not added to type A results.

NRC QUESTION
Evaluate applicability of Tech Spec 6.7.4.A to this system

VEGP POSITION
The piping outside containment is covered under the leakage assessment program as addressed in Tech Spec 6.7.4.A.

NRC QUESTION
Do we feel we violated Tech Spec 3.6.3 on the following two occasions?

Unit 2 0411 8/6/90 to 0122 8/7/90
21 hrs. 11 min.
Unit 1 2053 8/7/90 8/8/90
18 hrs. 47 min.

VEGP POSITION
No, the containment isolation valves were not inoperable on these dates. Further, in the past calibrations have been scheduled in accordance with Tech. Spec. requirements (approximately every 90 days).

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II. ANALYZER OPERATION FOLLOWING SI

NRC QUESTION
Are the analyzers placed in service 30 minutes after a safety injection?

Requirement: NUREG 0737 IIF1 Attachment 6
TMI requirement

Provide station position relative to NUREG 0737 also provide proof of implementation.

NRC has looked the following places:
19000-C
19251-C
Loss of primary & secondary coolant

Reference SER 6-4

VEGP POSITION
Procedure 19010-C (Loss of Primary or Secondary Coolant) step 12 currently addresses obtaining containment H2 samples following an SI. VEGP intends to enhance this procedure relative to placing the H2 analyzers in service for this purpose.

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Date: ugust 16, 1990
Time: 1600

Precision Heat Balance

NCU

NRC Concern

The corrective actions associated with LER 90-015 (Failure to Calibrate Computer Points Prior to Precision Heat Balance Flow Measurements) are technically correct, the decision not to re-perform the surveillance test was non-conservative.

NRC Documentation

LER 90-015-00
DC cards, RCN's Reactor Engineering Calculations
Completed test procedures

VEGP Position

The decision not to reperform the surveillance was conservative, based on Engineering evaluation of available data. Additionally, all associated data was reanalyzed assuming potential calibration errors. This reanalysis verified the initial engineering analysis.

When Unit 1 final feedwater temperature instrumentation was determined to be out of calibration, examination of the data indicated sufficient margin to address the out of calibration condition. Reanalysis of the data considering the out of calibration condition confirmed the conclusion.

For Unit 2 the calibration of final feedwater temperature was never suspect. Again, examination of calibration data taken after the test indicated sufficient margin to address potential miscalibration problems. Reanalysis of the data also confirmed this conclusion.

VEGP Documentation

None

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8/16/90
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POTENTIAL
MINOR CONCERN

PERSONNEL ACCOUNTABILITY

NRC Concern

Holding shift supervision accountable for the number of reactor trips, LER's, and ESFAS actuations has a potential negative influence on plant safety because personnel might not be open about reporting these types of plant problems.

NRC Documentation

Typical Shift Superintendent Accountabilities

VEGP Position

These accountabilities enhance reactor safety because they focus personnel attention on safety and compliance issues. Reporting problems is required to achieve good SALP ratings and is part of shift supervision accountabilities.

VEGP Documentation

Typical Shift Superintendent Accountabilities
1990 Organizational Goals
Performance Appraisal Forms

1600 8-16-90

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3.0.3 1 HR. ACTIONS

concern
one operator
said don't do
anything for
3 hrs.

NRC Concern

Inadequate documentation exists to demonstrate all actions taken during the first hour after entry into Technical Specification 3.0.3.

NRC Documentation

Control Room Narrative Logs

VEGP Position

Documentation of all actions taken during the first hour after entry into Technical Specification 3.0.3 does not exist as stated by the NRC. However, this information (documentation) is not procedurally required, nor is it a regulatory compliance issue. Appropriate actions have been taken in the past 3.0.3 entries to meet the time table of the action statement.

VEGP Documentation

As Above

DRAFT

Date: August 16, 1990
Time: 1600

ESFAS REPORTABILITY ISSUE

observation

NRC Concerns

In regard to reportability of ESF actuation NRC has developed a position that "If, for any reason (except expected responses to testing) the ESF components are caused to operate, then an ESF actuation did occur". This position was formulated in response to GPC Corporate internal memo (June 11, 1987) which provides guidance concerning ESF actuation reportability.

NRC Documentation

1. Internal NRC memo dated July 12, 1990 from Charles E. Rossi to Gus C. Lainas.

2. Internal GPC memo dated June 11, 1987 from R. Baker to L. T. Gucwa.

VEGP Position

The June 11, 1987 letter was written by a member of the Corporate Nuclear Safety and Licensing Department to his Supervisor. The information contained in the attachment to this letter was intended to be used as guidance when determining ESF actuation reportability. This information was never adopted by the Vogtle Electric Generating Plant (VEGP) to be used for reportability guidance. The Vogtle practice has been to report the ESF actuation regardless of "what caused the actuation" or "how the actuation occurred". Based on discussions with individuals who review deficiencies for reportability and a review of past deficiency evaluations identified no instances where the position described in the June 11, 1987 letter was utilized in ESFAS reportability determination at VEGP.

VEGP Documentation

1. W. F. Kitchens memo to OSOS dated June 9, 1987
   RE: ESF Actuation

2. Sort of DC's by keyword "ESFAS"

3. List of ESFAS LER's for Vogtle

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TECHNICAL SPECIFICATION INTERPRETATIONS

Non-conservative

NRC Concern

1. The control of the generation, approval and distribution is not formal enough.

2. The level of review and approval of Tech. Spec. interpretations is not a high enough level (i.e. PRB review and concurrence should be required). In fact, Tech. Spec. 6.4.1.6 a or d may apply.

3. If NRC guidance is used, author of guidance should be sent a "Info Copy".

NRC Documentation

None

VEGP Position

Since VEGP Tech. Spec. interpretations are not designed to modify the intent or breadth of the Technical Specification but merely to clarify the specification for the on-shift operations crews, a formal process is not required and T.S. 6.4.1.6 a and d are not applicable. The Operations Manager, being the senior member of plant management required to maintain a Senior Reactor Operators License, is the appropriate approval authority for the Technical Specification interpretations generated. As required by subject matter, input from other sections including Nuclear Safety and Compliance, is utilized in the development of the interpretations.

In order to provide the NRC with information on VEGP positions on Technical Specifications, the Resident Inspectors will be provided with copies of our interpretations.

VEGP Documentation

None

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1600

OVERTIME ?

D

NRC Concern

No provision exists in our procedures that would prevent operators from working an excessive amount of overtime during an extended period of time, i.e. monthly or yearly. Our restriction of no more than 72 hours worked in a seven day period would not prevent excessive overtime on a monthly or yearly basis.

NRC Documentation

None

VEGP Position

Operations department personnel use established procedures and guidelines, based on existing regulatory guidance, that limit the possibility of this situation occurring and ensure compliance with the regulations governing overtime.

Procedure 0C005-C gives the overtime guidelines and requires the department head to evaluate and approve the consistent use of overtime. The GMNP, or designee is also required to review excess overtime assigned to individuals each month to ensure proper authorization per Figure 1 of this procedure. Also, he reviews the overtime to ensure that assignment of excess overtime does not become routine.

For operators working under the union contract, additional guidance is provided for overtime assignment and equalization in the Memorandum of Agreement, paragraph 49.

Based on review of overtime records, LER's, Reactor Trips, and ESFAS Actuations, no conclusions can be drawn that indicate excessive overtime has caused operator fatigue or an increased frequency of operator errors. However, VEGP intends to review this item for potential enhancements.

VEGP Documentation

Week at a Glance
Paragraph 4C, Memorandum of Agreement
Procedure 0C005-C

DRAFT

NON-LICENSED OPERATOR TRAINING

NRC Concern

1. The PEO training program does not include under-instruction watches for building qualification.

2. The training program may not be adequate to train and evaluate the ability to make routine rounds; some operators may not have completed their actual rounds task properly.

NRC Documentation

In NRC interviews with new building operators some said they did not actually perform their rounds in training.

VEGP Position

1. Qualification is based on the successful completion of required knowledges and skills, which are arrived at through the analysis phase of a systematic approach to training (SAT) process. The INPO accredited program does not rely on any arbitrary number of under-instruction shifts for qualification. However, due to requests from Plant Equipment Operators surveyed, VEGP will re-evaluate the addition of under-instruction watches to the building operator qualification checklists.

2. The routine conduct of rounds is an identified task with associated supporting knowledges in the PEO training and qualification program. A comprehensive instructional unit provides sufficient information and guidance to assure that operators have the ability to perform routine rounds. A specific weakness has been identified in the implementation of the evaluation process for the task conducting rounds. This weakness will be handled through the SAT feedback process.

VEGP Documentation

Procedure 11958-C, "Auxiliary Building Operator Training Qualification Checklist"
Qualification Signoff Criteria Cluster 51 - NLO Administrative Duties
VEGP Instructional Unit NL-IU-51401-001-C, Conduct Auxiliary Building Rounds
VEGP Instructional Unit NL-IU-51401-002-C, Conduct Control Building Rounds
Management Observation Report (MORE) - TQ.3, "On-The-Job-Training"

DRAFT

OPERATOR ROUNDS GENERAL INSPECTION

NRC Concern

There are differences in the depth of general inspections performed by operators during their rounds.

NRC Documentation

NRC observation of rounds by new Auxiliary Building Operators.

VEGP Position

The general inspection is intended to identify any type of abnormal condition which may develop. The procedural guidelines are accordingly very broad. It is not our intent to detail every possible check which the operator could make in our procedures. The guidance in Operations Procedure 10001-C "Logkeeping", describes the overall areas of inspection required of the Plant Equipment Operators.

We expect there will be differences in the focus of different operators based on their personal experience and shift supervision instructions. This diversity is a plus to increase the breadth of the general inspection.

For the rest of 1990, VEGP will increase the number of supervisors and managers doing Management Observation Reports on operator rounds, during both day and night shifts. These observations will be reviewed to establish a baseline performance standard and any needed corrective actions will be implemented in procedures, training and practice.

VEGP Documentation

Procedure 10001-C, Logkeeping
Training Cluster 51-NLO Administrative Duties

DRAFT

8/16/90
16:00

ELECTRICAL SEPARATION ZONE 80

Reportability Review Inadequate

NRC Concern

Upon an LOSP a postulated fire in Zone 80 would render Train A inoperable, and may trip the Train B Diesel Generator output breaker. VEGP should insure that no equipment required to cope with this condition would be damaged by the fire while the diesel generator output breaker is being reclosed.

NRC Documentation

DC 1-90-299 and 2-90-080

VEGP Position

1. The design requirement for a fire in this area is to be able to shut down the plant using Train "B" equipment (FSAR 9A.1.40.L.1a). This scenario does not affect our ability to safely shut down the plant, is not a condition outside our design basis, and is therefore not a reportable condition.

   A. The postulated fire scenario would not damage the Train "B" safety related equipment necessary for safe shutdown of the plant. The D/G would continue to run and an annunciator would indicate D/G trouble.

   B. The operator would be required to observe a loss of power on the "B" Train safety related bus, recognize there was a fire in the room where the attached non-safety related bus is located, separate the nonsafety related bus from the safety related bus, and reclose the D/G output breaker. There is adequate time for the operator to take these actions.

2. As a conservative measure, the feature which could cause the D/G output breaker to open in this scenario is being eliminated so as not to rely on operator action to reclose the breaker.

3. VEGP intends to provide additional information to support these conclusions.

VEGP Documentation

FSAR 9A.1.40.L.1a
Letter #SG-9471
Letter #SG-9510
DC 1-90-299
DC 2-90-080

AUGUST 16, 1990

DRAFT

AREAS OF CONCERNS	NRC	VEGP CONTACT	CORPORATE CONTACT
* D/G Records Starts/Failures	Pete Taylor	G. Frederick	
* 3/1/90 S R Monitor Inop Mode Change	Neal Hunemuller	JES/D. Carter	
* Missed Surv. Cont. Isol.	Neal Hunemuller	JES/S. Swanson	
* March 15 RHR Train B	Ron Aiello	JES/J. Gasser	P. D. Rushton
* Temp. Change Notice to AOP 18028-C-7-90-1	Robert Carrol	JES/J. Cash	
* ESFA Sequencer Out of Service	Robert Carrol	JES/Horton	J. A. Bailey
* Alternate Radwaste Building	Ron Aiello	Ron LeGrand/JES	P. D. Rushton
* Snubber Reduction	Larry Garner	Gus Williams	Ward/Stringfellow
* LCO Action Statement	Larry Garner	Gus Williams	Ward/Stringfellow
* Cont. Integrity Hydrogen Monitor Valve Opened	Morris Branch	Dean Gustafson	Ward/Stringfellow
* Precision Heat Balance	Morris Branch	Gus Williams	B. Florian
* Personnel Accountability Methodology for Reporting	C. VanDenburgh	JES/GB	
* Tech. Spec. 3.0.3 Philosophy	J. D. Wilcox	J. E. Swartzwelder	J. Stringfellow
* ESFAS Reportability	J. D. Wilcox	R. M. Odom	J. A. Bailey
* Tech. Specs. Interpretation	Morris Branch	J. E. Swartzwelder	J. Stringfellow
* Overtime/Training & Qualification	Larry Garner	J. E. Swartzwelder	
* Electrical Separation Zone 80	Larry Garner	M. Horton	P. D. Rushton
* T. S. 3.4.7.3 CCW	J. D. Wilcox	J. E. Swartzwelder	

A/127

A/60

NRC Concern

1. The NRC is concerned about the incorrect number of diesel starts reported in LER 1-90-06 and the number of starts presented to the NRC on April 9, 1990 and in the confirmation response letter of April 9, 1990. The major issue remaining is to try and determine through personal interviews, how the number of 19 for diesel 1B was arrived at in the April 9 letter to the NRC. The revised response to LER 90-06 did not clarify the number of starts reported to the NRC April 9.

2. The inspector noted that documentation provided by Operations to support diesel trending (14980-C and 13145-C data sheets) does not contain an adequate description of what happens during the start attempt. The plant is not interpreting Reg Guide 1.108 properly with regard to reporting valid and non-valid failures. There may be valid and non-valid failures that were not reported. The NRC does not consider the current status of reporting diesel failures to be in compliance with commitments made to the NRC in Violation 50-424/87-57.

NRC Documentation

The NRC has reviewed the diesel start log and supporting documentation (14980-C and 13145-C data sheets). The NRC currently believes some problems identified on 14980's and 13145's should be classified as non-valid failures and reported to the NRC. The NRC has requested and received written analysis to explain the disposition of the following 1B diesel starts: #'s 123, 124, 132, 133, 134, 136, 160, 161, 162, 164, 165, and 190. LER 1-90-06, revision 1; QA Audit Report OP26-90/33; QA Audit Report OP09-90/31; and Special Report 1-90-05, dated August 7, 1990; GPC confirmatory action letter dated April 9, 1990.

VEGP Position

1. The error made in the number of diesel starts reported to the NRC on April 9, 1990, and in LER 1-90-06 is attributed to two factors:

   a. The testing as described in LER 90-06, revision 0, was in the "context of" and "in reference to" the diesel control systems. The first two sentences of the 5th paragraph explain actions taken with regard to sensor calibrations and control system testing. In this context, the test program correlates to testing discussed with the NRC on April 9, 1990, and reported in the April 9, 1990, confirmatory letter. The LER 90-06 comment of "subsequent to the test program" was not intended to exclude successful diesel starts before declaring the diesel operable. As a result, diesel starts after testing of the control systems, but before a declaration of operability were counted. The transmittal letter for LER 90-06, revision 1, describes the confusion and attempts to clarify the concern by redefining the types of starts and the point of counting.

~~A/53~~
Release A/R 8

b. LER 90-06, revision 1, was intended to clarify any inadvertent "misleading" of the NRC on successful operation of the diesel control systems. When Vogtle Management was aware of the problem in LER 90-06, revision 0, management notified the NRC Residents. Also at the corporate office on 6/11/90, W. Shipman contacted Ken Brockman and on about 6/11/90, W. G. Hairston, III, contacted Stu Ebneter of NRC Region II. The revised LER was submitted on 6/29/90.

2. After a thorough review of Reg Guide 1.108, Engineering Support (Mike Horton) agreed that all diesel start problems have not been reported as failures. GPC's response to NRC Violation 424/87-57 committed to report such equipment problems as failures; however, due to internal administrative problems, the commitment was not implemented. Engineering Support intends to review diesel start records for any unreported failures.

VEGP Documentation

- LER 1-90-06, revision 1; QA Audit Report OP26-90/33; QA Audit Report OP09-90/31; and Special Report 1-90-05, dated August 7, 1990; GPC confirmatory action letter dated April 9, 1990.
- 1B diesel start analysis available 8/15/90 and Reg Guide 1.108 position from Engineering Support.

8/17/90
Time: 12:00

Response to NRC Question Concerning
Diesel Starts Reported on April 9, 1990
and in LER 90-06, Revisions 0 and 1

<u>Question #1</u>

1. Who prepared the slide for the 4/9/90 presentation?
   Answer: G. Bockhold, Jr., J. P. Cash, and K. Burr working as a group.

2. Who approved use of the slide?
   Answer: G. Bockhold, Jr.

<u>Question #2</u>

1. Who prepared the confirmatory letter of April 9, 1990?
   Answer: C. K. McCoy, J. A. Bailey, W. G. Hairston, III as a group.

2. Who approved the letter?
   Answer: W. G. Hairston, III

<u>Question #3</u> (with regard to LER 90-06, revision 0, dated 4/19/90)

1. Who prepared the LER?
   Answer: Several draft revisions of the LER were prepared by Tom Webb and others of the NSAC group of the Vogtle Site Technical Support. These drafts were reviewed and commented on by the Plant Review Board. The final revision of LER 90-06, revision 0 was prepared by a phonecon between site management and corporate management. Those participating are believed to be G. Bockhold, Jr., A. L. Mosbaugh, J. G. Aufdenkampe, W. Shipman.

2. Who reviewed the LER?
   Answer: All revisions of the LER were reviewed by the PRB and the General Manager-Plant Vogtle.

3. Who approved the LER?
   Answer: The LER was approved by W. G. Hairston, III

<u>Question #4</u>

1. Who prepared the cover letter for LER 90-06, revision 1?
   Answer: The cover letter was prepared by H. W. Majors of the corporate staff. This letter was prepared under the guidance of W. G. Hairston.

2. What was the purpose (intent) in the wording of the cover letter with regard to the number of diesel starts?
   Answer: The cover letter was intended to document discussions with NRC Region II to clarify the starts documented in LER 90-06, revision 0. By picking a well defined point to specify "subsequent to the test program" it was possible to identify a substantial number of successful diesel starts. This was intended to remove any additional ambiguity.

<u>Question #5</u>

1. Who in corporate added the words "subsequent to the test program" in LER 90-06, revision 0?
   Answer: Corporate Licensing personnel in conjunction with the phone conversation described above made editorial changes as directed. Those present during the phone conversation are thought to be W. Shipman, G. Bockhold, Jr., A. L. Mosbaugh, J. G. Aufdenkampe, and J. Stringfellow.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
101 MARIETTA STREET, N.W.
ATLANTA, GEORGIA 30323

AUG 22 1990

Docket Nos. 50-424, 50-425
License Nos. NPF-68, NPF-81

Georgia Power Company
ATTN: Mr. W. G. Hairston, III
Senior Vice President -
Nuclear Operations
P. O. Box 1295
Birmingham, AL 35201

Gentlemen:

SUBJECT: CONFIRMATION OF MEETING - ENFORCEMENT CONFERENCE - VOGTLE (NUREG-1410, NRC INSPECTION REPORT NOS. 50-424/90-16 AND 50-425/90-16)

This confirms a telephone conversation between Mr. C. K. McCoy of your staff and Mr. K. Brockman of my staff on August 10, 1990, concerning an Enforcement Conference to be conducted at the NRC Region II Office at 10:00 a.m., on September 5, 1990. This meeting was requested to discuss numerous items identified by the Incident Investigation Team (IIT) which was chartered in response to the Site Area Emergency event of March 20, 1990. The team's findings are documented in NUREG-1410, "Loss of Vital AC Power and the Residual Heat Removal System During Mid-Loop Operations at Vogtle Unit 1 on Ma[illegible] 20, 1990."

The first issue to be discussed concerns the site's failure to make the required emergency notifications to state and local government agencies in a timely manner. Of particular concern is the root cause of this failure and what was the sequence of events which precluded the required information from being provided through alternate communications means. Additional information concerning this issue is provided in NRC Inspection Report Nos. 50-424/90-16 and 50-425/90-16.

The second issue to be discussed concerns the inability of the site personnel to establish containment integrity within the time limits required by procedures 12006-C, "Shutdown to Cold Shutdown," and 27505-C, "Opening and Closing Containment Hatch." Your response to Generic Letter 88-17 made numerous commitments concerning analyzing the conditions and time requirements necessary to ensure an appropriate establishment of containment integrity and for developing the procedures necessary to ensure the closing of the equipment hatch before the plant could enter reduced inventory conditions. You should be prepared to discuss how you determined the time limit for the equipment hatch closure, how you verified whether the time limit could be met, and why the hatch was not able to be closed in the required time on March 20, 1990. Both the worst case environmental conditions and the conditions on the day of the event should be addressed.

A/29

~~9010030285~~ 4PP

The third issue concerns the failure of the emergency diesel generator to provide AC power as intended. The plant has experienced a long litany of previous indicators of diesel generator problems, including a common need to "shake down" the system after maintenance outages. These "shake-downs" were often in response to, what now seems to have been, failures within the pneumatic control system. You should address why your root cause analysis program did not investigate the totality of continuing diesel problems which had been experienced at the plant and what has been done to improve the root cause analysis program at the Vogtle site.

Should you have any questions regarding these arrangements, we will be pleased to discuss them.

Sincerely,

Luis A. Reyes, Director
Division of Reactor Projects

Enclosure:
Proposed Meeting Agenda

cc w/encl:
R. P. McDonald
Executive Vice President-Nuclear
  Operations
Georgia Power Company
P. O. Box 1295
Birmingham, AL 35201

C. K. McCoy
Vice President-Nuclear
Georgia Power Company
P. O. 1295
Birmingham, AL 35201

G. Bockhold, Jr.
General Manager, Nuclear Operations
Georgia Power Company
P. O. 1600
Waynesboro, GA 30830

J. A. Bailey
Manager-Licensing
Georgia Power Company
P. O. Box 1295
Birmingham, AL 35201

(cc w/encl cont'd - See page 3)

cc w/encl: Cont'd
Ernest L. Blake, Esquire
Shaw, Pittman, Potts and Trowbridge
2300 N Street, NW
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J. E. Joiner, Esquire
Troutman, Sanders, Lockerman, and Ashmore
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127 Peachtree Street, NE
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D. Kirkland, III, Counsel
Office of the Consumer's Utility Council
Suite 225, 32 Peachtree Street, NE
Atlanta, GA 30302

Office of Planning and Budget
Room 615B
270 Washington Street, SW
Atlanta, GA 30334

Office of the County Commissioner
Burke County Commission
Waynesboro, GA 30830

Lonice Barrett, Commissioner
Department of Natural Resources
205 Butler Street, SE, Suite 1252
Atlanta, GA 30334

Thomas Hill, Manager
Radioactive Materials Program
Department of Natural Resources
878 Peachtree St., NE., Room 600
Atlanta, GA 30309

Attorney General
Law Department
132 Judicial Building
Atlanta, GA 30334

State of Georgia

ENCLOSURE

PROPOSED MEETING AGENDA
Georgia Power Company
Enforcement Conference
10:00 a.m., September 5, 1990

A. Opening Remarks	S. D. Ebneter W. G. Hairston, III
B. General Discussion	C. K. McCoy
1. Circumstances surrounding the failure to make a Site Area Emergency notification in a timely manner	G. Bockhold
2. Inadequate root cause analysis program	G. Bockhold
3. Circumstances surrounding the inability to establish containment integrity in a timely manner.	G. Bockhold
C. Closing Remarks	W. G. Hairston, III S. D. Ebneter

In eroffice Correspondence

Georgia Power

DATE: August 28, 1990

RE: Vogtle Electric Generating Plant
Audit Finding OP13-90/25 #430
Log: ELV-02051
Security Code: NC

FROM: C. C. Miller

TO: G. R. Frederick

Procedure revisions resulting from the findings in this audit report will be completed by December 21, 1990.

We welcome the opportunity to discuss any details in addition to those given in the audit report.

C. C. Miller

CCM/clr

xc: Georgia Power Company
P. D. Rushton
M. J. Ajluni
J. A. Rodgers
NORMS

#844

A1155
A1130
close

AUG-29-'90 14:17 ID:SONOPCO-VOGTLE TEL NO:1-205-877-7885 #716 P01

REC'D 6/13/91
FROM A.
MOSBAUGH.

GEORGIA POWER COMPANY
Inverness Building 40
P.O. Box 1295
Birmingham, Alabama 35242

TELECOPY COVER SHEET

SONOPCO-VOGTLE -- 4TH FLOOR
Telecopier: (205)-877-7885
Verfiy : (205)-877-7897

DATE: August 29, 1990

NUMBER OF PAGES: 13
(Excluding Cover Page)

RECIPIENT: Please notify us if you have problems receiving this telecopy.

FROM:	TO:
NAME: Lewis Ward	NAME: George Bockhold
EXTENSION: 7802	EXTENSION: 3118
LOCATION: Inverness	LOCATION: Admin Bldg.
	Plant Vogtle
	TELECOPIER #:

SENDER: Should this document be returned to you after it has been sent?

XX YES ___NO

COMMENTS:

Release

A-82

A-131

SUMMARY OF DIESEL GENERATOR

PRESSURE SENSOR (CALCON) PROBLEMS AT VOGTLE

A. Statement of Problem: NUREG-1410 lists ____ Calcon pressure switch problems, which far exceeded industry experience. This precursor indicator to the 3/20/90 event had not been adequately addressed.

1. Pressure Switch Out-of-calibration problems:

   . 12 DG pressure switches have been out-of-calibration and have required readjustment over the lifetime history of Vogtle from NUREG-1410.

   . This covers a population of 24 sensors (6 on each of 4 engines), over a lifetime of about 92 sensor-years.

   . Typical out-of-calibration values range from ____ to ____ psig.

   . This data does not reveal any unexpected results based on the device type (hydraulic-pneumatic), setpoint drift amount, or frequency. Thus, a formal root cause evaluation would not be warranted.

2. Pressure Switch (Defective) problems:

   . 5 pressure switches were replaced (prior to 3/26/90) due to being termed "defective" by the technician.

   . For 4 of these sensors, no reason for the failure was determined.

   . After the 5th "failure" (P-3 relay removed from DG 1A on 3/25/90 as part of post-event troubleshooting), the switch was sent to Cooper Industries for failure root cause determination. The sensor was determined to be operating properly.

   . Subsequent to the 3/20/90 event, all 3 low lube oil pressure switches were removed from the 1A DG to investigate a sensor malfunction alarm that had occurred during the event. The switches were tested by Cooper Industries on May 30 and 31, 1990 with the following results:

   A. The "B" and "C" switches were operating properly and were set correctly.

   B. The "A" switch was stuck in the tripped condition. This condition was stated by Cooper to be the same as reported by Cooper's 10CFR21 report addendum of May 12, 1988. The Part 21 states "Devices that are already installed and operating after several hours between tests have demonstrated their reliability. IMO Delaval recommends that all devices not installed, or that are installed but have not operated for several hours between tests, be returned to IMO Delaval for remachining, inspection and testing."

C. In 1988 VEGP reviewed this Part 21 and returned all spare sensors for rework. Installed sensors were not returned since they were believed to have met the reliability conditions stated in the Part 21.

D. Following the May 31, 1990 testing results from Cooper, modified switches were ordered to replace all installed Calcon pressure switches. These switches (6 per engine) were received, calibrated and installed; with the last DG completed on June 15, 1990.

E. Cooper was requested by Georgia Power Company to clarify the original Part 21 notification. On June 8, 1990, Addendum 3 was issued stating:

"Our recommendation of May 12, 1988 may have been confusing and in light of this failure after 9 years, it is appropriate to restate our recommendation. Cooper Industries recommends that all pressure sensor devices, Cooper P/N F-573-156, be modified or replaced by devices identified as Calcon P/N B4400B."

B. Root Cause

- A design deficiency existed in the Calcon pressure switches. Replacement switches had not been installed due to mis-interpretation on the May 1988 Part 21 notice.

C. Corrective Action

- All Calcon pressure switches have been replaced with the improved model within the last 6 months.

- The Part 21 was reissued for clarification.

D. Significance of Problem

- None in fact. Multiple simultaneous failures were unlikely, but could have rendered one or more DG inoperable.

E. Conclusion:

- Georgia Power Company took prompt actions to identify and correct the observed switch failure in 1990.

SUMMARY OF DIESEL GENERATOR

TEMPERATURE SENSOR (CALCON) PROBLEMS AT VOGTLE

A. Statement of Problem: NUREG-1410 lists _____ Calcon temperature switch problems, which far exceeded industry experience. This precursor indicator to the 3/20/90 event had not been adequately addressed.

- Following the 3/20/90 event, all 3 Jacket Water Temperature Switches were removed from the 1A DG for testing.

- A test program to determine the as-found condition and failure mechanism of these switches was developed. The purposes of the test were:

  1. Determine the reliability and potential failure mechanisms of two new sensors.

  2. Determine the cause of failure of the installed switches.

- This test program was conducted at Wyle Laboratories in Huntsville, Alabama from 4/23/90 to 5/4/90. Copies of the report were furnished to the NRC and all TDI Owners through the Owners Group. Pertinent conclusions from this test program include:

  1. Insufficient temperature stabilization period prior to calibration - The sensor exhibits a setpoint shift as the sensor body and internal components change temperature. Note: Failure to recognize this phenomena and properly compensate for it during switch calibration was a contributing cause of one switch trip on DG 1A on 3/20/90, and subsequently on DG 1B on 5/22/90 during switch replacement. Note: This phenomena is undesirable in a sensor that is designed to sense temperature.

  2. Contaminants on the temperature sensor (tip) - Direct immersion of the sensor tip in a calibration bath can result in residue buildup that can affect the setpoint. Although this is not a standard practice, isolated cases may have occurred during previous calibrations, which could have contributed to the numbers on the chart.

  3. Water bath heatup rate - A slow, controlled bath heatup rate is necessary to allow the sensor temperature and bath temperature to be approximately the same, while avoiding excessive sensor heatup. Previous calibration procedures did not recognize this affect. This affect could have produced some of the previous setpoint adjustments that contributed to the numbers on the chart.

  4. Thermowell setscrew tightness - This produced a 2^0 setpoint shift. Although not large, in relation to the tolerance band of $\pm 4^0$, it could have been a factor in making previous setpoint adjustments that contributed to the numbers on the chart.

5. Spacer-tube tightness - The spacer tube can self-loosen when not locked with thread-sealant, which produces a setpoint shift of about 80° per turn. This could have been a direct contributor to some of the past failures or setpoint shifts. The vendor has stated that all new switches are supplied with sealant; however, this deficiency has continued to be observed.

6. Internal contaminants - Several switches had internal contaminants in the poppet valve area, consisting of thread sealant material and metal slivers apparently from the inlet air port threads. Subsequent examination of new switches at the plant site revealed similar contamination, concluding that the manufacturer or vendor can be introducing contaminants during calibration. Internal contamination was the direct cause of one switch failing on DG 1A on 3/20/90, and was a contributor to the second failure.

- Upon completion of the above Wyle testing program, a calibration procedure was written specifically for these switches to include:

1. Disassembly of the switch, internal cleaning, and provisions to prevent re-contamination.

2. Inspection and application of thread-sealant to the spacer tube.

3. Requirements for the sensor to be calibrated in a thermowell.

4. Temperature stabilization prior to calibration.

- The new procedure was used to calibrate 3 new jacket water temperature sensors, which were installed on DG 1B on 5/22/90. During the subsequent maintenance start of the engine, these sensors tripped. They were then removed and carried to Wyle Laboratories where as-found setpoint testing showed all 3 to trip between 160° - 166°F.

- Subsequent evaluation of the differences between Wyle and plant test techniques led to the following conclusions:

1. The plant bath had internal flow blockage that did not permit a uniform temperature or heatup rate in the bath. This produced several degrees difference at the reference vs. test specimen locations.

2. The soak temperature requirement was poorly worded. Although the minimum soak times from the Wyle report had been observed, the sensors had been soaked too long (up to 6 hours) at near the setpoint (200°F). This resulted in further adjustments, longer soak, and more adjustment; producing switches that were far out of adjustment when subsequently returned to their normal ambient operating condition.

- Upon correction of the above bath and procedural deficiencies, consistent settings were obtained at the site on the same switches that had been reset at Wyle Labs.

- All jacket water temperature switches on all DG's were removed, cleaned and recalibrated per the latest procedure in early June 1990.
- A copy of the latest procedure, with lessons learned, was provided to the TDI Owners Group in June 1990.

B. Root Cause

- Internal contamination can cause a properly calibrated switch to trip.
- Calibration of the switches was inadequate to ensure the desired setpoints.

C. Corrective Action

- A calibration procedure that cleans and properly controls the calibration requirements has been written and implemented.
- Reliability of the basic switch component was established at an independent laboratory.
- Switches have been defeated in the emergency start mode.
- Currently evaluating replacement of pneumatic sensors with the vendor.

D. Significance of Problem

- Trip of DG 1A on 3/20/90, and trip of DG 1B on 5/22/90.

E. Conclusion

- Georgia Power Company took prompt action to identify the cause of temperature switch malfunctions, following the 3/20/90 event.
- Prior to the 3/20/90 event, in retrospect, the previous calibration drift and failure history should have been recognized as a problem to be resolved. However, discussions with the switch manufacturer did not produce any clues that the switch has to be inspected, cleaned and calibrated under the conditions that Georgia Power subsequently developed. In fact, the 10CFR21, Addendum 3, June 8, 1990 update to the NRC states: "While no specific component failures have occurred, the setting and verification of same, is procedure sensitive." Note that this was after VEGP had resolved the issue internally.

DIESEL GENERATOR AIR START VALVE

FAILURE AND ROOT CAUSE EVALUATION

A. History of Problem:

- 1/24/90 - DG 2A rolled but failed to start during routine surveillance testing. One air receiver had been isolated to test independent starting on the other bank. The operator noticed an air leak on one of the air start valves solenoids, unisolated the second bank, and successfully started the engine.

- 1/25/90 - (a) DG 2A start was attempted to satisfy Tech Spec action requirements. The engine slow rolled but did not start. A second start attempt was successful.

  (b) In order to isolate the cause of the start failure, troubleshooting was conducted including: replace air start distributor filters, checking operation of the governor, fuel racks, fuel system and cylinder air start valves. No apparent problems were found.

  (c) The engine was started 3 more times with no problems.

- 4/12/90 - DG 2A rolled but did not start during normal surveillance. Operators decided that the start pushbutton had not been depressed long enough and reran the test. The engine successfully started, which seemed to confirm the operator's decision. The first attempt was not considered to have been valid, and was not reported.

- 7/5/90 - A similar event occurred on DG 1B during surveillance testing. Again, the operator attributes this to short pushbutton action, and did not log or report the start attempt or failure.

- 7/11/90 - DG 2A again slow rolled and failed to start. The system engineer had developed a test plan for troubleshooting the start pushbutton and seal-in circuit root cause of the previous observed problems on 2A. This investigation led to the following conclusion:

  (a) The seal-in circuit for the start pushbutton seals in any attempted start. Operator belief that the start pushbutton had to be depressed for a certain minimum time period stemmed from a simulator phenomena, and does not exist in the plant.

  (b) Discussions with Cooper Industries resulted in parallel investigations in the air start solenoid valves, air start distributors, air line routing, and individual cylinder air start valves.

  (c) Several individual cylinder air start valves were determined to be sticking in the open position. If the engine crankshaft position was such that a particular combination of stuck valves also occurred, then certain cylinders would be opposing each other and

the engine would not start. This likelihood would be increased f half the cylinders were isolated, as in done during the surveillance test.

(d) During disassembly of the cylinder air start valves for troubleshooting, some of the pistons had to be removed with air pressure or excessive physical force, indicating that the pistons were binding in the valve caps.

(e) Detailed measurements of the pistons and caps were made for vendor review. These concluded that several caps were machined with the bore slightly oval-shaped and tapered by several mils. this distortion was subsequently observed in several new caps in the warehouse.

(f) Vendor direction to correct the binding was to polish each piston to provide at least 1 mil clearance with its matched cylinder cap, followed by an engine run to heat the assemblies to normal engine temperatures, followed by a pop test of each valve to ensure that it was cycling freely.

(g) This corrective action was taken on all 4 DG's.

7-13-90 DG 2A completed
7-18-90 DG 1B completed
7-21-90 DG 1A completed
7-23-90 DG 2B completed.

(h) Based on this problem, Cooper Industries issued a Part 21 report to the NRC on July __, 1990. Long term corrective actions have not been finalized but include areas such as:

(1) Factory QC check of cap dimensions.
(2) Possible material changes in the piston or cap.
(3) Expanded testing during routine overhauls.

(i) An Event Critique Team reviewed the July 11, 1990 start failure. This team concluded that the April 12, 1990 start failure should have been critiqued, but was not. Additional conclusions from the review process include:

(1) Clarification of plant vs. simulator pushbutton delay times for licensed operators.

(2) Instructions to operators to consider and log all attempted DG starts.

B. Root Cause of Events

. Manufacturing defect in individual air start valves, resulting in occasional binding in the open position.

C. Corrective Action

. Increase clearance in valves to prevent binding, and verify.

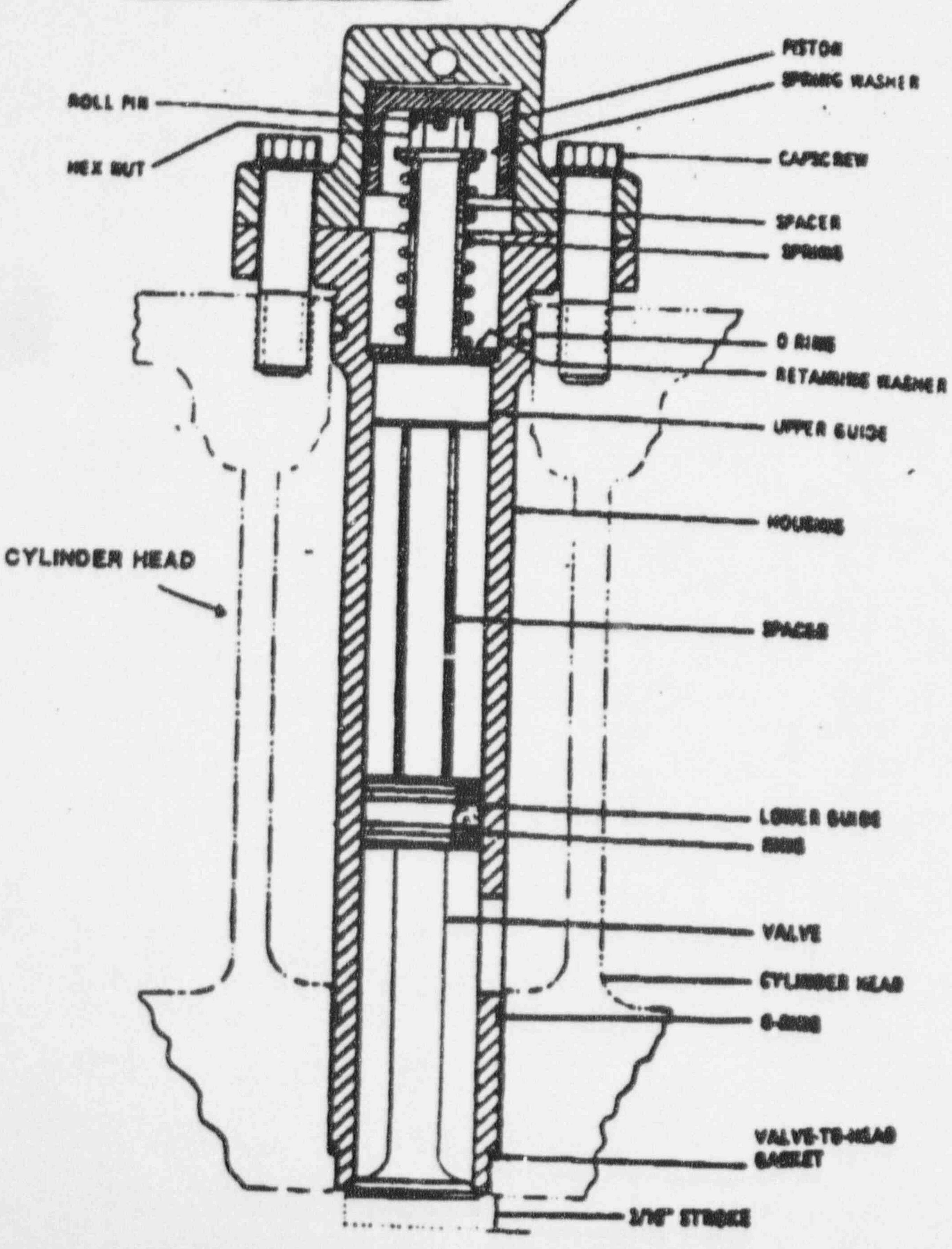
. Notify vendor for Part 21 issue.

. Long-term correction not yet complete' by vendor.

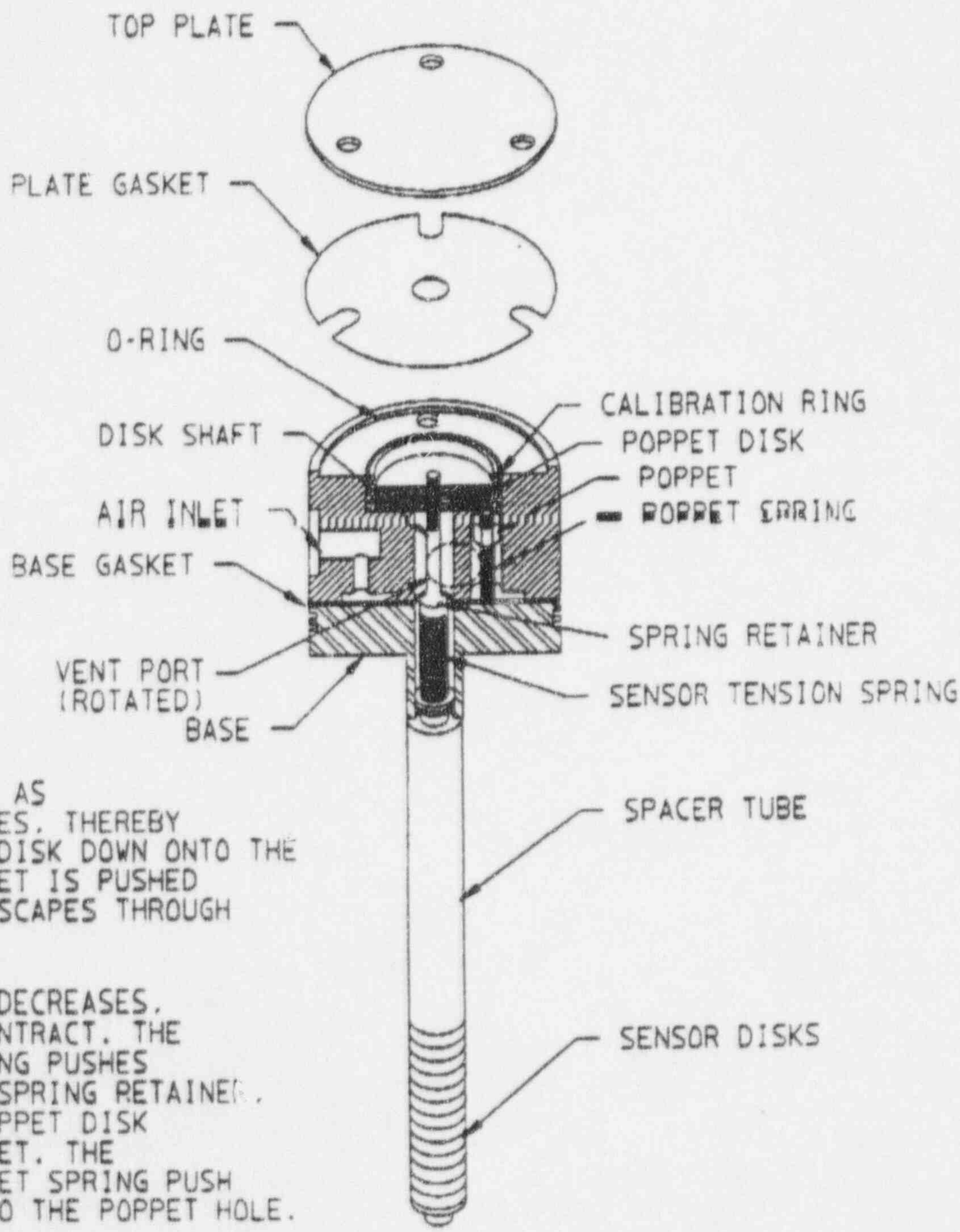
D. Significance of Problem

. DG would not start if a particular crankshaft alignment and stuck air valve combination existed. This condition was ex___ during surveillance testing when half of the cylinders were isolated from starting air.

AIR START VALVE

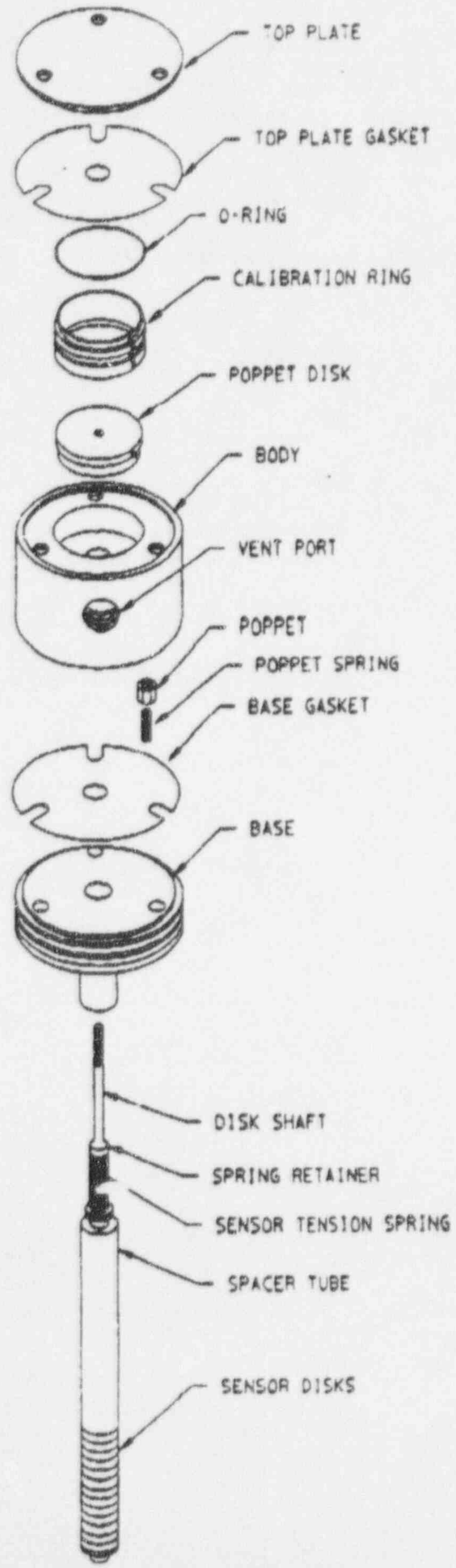


CALCON TEMPERATURE SENSOR
MODEL A3500-W3



SENSOR DISKS EXPAND AS TEMPERATURE INCREASES, THEREBY PULLING THE POPPET DISK DOWN ONTO THE POPPET. AS THE POPPET IS PUSHED DOWNWARD, THE AIR ESCAPES THROUGH THE VENT.

AS THE TEMPERATURE DECREASES, THE SENSOR DISKS CONTRACT. THE SENSOR TENSION SPRING PUSHES UPWARD AGAINST THE SPRING RETAINER, WHICH PUSHES THE POPPET DISK UPWARD OFF THE POPPET. THE AIR SUPPLY AND POPPET SPRING PUSH THE POPPET BACK INTO THE POPPET HOLE.


TOP PLATE
TOP PLATE GASKET
O-RING
CALIBRATION RING
POPPET DISK
BODY
VENT PORT
POPPET
POPPET SPRING
BASE GASKET
BASE
DISK SHAFT
SPRING RETAINER
SENSOR TENSION SPRING
SPACER TUBE
SENSOR DISKS

INSTRUMENT FUNCTION	PROBLEM	UNIT 1 CONSTRUCTION/STARTUP (8/85-12/86)	UNIT 2 CONSTRUCTION/STARTUP (1/88-12/89)	UNIT 1 1R1 (9/88-10/88)	UNIT 1 CYCLE 2 PRE-OUTAGE	UNIT 1 CYCLE 2 OUTAGE (PRE-EVENT)	POST EVENT 3/20 - 3/25/90	~~SUBSEQUENT TO 3/25/90~~
Jacket Water Temp.	Out-of-calibration	8	7	4	4		2	
Jacket Water Temp.	Defective	1		6			1	
Lube Oil Temp.	Out-of-calibration		1		1		1	
Bearing Temp.	Defective			10		1		
Miscellaneous								
Vibration Sensors	Defective		3					

SUMMARY OF DIESEL GENERATOR

PRESSURE SENSOR (CALCON) PROBLEMS AT VOGTLE

TIME FRAME

INSTRUMENT FUNCTION	PROBLEM	UNIT 1 CONSTRUCTION/STARTUP (8/85-12	UNIT 2 CONSTRUCTION/STARTUP (1/88-12/88)	UNIT 1 1R1 (9/88-10	UNIT 1 [illegible]AGE (PRE-EVENT)	POST-EVENT (3/20 - 3/25/90)	~~POST-NUREG 3/26/90 - PRESENT~~
e Oil Pressure	Out-of-calibration	4		2			~~(need to summarize subsequent #'s)~~
e Oil Pressure	Defective	1			(1)		
be Oil Pressure	Out-of-calibration	1	1				
be Oil Pressure	Defective				1		
mal Pressure (P-3)	Out-of-calibration			1			
mal Pressure (P-3)	Defective			1		1	
ket Water Pressure	Out-of-calibration	1		1	1		

Georgia Power Company
333 Piedmont Avenue
Atlanta, Georgia 30308
Telephone 404 526 3195

Mailing Address
40 Inverness Center Parkway
Post Office Box 1295
Birmingham, Alabama 35201
Telephone 205 868 5581

August 30, 1990 90 SEP 6 P2:54 the southern electric system

W. G. Hairston, III
Senior Vice President
Nuclear Operations

ELV-02059
0579

Docket No. 50-424 C

U. S. Nuclear Regulatory Commission
Region II
101 Marietta Street, N. W.
Atlanta, GA 30323
ATTN: Mr. S. D. Ebneter

Dear Mr. Ebneter:

VOGTLE ELECTRIC GENERATING PLANT
CLARIFICATION OF RESPONSE TO CONFIRMATION
OF ACTION LETTER

By letter dated April 9, 1990 (ELV-01516), Georgia Power Company (GPC) responded to a Confirmation of Action Letter dated March 23, 1990. In that letter and in our meeting notes, GPC reported that Diesel Generator (DG) 1A had been started 18 times and DG 1B had been started 19 times with no failures or problems between March 20 and April 9, 1990. Similar information was reported in Revision 0 of Licensee Event Report (LER) 50-424/1990-006 dated April 19, 1990 (ELV-01545). As reported in our telephone calls to the NRC, we subsequently discovered that this information was in error.

In Revision 1 to LER 50-424/1990-006 dated June 29, 1990 (ELV-01729), GPC attempted to clarify the correct number of DG starts occurring in this time period by using regulatory guide terminology (i.e., valid vs. successful starts). This revised LER accurately reports the number of valid DG starts during the period of March 21 through June 7, 1990. However, during a recent NRC inspection it was pointed out that the revised LER did not adequately clarify the numbers in the April 9th letter.

The confusion in the April 9th letter and the original LER appear to be the result of two factors. First, there was confusion in the distinction between a successful start and a valid test. For the purpose of this letter, a start was considered successful when the DG was started and either ran or was intentionally shut down due to testing in progress, as identified on the attached tables. Our use of the term "successful" was never intended to imply a "valid successful test" in the context of Regulatory Guide 1.108. Many start attempts were made to test the DG's 1A and 1B using applicable operating procedures. These procedures and data sheets do not contain criteria for determining if a start is successful which resulted in determinations of success which were inconsistent with the above definition. Second, an error was made by the individual who performed the count of DG starts for the NRC April 9th letter.

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GeorgiaPower

The purpose of this letter is to correct the figures related to the number of DG starts reported in the April 9th letter. Attached are tables 1 and 2 which summarize the DG starts for the period indicated. For DG 1A, there was a total of 31 start attempts and 29 of these attempts were considered successful after the two failures associated with the March 20 event. For DG 1B there was a total of 29 start attempts and 21 of these attempts were considered successful. Further, for DG 1B there were 12 successful sequential starts.

Sincerely,

C.K. McCoy FOR

W. G. Hairston, III

WGH,III/NJS/gm

Attachments

xc: <u>Georgia Power Company</u>
Mr. C. K. McCoy
Mr. G. Bockhold, Jr.
Mr. R. M. Odom
Mr. P. D. Rushton
NORMS

<u>U. S. Nuclear Regulatory Commission</u>
Document Control Desk
Mr. T. A. Reed, Licensing Project Manager, NRR
Mr. B. R. Bonser, Senior Resident Inspector, Vogtle

TABLE 1

DIESEL GENERATOR 1A

START No.	DATE	SUCCESS	RUN TIME	UNPLANNED TRIP	DISCUSSION
139	03-20-90	No	1 min	Yes	Failure to maintain load.
140	03-20-90	No	1 min	Yes	Failure to maintain load.
141	03-20-90	Yes	4 1/2hr	No	Manual start, load maintained.
142	03-20-90	Yes	45 min	No	Normal reserve auxiliary transformer swap method.
143	03-20-90	Yes	5 min	No	Observation run.
144	03-20-90	Yes	20 min	No	Observation run.
145	03-23-90	Yes	60 min	No	Observation run.
146	03-23-90	Yes	0 min	No	Started wrong diesel generator.
147	03-29-90	Yes	50 min	No	UV test start #1.
148	03-30-90	Yes	2 hr	Yes*	Bubble test #1, high temperature jacket water sensor vented.
149	03-30-90	Yes	6 min	No	Trip simulation test.
150	03-30-90	Yes	6 min	No	Trip simulation test.
151	03-30-90	Yes	3 min	No	Trip simulation test.
152	03-30-90	Yes	6 min	No	Trip simulation test.
153	03-30-90	Yes	4 min	No	Orifice modification functional test.
154	03-30-90	Yes	10 min	No	Orifice modification functional test.
155	03-31-90	Yes	2 min	No	Orifice modification functional test.
156	03-31-90	Yes	3 min	No	Orifice modification functional test.
157	03-31-90	Yes	10 min	No	Bubble test #2
158	03-31-90	Yes	1 min	No	Sensor trip timing test.
159	03-31-90	Yes	1 min	No	Sensor trip timing test.
160	03-31-90	Yes	2 min	No	Sensor trip timing test.
161	03-31-90	Yes	1 min	No	Sensor trip timing test.
162	03-31-90	Yes	75 min	No	Sensor trip timing test
163	03-31-90	Yes	27 min	No	UV test start #2.
164	04-01-90	Yes	1 1/2 hr	No	Normal surveillance test.
165	04-06-90	Yes	1 min	No	Jacket water temperature test.
166	04-06-90	Yes	1 min	No	Jacket water temperature test.
167	04-06-90	Yes	10 min	No	Jacket water temperature test.

TABLE 1 (CONTINUED)

DIESEL GENERATOR 1A

START No.	DATE	SUCCESS	RUN TIME	UNPLANNED TRIP	DISCUSSION
168	04-06-90	Yes	2 1/2 hr	No	LOSP trip modification functional test.
169	04-09-90	Yes	1 3/4 hr	No	Normal surveillance test.

* Unit tripped during bubble testing due to one sensor venting and another sensing line being disconnected for testing. This is further described in NUREG-1410.

TABLE 2

DIESEL GENERATOR 1B

START No.	DATE	SUCCESS	RUN TIME	UNPLANNED TRIP	DISCUSSION
120	03-21-90	No	0 min	No	Post-maintenance run, prime fuel lines.
121	03-21-90	No	0 min	No	Post-maintenance run, prime fuel lines.
122	03-21-90	No	15 min	No	Post-maintenance run, adjust governor.
123	03-21-90	No	2 min	No	Post-maintenance run, fuel oil delta pressure high.
124	03-21-90	No	4 min	No	Functional test run, fuel oil delta pressure high.
125	03-22-90	Yes	6 min	No	Functional test for maintenance.
126	03-22-90	Yes	1 min	No	Functional test for maintenance.
127	03-22-90	Yes	15 min	No	Post-maintenance overspeed test.
128	03-22-90	Yes	3 min	No	Post-maintenance overspeed test.
129	03-22-90	Yes	5 min	No	Post-maintenance overspeed test.
130	03-22-90	Yes	5 min	No	Voltage clamp circuit adjust.
131	03-22-90	Yes	2 min	No	Voltage clamp circuit adjust.
132	03-22-90	No	1 1/2 hr	Yes	Post-maintenance load test, high temperature lube oil trip.
133	03-23-90	Yes	7 hr	No	Post-maintenance load test
134	03-23-90	No	3 min	Yes	Post-maintenance load test, low pressure jacket water trip.
135	03-23-90	Yes	4 1/2 hr	No	Post-maintenance load test.
136	03-24-90	No*	33 min	No	Post-maintenance load test, high temperature jacket water alarm.
137	03-27-90	Yes	1 1/2 hr	No	Bubble test.
138	03-27-90	Yes	42 min	No	Trip simulation test.
139	03-27-90	Yes	3 min	No	Trip simulation test.
140	03-27-90	Yes	2 min	No	Trip simulation test.
141	03-27-90	Yes	6 min	No	Trip simulation test.
142	03-27-90	Yes	57 min	No	UV test.
143	03-28-90	Yes	1 3/4 hr	No	Normal surveillance
144	03-28-90	Yes	4 min	No	Low pressure lube oil modification functional test.
145	03-28-90	Yes	4 min	No	Low pressure lube oil modification functional test.
146	04-04-90	Yes	1 1/4 hr	No	Post-maintenance load test.
147	04-05-90	Yes	5 min	No	LOSP trip modification functional test.
148	04-05-90	Yes	2 hr	No	Normal surveillance.

* High temperature jacket water trip alarm was received and the engine kept running.

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* ADMITTED IN PA
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* ADMITTED IN DC
* ADMITTED IN MA

RECEIVED
SEP 11 1990
Office of the Secretary

September 11, 1990

Hon. Kenneth M. Carr, Chairman
United States Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Chairman:

Please find enclosed the original and one copy of Petitioners Marvin B. Hobby's and Allen L. Mosbaugh's request for proceedings and imposition of civil penalties. The exhibits identified in the petition will be forwarded under seperate cover.

Petitioners stand ready to assist the Commission in any way they can. Please do not hesitate to contact me in this regard.

Very truly yours,

Michael D. Kohn

Counsel to Marvin B. Hobby
and Allen L. Mosbaugh

cc: A.W. Dahlberg (petition and cover letter)

A1133
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Release

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UNITED STATES OF AMERICA
BEFORE THE NUCLEAR REGULATORY COMMISSION

In the Matter of,	)	Docket Nos. ________
GEORGIA POWER CO.,	)	

REQUEST FOR PROCEEDINGS AND IMPOSITION OF CIVIL PENALTIES FOR IMPROPERLY TRANSFERRING CONTROL OF GEORGIA POWER COMPANY'S LICENSES TO THE SONOPCO PROJECT AND FOR THE UNSAFE AND IMPROPER OPERATION OF GEORGIA POWER COMPANY LICENSED FACILITIES

To: Kenneth Carr, Chairman
U.S. Nuclear Regulatory Commission

I. Relief Sought

Petitioners Marvin B. Hobby and Allen L. Mosbaugh, hereby request that the Commission, sua sponte, pursuant to 10 CFR 2.104(C) institute licensing proceedings to determine whether Georgia Power Company ("GPC"):

i) Illegally transferred control of its licenses to SONOPCO Project (hereinafter SONOPCO) and the Southern Company System in violation of 10 C.F.R. 50.80(C);

ii) Can reasonably assure that SONOPCO or other entities operating GPC's licensed nuclear facilities are complying with NRC regulations;

iii) Can reasonably assure that SONOPCO's or other entities' operation of GPC's licensed nuclear facilities is not endangering the health and safety of the public;

iv) Has the character, competence, fundamental trustworthiness and commitment to safety to operate a nuclear facility.

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The need for immediate and swift action by the Commission, in light of the seriousness of the allegations contained herein, cannot be overstated.

II. Background of Petitioner

a) Marvin B. Hobby.

Petitioner Marvin B. Hobby, has devoted the last 22 years to promoting the safe and reliable use of nuclear power. In 1980, the Institute of Nuclear Power Operations ("INPO") was created and Admiral Dennis Wilkinson was named as its first President. In April of 1980, Admiral Wilkinson selected Mr. Hobby to serve as INPO's Communications Manager. He later became Assistant to the President. In 1983 Mr. Hobby became INPO's corporate Secretary as well as continuing in his role as Assistant to the President.

In 1984, Mr. Hobby accepted a position with the Nuclear Utilities Management and Resources Committee ("NUMARC") and served as the Project Manager, Congressional Education.

In June of 1985, Mr. Hobby accepted an offer of employment from GPC's then President Mr. J.H. Miller, Jr., to serve as his assistant. In 1986, in addition to being Assistant to the President, Mr. Hobby was also named as Assistant to GPC's Senior Executive Vice President. In 1987 Mr. Hobby was named GPC's Manager of Nuclear Support. The following year he served as GPC's Manager of Nuclear Support Services. In December of 1988 he was then named as GPC's General Manager of Nuclear Operations Contract Administration and Assistant to the Senior Vice President.

Mr. Hobby served as General Manager of Nuclear Operations Contract Administration until April of 1990, at which time he was forced from the company after attempting to bring to GPC management's attention that it had improperly transfered control of its nuclear licenses to SONOPCO and the Southern Company. In this regard, on April 27, 1989, Mr. Hobby wrote a highly confidential memorandum (co-signed by GPC's then Senior Vice President George F. Head). A redacted copy of this memo is attached as Exhibit A. This memo alerted GPC management to the fact that it appeared that GPC was violating its licenses by improperly transferring control of its nuclear facilities. On that day, and the following day, April 28, 1989, a GPC vice president, Fred R. Williams, instructed Mr. Hobby to destroy all copies of his April 27, 1989 memo. Mr Hobby, concerned about his potential liability, sought outside advice. On June 8, 1989, Mr. Hobby wrote to Admiral Wilkinson to explain the concern he had regarding his perception that GPC improperly transferred control of its nuclear facilities to the Southern Company and SONOPCO as well as GPC's reaction to his raising the concern to management. A redacted copy of Mr. Hobby's letter to Admiral Wilkinson is attached hereto as Exhibit "B". Before Mr. Hobby could get GPC to resolve his concern over the improper transfer of GPC's license to Southern Company and SONOPCO, he was forced from GPC. Inasmuch as Mr. Hobby was unable to resolve his concern internally, as he faithfully tried to do. he is now forced to petition the Commission directly.

b) Allen L. Mosbaugh.

Petitioner Allen L. Mosbaugh has devoted the last 16 years to the safe start up, operations, and testing of commercial nuclear power reactors. Mr. Mosbaugh has been trained and certified as a Senior Reactor Operator (SRO) and Shift Technical Advisor (STA), and holds undergraduate and graduate degrees in Nuclear Engineering.

Between 1986 and May of 1990, Mr. Mosbaugh served as GPC's Vogtle project Assistant Plant Support Manager. He was in charge of a staff of over 400 people in the areas of technical support, engineering support, security, administration, training, and quality concerns. Mr. Mosbaugh, until May of 1990, was the Vice-Chairman of the Plant Review Board ("PRB"), one of four Plant Duty Managers, as well as a Vogtle project Emergency Director. Mr. Mosbaugh was removed from the PRB by Plant General Manager George Bockhold after he attempted to resolve safety issues with the PRB.

Between 1984 and 1986, Mr. Mosbaugh served as GPC's Pre-operational Superintendent and Superintendent of Engineering Services for the Vogtle project. In this capacity he was responsible for start up and pre-operational testing of Unit 1. He assembled and managed 150 engineers and additional support and clerical staff.

By April of 1990, Mr. Mosbaugh came to the conclusion that the highest levels of the SONOPCO project, including R. Patrick McDonald and Joseph M. Farley, were in control of the operation of GPC's Vogtle project. Mr. Mosbaugh concluded that SONOPCO was needlessly endangering the

public's health and safety by: refusing to adhere to technical specifications in the interest of schedule; carelessly disregarding for reactor criticality safety; operating radioactive waste systems as to be in gross violation of NRC requirements; refusing to report adverse events and conditions to the NRC as required by regulations; submitting false information to the NRC; repeatedly allowing the Vogtle project to enter Technical Specification 3.0.3 "motherhood" conditions without notifying the NRC or correcting the adverse condition within the required time span; adopting a policy of intentionally "taking" Licensee Event Reports ("LER's") to keep the Vogtle plant on line and on schedule during planned shut downs; and rewarding managers for engaging in non-conservative and questionable compliance practices.

III. Facts

Petitioners submit the following information in support of their request for proceedings.

1. Illegal Transfer of Licenses to SONOPCO.

GPC improperly transferred control of its nuclear operating licenses to The Southern Company and to SONOPCO without first obtaining permission from the Commission to do so pursuant to 10 CFR 50.80 ("No license...shall be transferred, assigned, on in any manner disposed of, either voluntarily or individually, directly or indirectly, through transfer of control of the license to any person, unless the Commission shall give its consent in writing"). Evidence

that GPC transferred control of its licenses to SONOPCO was obtained as a result of Petitioners' witnessing the day-to-day operation of GPC's nuclear facilities both at the site (by Mr. Mosbaugh) and at GPC's corporate offices (by Mr. Hobby).

Although Mr. Farley asserts under oath that he is not a Corporate Officer of GPC (rather he is an officer of Southern Company Services and The Southern Company), he is the SONOPCO Chief Executive Officer ("CEO"). Indeed, GPC's Senior Executive Vice President through May of 1990, H. Grady Baker, Jr., acknowledges that Mr. Farley is SONOPCO's _de facto_ CEO: "The appropriate oversight of SONOPCO exists, in that the Chief Operating Officer, Pat McDonald amd tje CEO or not the CEO because its not a corporation -- but Farley and McDonald are officers of Georgia Power Company, reporting to the president, Bill Dahlberg." _See_, Baker Deposition Transcript at pp. 16-17 (excerpt attached as Exhibit "C"). As the above quotation also demonstrates, Mr. Baker was also led to believe that Mr. Farley was an officer of GPC reporting to GPC's President. _Id_. Yet Mr. Farley denies that he is an officer of GPC. _See_ Farley Deposition at p. 10 (excerpt attached as Exhibit D).

A thorough review of SONOPCO's operation will demonstrate that SONOPCO's CEO is Mr. Farley, not Mr. Dahlberg. The actual chain of command is General Plant Manager George Bockhold to SONOPCO Vice President McCoy; McCoy to SONOPCO's Senior Vice President, George Hairston, Hairston to SONOPCO's Executive Vice President and Chief Operations Officer, R. Patrick McDonald; McDonald to

SONOPCO's Chief Executive Officer, Mr. Farley.

In an April 27, 1989 memo, Mr. Hobby advised GPC Vice President Fred Williams in writing that in the course of attempting to perform his function as General Manager, Nuclear Contract Administration, he observed that Messrs. Farley and McDonald -- not Mr. Dahlberg -- were in control of and were operating GPC's nuclear facilities. Mr. Hobby was instructed to destroy all copies of the memorandum. Thus, since April of 1989, GPC was advised in a confidential memorandum that in the opinion of its contract Administration Group, GPC had illegally transferred control of its nuclear licenses to SONOPCO; and SONOPCO's CEO, Mr. Farley, and Chief Operating officer, Mr. McDonald, were in control of GPC's nuclear licenses in violation of NRC regulations.

2. GPC misled the Commission about the chain of command from the Vogtle project's Plant Manager to its CEO.

GPC's Executive Vice President of Nuclear Operations, R. Patrick McDonald, knowingly made false statements to the NRC commissioners in the presence of GPC's President, A.W. Dahlberg; Vice President for Nuclear Generation, C. Ken McCoy; and General Manager of the Vogtle Project, George Bockhold. Yet no one from GPC attending the meeting with the Commission corrected the false statement made by Mr. McDonald to the Commission.

On March 30, 1989, during the course of a transcribed proceeding held before the Commission, Mr. McDonald was asked by then Commissioner (now Chairman) Carr to state the

"hierarchy between the CEO and the plant manager" of the Vogtle project so Mr. Carr could evaluate his "management concern" he had that the plant manager, Mr. Bockhold, being "a long way from the CEO." Mr. McDonald misled Commissioner Carr when he eliminated one entire level of management between the plant manager and the CEO. The transcript of the proceeding demonstrates that Mr. McDonald stated that the reporting chain was General Manager, George Bockhold, to Vice President for Nuclear Generation, Ken McCoy, who reported to Executive Vice President of Nuclear Operations, R. Patrick McDonald, who in turn reported directly the the CEO, GPC's President, A. William Dahlberg. A copy of the relevant transcript pages is attached hereto as Exhibit E. In reality, Mr. McCoy did not report to Mr. McDonald but rather to SONOPCO's Senior Vice President, Mr. George Hairston, who then reported to Mr. McDonald. While it may be conceivable that Mr. McDonald may have suffered from a lapse of memory, it is inconceivable that Messrs. Dahlberg, McCoy and Bockhold suffered the same lapse of memory at the exact same time. Messrs. Dahlberg, McCoy and Bockhold should have known that Mr. McDonald's statements were false and should have brought this to the immediate attention of the Commission and otherwise corrected the record <u>before</u> the Commission acted on the Vogtle full power license request. Although GPC eventually corrected a portion of the false statement, it was not corrected until <u>after</u> the full power license was granted. Moreover, the correction did not address the fact that Mr. McDonald was and continued to

reports to SONOPCO's *de facto* CEO, Mr. Joseph Farley, rather than to GPC's President. *See* 1. *infra*.

3. <u>SONOPCO intentionally misled the NRC about the condition of the Vogtle Plant after a Site Area Emergency in order to hasten the restart of the reactor</u>.

As the Commission is well aware, a near disaster befell the nuclear industry when, in March of 1990, the Vogtle nuclear station had a total loss of electrical power while the Reactor Coolant System was at "midloop" and without containment integrity. Following the station blackout, SONOPCO submitted a Confirmation of Action Response letter (COAR) and a follow-up Licensee Event Report (LER), No. 90-006. Both the COAR and LER contained known false statements intended to mislead the NRC with false assurances about the reliability of the diesel generator whose failure resulted in the Site Area Emergency. The NRC was advised in the COAR and LER that the back-up diesel generator that failed to start and caused the blackout had been returned to a safe operating and reliable condition. SONOPCO alleged in the COAR and LER that the diesel was reliable because it had been successfully started multiple times without suffering a failure, trip or problems. But, as SONOPCO knew, the diesel generator had actually continued to experience an excessive number of trips, failures and problems similar in nature to the failure which led to the March 20, 1990 station blackout. Indeed, on April 10, 1990, Mr. Mosbaugh wrote a memo to the Vogtle General Manager, Mr. Bockhold, and informed him that the diesel air quality statements made in the COAR were false. On April 19, 1990 Mr. Mosbaugh had

informed SONOPCO's Senior Vice President, Mr. George Hairston, that the diesel had suffered trips and failures. Nonetheless, later that same day SONOPCO's Senior Vice President, George Hairston, signed LER 90-006-00, after he was advised that the information stated therein contained false information. On April 30, 1990 Mr. Mosbaugh submitted a memorandum to Mr Bockhold stating diesel start data contained in the LER was incorrect. On May 10, 1990, Mr. Mosbaugh, acting as the Chairman of the Vogtle Plant Review Board ("PRB"), issued an action item to Mr. Bockhold requiring the resolution of the incorrect statements contained in the COAR. The following day, May 11, 1990, Mr. Mosbaugh was removed from the PRB.

An independent investigation into this matter will demonstrate that:

a) The statements made in the COA and LER were used by the NRC to make decisions "significant to the regulatory process."

b) The LER wording is false because it overstated the reliability of the diesel and did not count numerous failures and problems of the diesel when GPC attempted to start it up.

c) Concern over the accuracy of the data contained in the LER was raised by Mr. Mosbaugh before the LER was submitted.

d) SONOPCO personnel recognized that the COAR statements were false before submitting the LER;

e) Petitioner Mosbaugh submitted detailed factual

information to CPC after the LER was submitted which conclusively demonstrated that the LER contained false information;

f) SONOPCO intentionally delayed revising the LER until after critical meetings with the NRC and Commission were held on June 8, 1990 (ITT presentation to Commissioners);

g) After the ITT Presentation to the NRC Commissioners, SONOPCO further delayed correcting the LER even though QA had already substantiated the inaccuracies contained in the LER.

h) SONOPCO proceeded with actions to submit the revised LER only after Petitioner Mosbaugh continued to ask "why the revised LER has yet to be submitted" to the NRC (Mr. Mosbaugh raised this question in a meeting attended by the NRC and the General Manager);

i) On June 28, 1990 and June 29, 1990, SONOPCO drafted at least six cover letters to be submitted with the revised LER. Each of these six cover letters states false explanations and were concocted after the fact without regard to the truth. The multiple drafts of the cover letters demonstrate that SONOPCO does not intend to advise the NRC why the errors in the LER were actually made. To wit, the different explanations stated in five of the cover letters are as follows:

DATE/TIME		EXPLANATION CONTAINED IN DRAFT
6-28-90	07:51	All tests of diesel were counted but only valid failures were considered in the conclusion that no problems or failures occurred
6-28-90	08:55	All tests were counted regardless of whether they were valid or not.
6-29-90	07:55	This draft asserts that the LER really meant to read "subsequent to the event" but was inadvertently worded "subsequent to the test program."
6-29-90	11:42	This draft states that the LER did not consider failures and problems associated with the troubleshooting and restarting the diesel.
6-29-90	13:11	This draft states that the error occurred due to poor record keeping practices and no definition of the end of the test program;

j) A review of the performance records of the diesel generator will demonstrate that it was unreliable and that the statements provided to the NRC were false and intentionally misleading. Indeed, the diesel generator was so unreliable after the site area emergency that GPC was eventually forced to initiate three different design changes to remove or modify numerous unreliable components from the design and control logic when the diesel experienced additional failures after the LER was submitted.

k) The unreliability of hte components that caused the diesel to fail to perform its intended safety function when actually called upon to work was known to be unreliable for years and remained

uncorrected by the licensee;

1) SONOPCO retaliated against Allen Mosbaugh by removing him from the PRB after submitting memorandum to George Bockhold demonstrating that his personal presentation to the NRC contained incorrect information and that the LER and COAR letters contained false information.

Thus, the diesel was not as reliable as the COAR and LER conveyed to the NRC. As such, SONOPCO provided false and misleading information to the NRC about the actual reliability of the diesel generator and the actual failure rate of the generator. Thereafter, a cover-up of the reliability of the diesel followed and Mr. Mosbaugh's attempts to correct the false statements contained in the LER and COAR resulted in his removal form the PRB.

By misleading the NRC about the safe operating condition of the diesel generator, SONOPCO demonstated a complete lack of concern for the safe operation of the Vogtle facility.

4. <u>GPC's Executive Vice President submitted perjured testimony during the Course of a proceeding commenced under Section 210 of the Energy Reorganization Act</u>.

GPC's Executive Vice President, R. Patrick McDonald, knowingly submitted false testimony during proceedings commenced pursuant to Section 210 of the Energy Reorganization Act. In the <u>Fuchko & Yunker v. Georgia Power Company</u> Section 210 proceedings, Mr. McDonald stated under oath at the hearing and in a previous deposition that the staff of the newly formed SONOPCO project was chosen

"from the top down" (i.e. management picked the Vice Presidents, the Vice Presidents picked the General Managers, then the General Managers picked their own Managers, then the Managers picked their Supervisors, etc.). See Deposition transcript at pp. 50, 62, and January 4, 1989 Hearing Testimony at p. 429 (both of which are attached hereto as Exhibit F).

Moreover, GPC's counsel was advised that Mr. McDonald's testimony was false prior to that counsel's calling Mr. McDonald as a witness at the Yunker/Fuchko hearings on January 4, 1989. In this regard, on January 2, 1989 (two days before Mr. McDonald was to testify), a meeting of GPC's witnesses was scheduled by GPC's counsel, the Troutman, Sanders, Lockerman and Ashmore law firm. As Mr. Hobby was one of the witnesses GPC planned to call at the hearing, he was asked to attend and did, in fact, attend this meeting. At this meeting, Mr. McDonald addressed the group and stated how the SONOPCO staff was selected. Mr. McDonald stated that the "top down" approach was used to staff the SONOPCO project. Near the end of the meeting, Mr. Hobby informed GPC's counsel that Mr. McDonald's statement regarding how the SONOPCO staff was chosen "from the top down" was false. Mr. Hobby also advised GPC's counsel that other statements made by Mr. McDonald were also false. GPC's counsel responded to Mr. Hobby's statement that he believed Mr. McDonald's statements to be false by advising Mr. Hobby that he would have to change his testimony. When Mr. Hobby was instructed to change his testimony, he refused to do so and

advised GPC counsel that he would not cooperate in their attempt to submit perjurious testimony during the course of the *Yunker* and *Fuchko* proceedings.

The following day, on January 3, 1989, Mr. Hobby advised Mr. Thomas McHenry of the false statements Mr. McDonald made at the January 2, 1989 meeting. Mr. McHenry confirmed that if the statement about the "top down" approach of filling SONOPCO staff positions was made by Mr. McDonald, then Mr. McDonald would not be telling the truth. Mr. McHenry advised Mr. Hobby that he had 1st hand knowledge that the assertion was false. Mr. Hobby further advised Mr. McHenry that he was instructed to change his testimony to coincide with Mr. McDonald's but he refused to do so. *See* Affidavit of Thomas McHenry, attached as Exhibit "G". Indeed, the false "top down" statement was made by Mr. McDonald under oath on December 23, 1988 and January 4, 1989, during the course of the section 210 proceedings. *See* Exhibit "H". More troubling is the fact that *prior* to allowing Mr. McDonald to take the witness stand at the *Yunker/Fuchko* proceedings, GPC's counsel had arranged, during confidential settlement discussions, that Mr. Yunker's and Fuchko's counsel would not subject Mr. McDonald to vigorous cross-examination when he testified thereby assuring that 1) his perjurous testimony would not be challenged and, 2) it could be used in subsequent proceedings before the NRC.

The false testimony Mr. McDonald gave prior to and during the *Yunker* and *Fuchko* hearing was of a critical nature. Messrs. Fuchko and Yunker alleged that they were

prohibited from transferring out of GPC's nuclear security department and into SONOPCO's organization because they had raised valid safety concerns about GPC's improper handling of safeguards materials. In an attempt to demonstrate that Messrs. Yunker and Fuchko were not improperly kept out of SONOPCO, GPC alleged, through the testimony of Mr. McDonald, that SONOPCO positions could not be filled until a manager of the security department was chosen because of the "top down" method routinely employed to fill all positions at SONOPCO. See December 23, 1988 Deposition Transcript of R. Patrick McDonald at p. 50, 62 (attached as Exhibit I).

5. <u>SONOPCO routinely threatens the safe operation of GPC's nuclear facilities by allowing them to enter "motherhood."</u>

SONOPCO repeatedly allowed the Vogtle Plant to violate Technical Specification 3.0.3 and enter "motherhood" without correcting the situation or notifying the NRC. Technical Specification 3.0.3 is the "last eschelon of defense" in assuring that sufficient redundancy and margins of safety are maintained for safe plant operation. To wit, under Technical Specification 3.0.3, a plant shut down and NRC notification are required within one (1) hour. The following are some examples where Plant Vogtle entered "motherhood" without management appropriately correcting the situation or notifying the NRC:

a) Both Unit 1's and Unit 2's A and B train safety related load sequencers have been inoperable due to failure, downpowering and other conditions rendering them inoperable on numerous occasions.

Licensed operators were not knowledgeable that the loss of this sequencer resulted in the plant entering "motherhood" and as such the NRC was never notified of the condition pursuant to Technical Specification 3.0.3.

6. <u>SONOPCO routinely endangers the public's safety by ignoring technical specifications</u>.

SONOPCO has made a conscious decision to endanger the public's safety by subverting technical specifications so as to be able to keep the Vogtle Plant operating and/or to speed up the restart of the Vogtle facility. Illustrative examples of the willful and repeated technical specification violations as follows:

a) The licensee willfully and knowingly violated Vogtle Unit 1 Technical Specifications by opening "Dilution Valves" required locked closed by technical specifications. The valves were opened while the Reactor Coolant System was at "mid-loop," thus placing the plant in an unanalyzed condition and risking an uncontrolled "dilution accident" and "inadvertent reactor criticality." The valves were opened to speed the outage so the plant could be placed back on line according to the outage schedule. Breaching technical specifications to stay on schedule is undoubtedly due, in part, to SONOPCO's philosophy -- attributed to Messrs. Farley, McDonald, Hairston and three SONOPCO Vice Presidents (it was <u>not</u>

attributed to Mr. Dahlberg which further demonstrates that Mr. Farley and not Mr. McDonald controls GPC's operating licenses) -- that outages must be scheduled assuming that:

> "...everything goes right. Everything falls into place right. That you do not put any contingency or extra time in there..." (quotation verbatum from Vice President McCoy).

The pressure to keep on schedule will necessarily result in managers intentionally breaching technical specifications and "taking" LERs in order to remain on schedule.

b) On February 26, 1990, the NRC found that the same dilution valves identified in 6(a) above were again unsecured while at "mid loop" in violation of Technical Specifications. Vogtle senior management willfully violated Technical Specifications again by not "immediately" securing the valves as required by technical specifications because management was too busy due to the outage schedule. Instead, they argued for five hours that locking the valvues was not required and that a paper "hold" tag would be sufficient.

c) On January 20, 1989 procedural errors made by two shifts of licensed operators miscalculated the shutdown margin for Vogtle Unit 1 which was shutdown at the time. The RCS boron concentration was dangerously low at 1396 ppm and xenon was decaying rapidly (zero percent shutdown margin was 1420 ppm boron). There were no plans to alter RCS

boron concentrations. By pure luck, a reactor engineer came to the control room and felt uneasy with the low boron concentration. He recalculated the shutdown margin revealing an error of 3.6%. Immediate boration was ordered to averting an inadvertent criticality. Senior plant management clearly realized the gravity of the event by responding in a private meeting that this event could have caused not only a shut down of Unit 1 but also that it could have interfered with the licensing of Unit 2. Moreover, after these events were brought to Mr. Bockhold's attention, no deficiency was writen, no critique ever conducted, no reivew to assure Technical Specifications were not violated was conducted and no report to the NRC was made.

d) On March 22, 1990, GPC employees were told to keep planned shutdowns on schedule by "taking" LER's (i.e. create an LER situation in order to keep the plant running). This practice is indicitive of the philosophy employed by SONOPCO's CEO, Mr. Farley, and COO, Mr. McDonald. <u>See</u> 6(a) above for the "philosophy" being employed on site.

e) Other lesser examples include:

   i) The Licensee knowingly concealed a technical violation which if recognized would have resulted in a safety-related shut down of Vogtle's Unit 1. This technical violation

concerned the failure to properly test approximately 39 containment isolation valves in violation of technical specification surveillance 4.6.1.1.a. Because the surveillance tests had not been performed, the valves were to be considered inoperable. The licensee had one hour to assure the valves were in a "safe" condition. The surveillance tests were not completed for two hours, thereby requiring the shut down of Unit 1 for one hour. Instead, SONOPCO subverted the safety procedure by performing the surveillance tests without initiating a Limiting Condition of Operation, (LCO). Had the LCO been initiated, SONOPCO would also have been required to submit to the NRC a Notification of Unusual Event, (NUE), causing further embarrassment since Unit 1 had to report a NUE for the same reasons on February 23, 1990. __See__ D.C. 2-90-0022;

ii) SONOPCO knowingly concealed another technical violation on March 1, 1990 when a mode 5 to mode 6 change occurred even though required equipment was not operable. The failure to comply with the technical specification translated to a 12 hour schedule enhancement at a critical juncture;

iii) On March 5, 1990, SONOPCO knowingly concealed another technical violation when "B

train" RHR pump vibrations resulted in the cracking of a NSCW water cooling line. With the pump vibrating severely and with a failed cooling line, the pump should have been declared inoperable. At the same time the "A train" RHR pump was drained due to outage-related work. Under Technical Specification 3.9.8.1, both trains were not operational. A LCO and action statement for this condition should have been entered. Had an LCO and action statement been entered, certain "actions or operations" would have had to be suspended. Instead, the pump was not declared inoperable and the LCO was not entered.

7. <u>SONOPCO repeatedly concealed safeguards problems from the NRC</u>.

a) SONOPCO personnel (including a SONOPCO Vice President and SONOPCO General Manager, and a Southern Company Services Manager) knowingly and repeatedly hid safeguards problems from the NRC and willfully refused to comply with mandatory reporting requirements. Moreover the SONOPCO Vice President made false statements to the NRC during an Enforcement Conference about the status of safeguards materials in Birmingham, Alabama. The false and misleading information presented at the Enforcement Conference and other information withheld from the NRC are highly significant to

the regulatory process and were relied upon as a basis for NRC decisions, which had the NRC had the benefit of complete, factual information, the NRC would have, most probably, increased the civil penalties from the minimum $50,000 into the hundreds of thousands of dollars (i.e. 100% increase in the base penalty due to past performance, 100% increase in the base due to multiple events, 50% increase for failure to report, a 50% increase for no prompt corrective action, and an increase due to "willfulness").

b) On July 23, 1990, Plant and SONOPCO senior management prevented the Site Security Manager from making a Red Phone notification within one hour as required by 10 C.F.R. 73.71. The manager was so prevented hoping to delay or defuse NRC knowledge of programatic problems within SONOPCO (and its design agencies which include Southern Company Services) with safeguard documents.

8. <u>SONOPCO has endangered the public's health and safety by operating radioactive waste systems and facilities known to be in gross violation of NRC requirements</u>.

In early 1988, Plant Vogtle's radioactive waste filter system was installed and operated at Plant Vogtle in gross violation of Regulatory Guide 1.143. Although the system was shutdown by Quality Assurance due to programatic breakdowns in procurment and design, in February of 1990, SONOPCO approved its resumption eventhough the violations

observed by Quality Assurance had not been remedied. When the Plant Review Board attempted to consider whether a the system shold be resumed, Vogtle's General Manager, George Bockhold, intimidated members of the PRB. The end result was that the system was returned to service eventhough 10 C.F.R. 50.59 safety evaluations and accident analysis are inadequate and/or incorrect. As such, a spray of radioactive leakage from rubber hoses and plastic pipe used in this makeshift system can flow unrestricted into storm drains which would result in its discharge into Beaver Creek.

9. <u>SONOPCO management routinely risks the safe operation of GPC's Nuclear facilities through non-conservitive and questionable management practices</u>.

SONOPCO encourages non-conservative and questionable compliance practices by:

a) Praising managers for taking risks.

b) Not taking any adverse action against managers or employees who engage in non-conservative and questionable compliance practices.

c) Refusing to critically investigate events or practices resulting in LERs.

d) Retaliating against managers who make their regulatory concerns known to GPC and/or SONOPCO management.

WHEREFORE, Petitioners request that this Honorable Commission, <u>sue sponte</u>, institute proceedings to consider:

a) whether GPC illegally transfered control of its licenses to SONOPCO and the Southern Company System in violation of 10 C.F.R. 50.80(C);

b) whether there is reasonable assurance that SONOPCO's or other entities' operating GPC's licensed nuclear facilities are complying with NRC regulations;

c) whether there is reasonable assurance that SONOPCO or other entities operation of GPC's licensed nuclear facilities is endangering the health and safety of the public; and

d) whether the licensee has the character, competence, fundamental trustworthiness and commitment to safety to operate a nuclear facility.

Respectfully submitted,

Michael D. Kohn
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Counsel to Marvin B. Hobby
and Allen L. Mosbaugh

Dated: September 11, 1990
65/NRCGP

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Release

DW Sent 9/13/90

DECLARATION OF LARRY L. ROBINSON

I, Larry L. Robinson, do hereby declare that the following is true and correct, under penalty of perjury, to the best of my ability.

1. My name is Larry L. Robinson. I am employed as an Investigator with the Office of Investigations, United States Nuclear Regulatory Commission. My duties include the conduct of investigations of licensees, applicants, their contractors or vendors, including the investigation of all allegations of wrongdoing by other than NRC employees and contractors.

2. I make these statements based upon my own personal knowledge, or upon knowledge obtained by me during the course of my employment, and is relied upon by me in the performance of my official duties.

3. The Office of Investigations (OI), Region II (RII), NRC, currently has two pending investigations regarding allegations of intentional wrongdoing on the part of Georgia Power Company (GPC) Managers at the Vogtle Electric Generating Plant (VEGP). These investigations basically involve allegations of deliberate Violations of Technical Specifications, and Material False Statements. If these allegations are substantiated, they could constitute violations of NRC regulations enacted to protect the public health and safety. In addition, a recent Special Inspection, conducted by NRC at VEGP during the period August 6-17, 1990, addressed additional related allegations of wrongdoing by GPC Management at VEGP that will, in all likelihood, be referred to OI in the near future.

4. On September 12, 1990, Stephen Kohn, of the Law Offices of Kohn, Kohn, and Colapinto, telephoned me and advised me that their client, Allen L. Mosbaugh, a GPC employee at VEGP, was in possession of audio tape recordings that he, Mosbaugh, had made of conversations with VEGP Managers that may be pertinent to the ongoing NRC investigations/Special Inspectic... Kohn advised me that Mosbaugh had been officially ordered to turn these tapes over to the Law Offices of Troutman, Sanders, Lockerman, and Ashmore, representatives of GPC in a Department of Labor (DOL) Case, No. 90-ERA-58, initiated by Mosbaugh. Kohn stated that his understanding was that Mosbaugh was going to have to turn over these tapes on Sept. 13, 1990.

5. Allen L. Mosbaugh had been interviewed by me on February 8, 1990, during the course of my investigation of one of the aforementioned allegations.

6. On September 12, 1990, I telephoned Mosbaugh, and he verified that he did make such tape recordings, that he was in possession of them, that he had been ordered by a DOL Administrative Law Judge to turn them over to the Troutman, Sanders Law Firm. Mosbaugh told me that, in his opinion, some of these tapes show evidence of intentional wrongdoing on the part of GPC Management at VEGP, and GPC Management at the offices of SONOPCO Project, Birmingham Alabama, in connection with the allegations in the ongoing OI investigations and the Special Inspection.

7. Also on September 12, 1990, in response to my message, Michael Kohn, also with the Law Firm of Kohn, Kohn, and Colapinto, telephoned me at my residence and advised me that his client, Mosbaugh, per an order from DOL Administrative Law Judge Bernard J. Gilday, Jr., was required to turn over the tapes to the

GPC attorneys by midnight, September 13, 1990. Michael Kohn said that he had not been able to personally review all the tapes, but that it was his understanding from conversations with his client that there was evidence of wrongdoing on the part of GPC Management, pertaining to the ongoing investigation/Special Inspection issues, contained in the conversations on the tapes. Kohn stated that his client would be willing to turn the tapes over to NRC for review for evidentiary purposes. Kohn stated that he would prefer to have the NRC subpoena the tapes.

8. The Office of Investigations has reasonable cause to believe that these tapes contain direct evidence of intentional violations of regulatory requirements by GPC personnel that pertains to ongoing NRC investigations/inspections.

9. OI has reasonable cause to believe that the review of these tapes by GPC personnel, or their representatives, prior to the completion of the aforementioned investigations, would severely compromise the integrity of these investigations.

Further, declarant sayeth naught.

Dated this 13th day of September, 1990 at Atlanta, Georgia.

Larry L. Robinson

Larry L. Robinson

SEP 13 '90 07:25AM ALLEN L MOSBAUGH P.1

att 015

07:27 9-13-90

Fax to: Larry Robinson

NRC Office of Investigations

Please hand deliver Immediately

119

Rm. 265

September 14, 1990

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2. Documents Collected and Provided by Augmented Inspection Team (AIT)

2-1 Instruction Manual For S & C Circuit Switches (Outdoor 230kv)

2-2 Instruction Manual For Westinghouse Type Co Overcurrent Relays

2-3 Instruction Manual For ASEA Type RADSE Transformer Differential Protective Relays

2-4 Instruction Manual For Brown Boveri Type AOT, AOK & AOS High Voltage Current Transformer

2-5 Instruction Manual For General Electric Type PVD Differential Voltage Relay

*2-6(124) Nuclear Plant Maintenance Work Order No. 28900466

*2-7 (25[illegible]) Bechtel Drawings [illegible]X3D-AA-A01A (Main One Line - Unit 1)

2-8 Bechtel Drawings 2X3D-AA-A01A (Main One Line - Unit 2)

2-9 Bechtel Drawings AX3D-AA-A01A (Main One Line - Common Units 1 & 2)

2-10 Bechtel Drawings AX3DL060 (Switchyard General Arrangement)

2-11 Bechtel Drawings 1X3DH7A1 (Low Voltage Switchyard Plan)

2-12 Company Services/GPC Drawing AX3D-BA-L55A (230kv Offsite Source No. 1 C.T. & P.T. Connecting - Sheet 1)

2-13 Bechtel Drawings 2X3D-BB-BOIL (Elementary Diagram -Electrical System Generator Tripping)

2-14 Southern Company Services/GPC Drawing AX3D-BA-L57D (500kv PCB No. 161520 Close & Trip No. 1)

2-15 Southern Company Services/GPC Drawing AX9D-AA-L50T (500kv PCBs No. 161520/161620/161660 Single Line)

2-16 Southern Company Services/GPC Drawing AX3D-BA-L55A (230kv Offsite Source No. 1 C.T. & P.T. Connecting - Sheet 1)

2-17 Southern Company Services/GPC Drawing [illegible] L55B -(230kv Offsite Source No. 1 [illegible] Connecting - Sheet 1)

2-18 Southern Company Services/GPC Drawing AX3D-BA-L55C (230 Offsite Source No. 1 Diff. & Backup Relaying)

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*Refer to document number in parentheses

2-19 Southern Company Services/GPC Drawing AX3D-AA-L50B (230kv Single Line For PCBs 161760/161860/161960)

2-20 Bechtel Drawings 2X3D-AA-B04A (Three Line Diagram - Unit 2 AC Generator)

2-21 Bechtel Drawings 2X3D-AA-B02A (One Line - Relays & Meters For RATs)

2-22 Bechtel Drawings 2X3D-AA-B01A (Relays & Meters - Generator, Main, & UAT)

2-23 Bechtel Drawings 1X3D-BH-B55B (Res. Aux. XFMR INXRB CKT Switcher)

2-24 Southern Company Services/GPC Drawing AX3D-AAL50A (500kv & 230kv Sunstation Single Line Index Drawing)

2-25 Nuclear Plant Maintenance Work Order No. 18906364

2-26 Southern Company Services/GPC Drawing AX3D-BA-L52R

2-27 Southern Company Services/GPC Drawing AX3D-BA-L52N Elementary

2-28 Southern Company Services/GPC Drawing AX3D-BA-L52P Diagrams For 230kv PCB

2-29 Southern Company Services/GPC Drawing AX3D-BA-L52Q No. 161860

2-30 Southern Company Services/GPC Drawing AX3D-BA-L52S

2-31 Southern Company Services/GPC Drawing AX3D-BA-L52H

2-32 Southern Company Services/GPC Drawing AX3D-CA-L72K

2-33 Bechtel Drawing 1X3D-AA-D03B (4160 V Switchgear 1BA03)

*2-34 (2-40) Bechtel Drawing 1X3D-AA-D03A (4160 V Switchgear)

2-35 Bechtel Drawing 1X3D-AA-D02B (4160 V Switchgear 1AA02)

2-36 Bechtel Drawing 1X3D-AA-D02A (4160 V Switchgear 1BA02)

2-37 Bechtel Drawing 1X3D-AA-D01A (4160 V Switchgear 1NA01)

2-38 Bechtel Drawing 1X3D-AA-D04A (4160 V Switchgear 1NA04)

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2-41 Bechtel Drawing AX3D-BA-D02C (4160 V Breaker ANA0203)

2-42 Bechtel Drawing AX3D-BA-D03C (4160 V Breaker [illegible])

2-43 Bechtel Drawing AX3D-BA-D03B (4160 V Breaker ANA0301)

2-44 Bechtel Drawing 1X3D-BA-D01J (4160 V Breaker 1NA0111)

*Refer to document number in parentheses

2-45 Bechtel Drawing 1X3D-BA-D04D (4160 V Breaker 1NA0412)

2-46 Bechtel Drawing 1X3D-BA-D02C (4160 V Swgr 1AA02 INCM Brkr 1NXRB)

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2-50 GPC Procedure No. 13145-1 "Diesel Generators"

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2-53 Delaval Drawing 09-500-76021 Sh 1

2-54 Delaval Drawing 09-500-76021 Sh 2

2-55 Delaval Drawing 09-500-76021 Sh 3 Diesel Generator

2-56 Delaval Drawing 09-500-76021 Sh 4 Engine Control

2-57 Delaval Drawing 09-500-76021 Sh 5 Panel Schematics

2-58 Delaval Drawing 09-500-76021 Sh 6

2-59 Delaval Drawing 09-500-76021 Sh 7

2-60 Delaval Drawing 09-500-76021 Sh 8

2-61 Delaval Drawing 09-695-76021 "Engine Pneumatic Schematic"

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2-63 10 Mile EPZ Map

2-64 NOUE ED Checklist Of 3/23/90

2-65 EOF Personnel For 3/20/90

2-66 Security Emergency Response Organization for 3/20/90

2-67 TSC Personnel 3/20/90

2-68 CR Personnel 3/20/90

2-69 OSC Personnel 3/20/90

2-70 Training Records EOF

2-71 Communicator Package Consisting Of:
Course Completion & Attendance Records
Training Student Handout Dated 6/27/89
Lab/Performance Exercise Guides
Lesson Plan RE LP 07001-02
Training Student Handout Dated 4/12/88
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2-72 Control Room Layouts
Procedure 10003-C

2-73 Previous Inspection Report:
50-424,425 89-21 & 89-25 8/25-27/89 Exercise
50-424/425 88-38 & 88-42 8/15-19/89 ERF Appraisal
2-74 QA Open Item OQA-87-292
2-75 Safety Standard Handbook
2-76 Emergency Director Log
2-77 TSC Log
2-78 Emergency Notification Messages:
#1
#2
#3
#4
#5
#6
#7
#8
#9
2-79 EOF Manager Log
2-80 Met Info
2-81 Dictated From GEMA To VEGP (FAX 3/22/90) Notification Information
2-82 Assorted News Dispatcher & Newspaper Articles
2-83 Procedures:
91001-C, Rev. 7
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2-84 GEK Training Student Handout
2-85 Handbook For General Employee Badge Training
2-86 3/2/90 Test "Data Sheet 4" Of 91204-C & Data Sheet 7
2-87 3/20/90 Notes "Emergency Response" - From Region II
2-88 3/21/90 Notes 3 Pgs. Notes
2-89 8 Hr Report Dated 3/20-90
2-90 3/22 Results Sheet On Accountability & TSC Procedure Verification; 3/22 Restriction On Offsite Interaction - Phone Call; 3/22 BUENN Notes
2-91 3/22 Interview - Herb Whitener RII
2-92 3/23 Notes CR Procedure Verification
2-93 RII Plant Note On Capt. T. Nash
2-94 Interview Pauline Jenkins - Communication
2-95 Interview Theresa Jones - Communication
— * 2-96 Interview Jimmy Cash - Operations Supt.
— * 2-97 Interview William Burmeister - Plant Duty Manager
2-98 Results Of Selected Sampling Of Employee In Security Bldg.

2-99	E Plan Section H-3 "Activities And Staffing of Emergency Facility"
2-100	Interview John Hopkins - Shift Supt. Unit 1 (Interview Incomplete)
2-101	Interview Capt. William Johnson Accountability & Log Form
2-102	Interview Lt. William Stewart Accountability
2-103	Historical Classification of Emergency & Phone Communication Notes - Bob Trojanowski - RII GAO
2-104	E Plan, Section B, "Onsite Emergency Organization"
2-105	Results Of CR Walkdown Of Procedure File As Found
2-106	Backup ENN Description
2-107	Met Bldg. Inspection Notes
2-108	Listing Of Onshift People For Inspection
2-109	Drug And Alcohol Screening Report On Donnie Willhite (Fuel Truck Driver)
2-110	Inventory Of Fuel Truck
2-111	Procedure 00656-C, Traffic And Parking Control, Rev. 0
2-112	Procedure 90015-C, Vehicle Access, Rev. 8
2-113	Procedure 00653-C, Protected Area Entry/Exit Control, Rev. 8
2-114	Procedure 70030-C, Traffic And Parking Control, Rev. 2
*2-115(2-112)	Procedure 90015-C, Vehicle Access To The PA
2-116	Procedure 00260-C, Hazardous Substance And Waste Control, Rev. 5
2-117	Procedure 00261-C, Fuel Oil Handling And Safety, Rev. 5
2-118	The Alvin W. Vogtle Electric Generating Plant: Safety Standards
2-119	Figure 16-1 "Offsite AC" (FSAR)
*2-120(307)	GPC - VEGP - Unit 1 Control Log - pp 5283-84 (3/20/90)
2-121	GPC - VEGP - Unit 2 Control Log - pp 2601-03 (3/20/90)
2-122	GPC - VEGP - Unit 1 SS Log - pp 5413-14 (3/20/90)
2-123	GPC - VEGP - Unit 2 SS Log - pp 2875-77 (3/20/90)
2-124	VEGP - Unit 2 Trip Report
*2-125(33)	Proteus Alarm Printout - Unit 2 (3/20/90)
*2-126(32)	Proteus Alarm Printout - Unit 1 (3/20/90)
2-127	DC 1-90-0097 (1A D/G Sequencer Prob)
2-128	Control Room Layouts
2-129	Procedure 13145-1, Rev. 0 "Diesel Generators"
2-130	Licensed Operator Training Materials For Loss Of Electrical Power
2-131	Licensed Operator Training Materials For Loss Of Residual Heat Removal
2-132	Licensed Operator Training Materials For Emergency Plan Implementing

*Refer to document number in parentheses

2-133 Outside Operator Training Material For Diesel Generator Operation

2-134 Procedure No. 12006-C Unit Cooldown to Cold Shutdown 3/8/90

2-135 Procedure No. 13005-1 Reactor Coolant System Draining 2-23-90

2-136 Procedure No. 13011-1 Residual Heat Removal System 3-26-90

2-137 Procedure No. VEGP 00052-C Procedure No. 12007-C Refueling Entry 3-26-90

2-138 Procedure No. 12007-C Refueling Entry 3/8/90

2-139 Procedure No. 12000-C Refueling Entry 3/8/90

2-140 Procedure No. 18019-C Abnormal Operating Procedure Loss of Residual Heat Removal 3/8/90

2-141 Procedure No. 19100-C Emergency Operating Procedure ECA-O.O Loss of ALL AC Power 3/16/90

2-142 Procedure No. 23985-1 RCS Temporary Water Level System Emergency Operating Procedure ECA - O>=O.O Loss of All AC Power 5/12/89

2-143 Procedure No. 54840-1 Installation and removal instructions for the RCS Temp. Level Indication Tygon Tube and the Defeat of the Residual Heat Removal Suction valve auto closure interlock

2-144 0950076021 pp. 9 of 9 Delaval Drawing Engine Control Panel

2-145 0968876021 pp. 1 of 3 Devalve Drawing Engine and Skid Ele. Schem

2-146 1X4DB111 Rev. 17 P&I Diagram Reactor Coolant System No. 1201

2-147 1X4DB112 Rev. 24 P&I Diagram Reactor Coolant System No. 1201

2-148 1X4DB114 Rev. 25 P&I Diagram Reactor Coolant System No. 1208

2-149 1X4DB115 Rev. 23 P&I Diagram Chemical & Valve No. 1208

2-150 1X4DB116-1 Rev. 23 P&I Diagram Chemical & Valve No. 1208

2-151 1X4DB116-2 Rev. 16 P&I Diagram Chemical & Valve No. 1208

2-152 1X4DB117 Rev. 19 P&I Diagram Chemical & Valve No. 1208

2-153 1X4DB118 Rev. 20 P&I Diagram Chemical & Valve No. 1208

2-154 1X4DB119 Rev. 21 P&I Diagram Safety Injection System No. 1204

2-155 1X4DB120 Rev. 15 P&I Diagram Safety Injection System No. 1204

2-156 1X4DB121 Rev. 25 P&I Diagram Safety Injection System No. 1204

2-157 1X4DB122 Rev. 27 P&I Diagram Residual Heat Removal System No.1205

2-158 Loss of All AC/4160V IE Power Lesson Plan

3. Briefing Book document for IIT Leader

3-1 Ltr to W. G. Harston, III fm S. D. Ebneter dtd 3/23/90. Subject: Confirmation of Action Letter

3-2 PNO-II-90-16, Subject: Site Area Emergency at Vogtle Unit 1 Loss Of Offsite Power (3/20/90)

3-3 PNO-II-90-16A, Subject: Site Area Emergency at Vogtle Unit 1 Loss Of Offsite Power (3/21/90)

3-4 PNO-II-90-16B, Subject: Augmented Inspection Team is Dispatched to Vogtle Unit 1 (3/22/90)

3-5 EVENT NUMBER: 18024 (3/20/90)

3-6 DRAFT: NRC Staff Dispatches Augmented Inspection Team to Vogtle Nuclear Power Plant

3-7 Meeting Purpose: NRC ENTRANCE (3/22/90)

3-8 Status Summary 1: (3/20/90)

3-9 Status Summary 2: (3/20/90)

3-10 Article: Alert Declared at Vogtle after Truck Hits Tower (3/21/90) (The Augusta Chronicle)

3-11 Article: Outage at Vogtle Means Moment of Fear of Some (3/21/90) (The Augusta Chronicle)

3-12 Questions and Responses

3-13 Electrical Distribution Schematic Charts

3-14 VR-1 Update 1500 3/22, Site Area Emergency (3/20/90)

3-15 VR-2 Update 1500 3/23, Site Area Emergency (3/20/90)

3-16 Attention - Quarantine List and Licensee Restrictions

3-17 Chart - D/G Testing Unit 1 (3/24/90)

3-18 1st Draft of GPC's Event Critique (uneventful). Event Title: Loss of Offsite & Onsite AC Power (3/20/90)

3-19 Power Level/Mode and Inoperable Equipment/Abnormal System Alignment (Info)

3-20 Interview Notes (3/20/90)

3-21 Interview List (3/23/90)

3-22 Miscellaneous Notes

3-23 Status of AIT Charter Item Assigned to Rick Kendall (Items Nos. 5 & 6) (3/24/90)

* 3-24 Discussion w/ Paul Kochery (10:30 a.m.) (3/25/90) notes from Rick Kendall

3-25 Offsite Commendation 5a

3-26 Onsite Notification 5b

3-27 GPC, Vogtle Plant, Unit 1 Auxiliary Building Radwaste Operator Log (3/20/90)

3-33 GPU, Vogtle Electric Generating Plant, Unit 1 Outside Area Operating Log (3/20/90)

3-34 Preliminary Thermocouple Reading Charts for Unit 1

3-35 Interview Schedule (3/23/90)

3-36 Interview Schedule - AIT (3/24/90)

3-37 Warren Lyon - Status Report Notes for Item 1 (3/24/90)

3-38 Status Regarding AIT Charter Item No. 3, Charter Item Description

3-39 Status Regarding Summer (#4) (3/24/90) (Testa)

3-40 Vogtle AIT Chart Item 7 (Trager)

4. Evaluation of Potential Explosion in the Vogtle Switchyard

   4-1 Event Evaluation #1 (3/24/90)
   4-2 Event Evaluation #2 (4/10/90)
   4-3 Event Evaluation #2 (4/12/90)

5. Entrance Presentation by Georgia Power Company 3/26/90

6. PNO-IIT-90-02 3/26/90 Subject: IIT arrive at Vogtle Site

7. Agreements signed by (3) industry representatives on Waiver of Compensation, Conflict of Interest and release of Investigation Information for industry participating in IIT

8. Bulletin Board Notice

9. Letter to Document control desk, USNRC from W. G. Hairston III, Sr. V.P., Nuclear Operation, Georgia Power Co., dated 11/30/89. Subject Vogtle Electric Generating Plant Hardware Modifications pursuant to generic Letter 88-17

10. Letter to document control desk, USNRC from W. G. Hairston III, GPC, dated 2/2/89. Subject: Plant Vogtle - Units 1 and 2, NRC docket 50-424, 50-425, Operating License NPF-68, Construction Permit CPPR-109 Response to Generic Letter 88-17

11. Letter to document control desk, USNRC W. G. Hairston, III, GPC, dated 12/29/88; Subject: same as #10

12. Sequence of Events Chronology of Site Area Emergency 3/20/90 (Received 3/27/90) from license

13. Letter to W. G. Hairston, III, GPC, from A. R. Herdt, NRC, dated 7/31/89. Subject: Notice of Violation (Inspection Report Nos. 50-424/89-19 and 50-425/89-23) w/NRC Inspection Manual Temporary Instruction 2515/101

14. Local newspaper coverage - March 27, 1990

15. Interoffice Memo from G. Bockhold, Jr., Plant Manager, to Vogtle Site Personnel dated 3/27/90. Subject: Vehicles in Perimeter Area

16. Entrance Meeting with Licensee and Personnel Statements (3/26/90)

16-1	Entrance Meeting Notes
16-2	J. Hopkins - SS
16-3	R. B. Snyder - SS
16-4	P. Vannier - RO
16-5	K. Jones - CRO

17. Order to quarantine

18. Letter to C. C. Miller, Mgr. of Engineering, Vogtle, from W. C. Ramsey, Jr., dated 2/16/90. Subject: Vogtle, Units 1 & 1, Final Response to Request for Engineering Assistance. Attachment: Loss of Decay Heat removal Analytical Studies for Vogtle 1 & 2, A response to GL 88-17

19. Training Student Handout No. GE-HO-88002-00-001-C Continuing Training--RHR Mid-Loop Oper"

20. Training Lesson Plans:

    20-A Continuing Training--RHR Mid-Loop Oper. No. GE-LP-88002-00-C

    20-B Emergency Diesel Generator Auxiliaries Fuel Oil System No. NL-LP-11202-01-C

    20-C Emergency Diesel Generator General Overview No. NL-LP-11201-00-C

    20-D Emergency Diesel Generator Auxiliaries No. NL-LP-11203-02-C

21. EOP No. 19100-C, Revision 4, ECA-0.0 Loss of All AC Power

22. 4160V AC 1E Electrical Distribution, Procedure No. 13427-2, Revision 5

23. Loss of Class 1E Electrical System, AO Procedure No. 18031-C, Revision 6

24. Boron Injection Flow Path Verification - Shutdown, Procedure No. 14406-1, Revision 3

25. Generator and Engine Control Panel Functional Test Procedure No. 27563-C, Revision 2

26. T-ENG-90-11, Rev. 1, "A-TRAIN UNDERVOLTAGE TEST" Expiration Data: 4/8/90

27. Temporary Procedure No. T-ENG-90-12, B-Train Undervoltage Test

28. Temporary Procedure No. T-ENG-90-13, Sequencer Operability Check

29. Temporary Procedure No. T-ENG-90-14, Unit One Train B DCP 88-VIN0070 Sequencer Functional Test

30. T-ENG-90-15, Unit One Train A, DCP 88-VIN0070 Sequencer Functional Test

31. Maintenance Work Order (MWO) 19001576, 3/28/90 (D/G 1A)

*32. Proteus Alarm Printout (U2) *(33)

*Refer to document number in parentheses

33. Proteus Alarm Printout (U2)

34. List of Quarantined Equipment (Revised 3/29/90 Rev. 2)

35. Personnel Interviews

35-1	K. Pope - SS
35-2	W. Burmeister - Unit Superintendent
35-3	N. Dewbre, P. Jenkins, T. Jones - Shift Clerks
35-4	F. Thompson - EGS; R.Moye - ESS
35-5	G. Bockhold - PM
35-6	D. Vineyard - SS
35-7	W. F. Kitchens - Ass't. PM
35-8	D. Hines, D. Daughhetee, E. Pickett, J. Stanley
35-9	J. P. Cash - OS
35-10	T. C. Eckert - Oper. Dept.

36. List of quarantined equipment (Revised 3/29/90 Rev. 3)

37. Personnel interviews

*37-1	W. Johnson, W. Stewart - Security
37-2	M. Lackey - Outage Planning Mgr.
37-3	M. Lackey, R. Barlow, J. D'Amico - Scheduling
37-4	J. Roberts - EP
37-5	H. Handfinger - Maintenance Mgr.
37-6	G. Bienenborg, I. Kochery

38. PNO-IIT-90-02A

39. Training Lesson Plan No. NL-11204-OOC Emergency Diesel Generator-Engine Control and Protection 5/11/89

40. Procedure No. 13415-1,Rev.6, Reserve Auxiliary Transformers, 6/30/89

41. Training Lesson Plan No. LO-LP-28201-09-C Sequence Operation, 7/26/89

42. EOP No. 19101-C, ECA-0.1 Loss of all AC Power Recovery without SI Required

43. Training Lesson Plan No. NL-11205-01C, Emergency Diesel Generator Control and Protection, 8/29/89

44. Training Lesson Plan No. LO-LP-11102-05-C Emergency Diesel Generator Auxiliaries Air Start System, 12/8/89

45. Training Lesson Plan No. LO-LP-11001-06-C, Emergency Diesel Generator Introduction and Overview, 12/11/89

*SAFEGUARDS DOCUMENT NOT TO BE RELEASED

46. Personnel Interviews

46-1	E. Dannemiller, D. Huyck - Security
46-2	J. D. Jiles - Safety Specialist
46-3	R. Berry - Security
46-4	S. Chestnut -Training
46-5	K. Stokes - Sr. Plant Eng.

47. Procedure No. 13426-C, 4160V AC Common Non IE Electrical Distribution System, 1/26/90

48. Training Lesson Plan No. LO-LP-11103-06-C Emergency Diesel Generator Auxiliaries: Combustion air and exhaust, 2/28/90

49. Training Lesson Plan No. LO-LP-11104-C, Emergency Diesel Generator Auxiliaries Lube Oil and Crank Case Ventilation, 12/8/89

50. Training Lesson Plan No. LO-LP-11105-08-C, Emergency Diesel Generator Auxiliaries Jacket water cooling system, 12/8/89

51. Training Lesson Plan No. LO-LP-11101-07-C Diesel Generator Auxiliaries Fuel Oil System 12/8/89

52. Letter to J. P. Kane, GPC, from W.C. Ramsey (unsigned and undated). Subject: Response to REA VG-9010, Loss of decay heal removal

53. Emergency Response Facilities Input List, Revision 07.05, 12/11/86

54. Procedure No. 14406-1, Revision 3, Boron Injection Flow Path Verification-Shutdown, 2/6/89

55. VEGP Standing Order No. 1-90-05, Emergency Boration Flow Path, 3/1/90

56. Memorandum for A. Chaffee from S. Ebneter, undated (received 3/30/90). Subject: Designated Regional Point of Contact

57. Procedure No. 17038-1, Rev.7, Annunciator Response Procedures for ALB 38 on EAB Panel, 3/11/90

58. Procedure No. 13011-1, Rev.18, RHR System, 3/11/90

59. Vehicle Access Request, 3/20/90

60. (AOP) Procedure No. 18019-C,Rev.7, Loss of RHR,3/16/90

61. PRB Comment Review Sheet (PRB-90-44) for Temporary Procedure No. T-ENG-90-14

62. GPC VEGP Handbook for General Employee Badge Training, GE-HO-00101-001-C, Rev. 5, 10/23/89 w/record of training dates for D. Willhite

63. Event Report No. 1-90-003, Additional Support Items

64. SPDS Checklist

65. Items on Fuel Truck 3/2-30/90

66. Photographs

66-1 Roll 1 - (3/24/90)

66-1-1	Voltmeters
66-1-2	Auto XFML No. 2
66-1-3	Auto XFMRs Nos. 1 & 2
66-1-4	Electrical Panel
66-1-5	Annunciators and Electrical Panel
66-1-6	Electrical Panel and Annunciator
66-1-7,8	Letdown/Chg. Flow and Bit Press
66-1-9	Accumulator Pressure Tanks 1 and 2
66-1-10	Accumulator pressure, Tanks 3 and 4
66-1-11	PROTEUS Computer Display
66-1-12	RWST Reset Switches (2) and RHR Suct Vent Line TRN-B
66-1-13	RHR to HL, RHR Suct Vent Line TRN-A, RWST Reset
66-1-14	RHR X Train A & B Outlet and Bypass
66-1-15	RHR Pump Pressure Trains A & B
66-1-16	Incore TC
66-1-17	Operator Aid for mid-loop on RCS Loop 1 Hot Leg NR Level, and RCS Loop 4 Hot Leg WK Level
66-1-18	SI Pump Disch Trains A & B

66-2 Roll 2 - (3/24/90)

66-2-1,2	Plant Safety Monitoring System
66-2-3	Control Rod Position
66-2-4	RCS Flow Trip Alarms
66-2-5	PRZR Pressurizer, PRZK Spray, PRZK LVL
66-2-6	RCS Press, RCS HL Temp
66-2-7	RCS CL Temp OP delta T, OT Delta T, Delta T
66-2-8	Chg Flow, RCS Loops 1-4, Delta T
66-2-9	Press, LTDN Flow, RCS Loop 4, RCS Temp Loop 3, Delta T
66-2-10	RCS Loop 4, RCS Temp Loop 2, Delta T
66-2-11	Safety A, RCS Loop 1 HL Press, RCS Temp Loop 1, Delta T

66-2-12 RCS Loop 4 (Delta T, OP Delta T, OT Delta T, T-AVG & RC Flow Loop 4)
66-2-13 RCS Loop 4, & RCS Flow Loop 4
66-2-14 RCS Loop 2, & RCS Flow Loop 2
66-2-15 RCS Loop 1, & RCS Flow Loop 1
66-2-16 RCS Loop 1 HL Press, RCS Temp Loop 1, Delta T
66-2-17 PRZR Relief Temp, RCS Loop 1
66-2-18 In Core TC
66-2-19-23 Electrical Panel and Annunciators

66-3 Roll 3 - (3/25/90) - Truck

66-3-1,2 Rear View
66-3-3-6 Blind Spot Assessment (180 to 195 feet)
66-3-7 Left View
66-3-8 Fuel Can
66-3-9 Closeup of event-related damaged area 1
66-3-10 Closeup of event-related damaged area 2
66-3-11,12 Closeup of event-related damaged area 1
66-3-13 Closeup of event-related damaged area 2
66-3-14 Fire Extinguisher
66-3-15 Fire Extinguisher
66-3-16,17 Front View
66-3-18 Inside Cab (Driver's Door Open)
66-3-19 Front Windshield (from inside)
66-3-20-23 Blind Spot Assessment through left and right mirrors
66-3-24 Left side view

67. Personnel Interview of G. Lee, J. Aufdenkampe, W. F. Kitchen, W. Burmeister & D. West

68. Diesel Generator Failure Analysis

69. Personnel Interview: D. DeLoach, J. Jackson, and S. Whitman

70. Photographs

70-1 Roll 4 - (3/24/90)

70-1-1 Operator Aid - RCS Loop 1 HL NR Level, and RCS Loop 4 HL WK Level
70-1-2 SI Sump Disch Trains A & B, and Containment Press
70-1-3 Letdown/Chg. Flow, BIT Press
70-1-4,5 Proteus Computer: In Core TC
70-1-6 Accumulator Level
70-1-7 RHR Pump Pressure

70-2 Roll 5 - Photos provide by Licensee (3/30/90) - The Welder

70-2-1-3 Side View
70-2-1-4,5 Front View

71. Meeting Attendance Record

71-1 IIT Entrance (3/26/90)
71-2 Diesel Generator (3/28/90)
71-3 Diesel Generator (3/30/90)

72. Access Record History, 3/20/90

73. Badge Record - 3/30/90

74. Evaluations of Initial Plant Conditions - J. D'Amico

75. Procedure No. PTDB-1 Tab 8.0, Rev. 2 - Pictorial Aids: RCS Elevations and Mid Loop Level Instrumentation (3/2/90)

76. Procedure No. 18015-C, Rev. 5, Loss of Instrument Air

77. Procedure No. 12000-C, Rev. 16, Refueling Recovery 3/8/90

78. Procedure No. 12006-C, Rev. 15, Unit Cooldown to Cold Shutdown 3/8/90

79. Procedure No. 12007-C, Rev. 14, Refueling Entry (Mode 5 to Mode 6), 3/8/90

80. Procedure No. 13005-1, Rev. 10, Reactor Coolant System Draining, 2/23/90

81. Procedure No. 17006-1, Rev. 11, Annunciator Response Procedure for ALB 06 on Panel 1A2 on MCB, 3/6/90

82. Bechtel Drawings

1X3D-AA-G02 C, Rev. 6,Vital Inster. Distr. Pnls.
AX3D-AA-A01A, Rev. 13, Main One Line 1 & 2
AX3D-AA-F19A, Rev. 10, 480V Motor Control Center
1X3D-AA-G05B, Rev. 14, 120V AC Non-Class IE key Instr.
AX3D-AA-HO1A, Rev. 9, TSC 125V DC/120V AC Non-Class IE Distr. Panels
AX3D-AA-AO1A, Rev. 16, Main One Line Unit 1
AX3D-AA-G02B, Rev. Main One Line Non-Class IE 125V DC & 120V Ess. AC System

83. MWO 18906587, SG#4, Unit 1, 12/23/89

84. Procedure No. 25270-C, REv.6, SG Nozzle Dam Checkout, Installation and Removal 12/16/88

85. Procedure No. 18004-C, Rev. 6, AOP RCS Leakage, 12/9/90

86. (AOP) Procedure No. 18006-C, Rev. 2, Fuel Handling Event, 8/4/88

87. (AOP) Procedure No. 18020-C, Rev.3, Loss of Component Cooling Water, 2/14/90

88. (AOP) Procedure No. 18021-C,Rev. 4, Loss of Nuclear Service Cooling Water System, 3/16/90

89. Procedure No. 18034-1,Rev. 1, Loss of Class IE 125V DC Power, 9/12/89

90. (AOP) Procedure No. 18028-C, Rev. 7, Loss of Instrument Air, 3/16/90

91. (AOP) Procedure No. 18038-1, Rev. 10, Operation from Remote Shutdown Panels, 8/29/89

92. Radiation Monitors Status

93. Relief Request 7 and 11 (with accompanying documents) Safety Injection System No. 1204

94. Procedure No. 00350-C, Rev.19, Work Request Program (12/27/89)

95-1. MWO #188[illegible]46 3/31/90

95-2. MWO #190[illegible]339 3/31/90

96. Section 6.0, Administrative Controls (Vogtle 1 & 2)

97. Information on Critical Safety Function Status Frees on SPDS (on EXF computer)

98. Training Records: A. L. Blalock, F. Redivannz, W. Hennessy, J. D. Williams, W. P. Stephens, W. M. Watkins, D. Haile, J. W. Covington, R. LeGrand, G. J. Durrence

99. Information on Maintenance Personnel Training on Mid-Loop Ops

100. Procedure No. 00400-C, Rev. 11, Plant Design Control, 2/24/90

101. Request for Engineering Review (Procedure No. 00400-C) of System 500 kv [illegible]

102. Relaying Data Sheet - Gen. No. 2 Main Bk. Primary

103. Sequence of Events w/source of information

104. 2.0 Hardware Configuration

105. Vendor Document Status Sheet-Manual Change No. 76021-9, 9/9/86

106. Security Department Report No. 3941-90

107. Joint News Release - 3/20/90

108. MWO (continued) No. 19001576

109. T-ENG-90-11, Rev. 1, A-Train under voltage Test w/related MWO's

110. 1A Diesel Generator Reference Material

111. 3/20/90 Logs: Unit 1 & 2 Shift Supervisor; 1 & 2 Control

112. Memo to G. Bockhold from G. R. Fredrick, dated 9/27/89. Subject: VEGP-1 and 2 QA Audit finding Report 350

113. Release #1 and # 2 - 3/20/90

114. OSC Log

115. MWO 18906328, 3/27/90

116. Attendance Roster - Diesel Generator Mtg. (3/31/90)

117. Personnel Interviews:

    117-1 K. Exly
    117-2 P. Humphrey
    117-3 J. Williams, D. Gustafson, G. McCarley

118. Operational Journals, 3/20, w/Situation Chart

119. MWO No. 19001684, (3/31/90) Diesel Generator - To Verify Timing of Two Trips During LOSP

120. Personnel Interviews:

    120-1 G. Schnieder (EP) and S. Threatt (PRH-EPC)
    120-2 R. Dorman - Training

120-3 M. Cagle - Maintenance
120-4 S. Driver - Training
120-5 D. Willhite - Truck Driver
120-6 E. J. Kozinsky - SS
120-7 S. Young - PFS

121. Design Change Request No. 88-VIN0070 w/Design Change Package Closure - Train A & B Sequencer Panels

122. Operation and Maintenance Instructions (SFSS) w/drawings

123. Vogtle Operating Records - VPO401GP

124. MWO 19001684 - 3/31/90
MWO 19001576 - 3/31/90
MWO 28900466 - 3/31/90

125. PROPRIETARY DOCUMENTS: W FIELD SERVICE PROCEDURES*

125-1 MRS 2.2.2 GPC-1, Post Activity Sign-Off for Area Cleanliness
125-2 MRS 2.2.2 GPC-21 Rev. 0, Nozzle Dam Hydrotest
125-3 MRS 2.2.2 GPC-22 Rev. 0, Nozzle Dam
125-4 MRS 2.2.2 GPC-23 Rev. 0, Nozzle Dam Manual
125-5 MRS 2.2.2 GPC-24 Rev. 0, Nozzle Dam, Leak Detection Manual

126. Procedure No. 29536-C, Cutage Management Program (3/4/88)

127. Procedure No. 27505-C, Rev. 2, Opening and Closing Containment Equipment Hatch (7/18/88)

128. Procedure No. 10013-C, Rev. 6, Writing EOPs from the Westinghouse Emergency Response Guidelines (9/22/88)

129. Procedure No. 29537-C, Rev. 1, Outage Scheduling (4/11/89)

130. Procedure No. 01000-C, Rev. 1, Management of Outages (6/9/89)

131. Procedure No. 10011-C, Rev. 13, Operations Procedure Preparation and Review Guidelines (7/5/89(

132. Procedure No. 10018-C, Rev. 11, Annunciator Control (9/26/89)

133. Procedure No. 10000-C, Rev. 16, Conduct of Operations (3/22/90)

*NOT TO BE RELEASED

134. Procedure No. 11011-1, Rev. 7, RHR Removal System Alignment (4/18/89)

135. Temporary Change to Procedure Form (TCP) No. 13001-1-12-90-1, RCS Filling and Venting (Expiration Date: 4/6/90)

136. Procedure No. 14230-1, Rev. 4, AC Source Verification (7/27/88)

137. Procedure No. 91403-C, Rev. 4, Site Evacuation (12/6/88)

138. (EOP) Procedure No. 19101-C, Rev, 8, ECA-0.1 Loss of All AC Power Recovery Without SI Required (7/26/89)

139. Procedure No. 19111-C, Rev. 8, ECA-1.1 Loss of Emergency Coolant Recirculation (7/26/89)

140. Procedure No. 13001-1, Rev. 12, Reactor Coolant System Filling and Venting (10/10/89)

141. System Block Diagrams - SFSS

142. Bechtel Drawings:

1X4DB113, RTD BY-PASS REACTOR COOLANT SYSTEM NO. 1201
1X4DB100, P & ID'S AND FLOW DIAGRAM LEGEND
AX3AE03-9-10, ELECTRICAL SCHEMATIC
ZX3AE03-10-10, ELECTRICAL SCHEMATIC

143. Unit 1 D/G Trip Sensor History w/Licensee's Draft Analysis of 1A Diesel Shutdown

144. PNO-IIT-90-02B - 4/2/90

145. Transcript: Briefing Meeting - 3/28/90

146. Personnel Interviews

1 - J. Acree - SS
2 - J. Aufdenkampe - Tech Support
3 - F. Pope - EO
4 - S. Owyoung
5 - J. Ealick
6 - M. Cagle
7 - M. Lackey (3/30/90)

* 147. Transcript: Meeting IIT and Licensee Personnel re Diesel Malfunction (3/30/90)

148. Transcript: Meeting w/ Event Critique Team - 3/31/90

149. Personnel Interview - D. Hines - PE

150. Transcript: Discussion re Results of Testing on A-Diesel Performed on 3/30/90

151. Memo for A. Chaffee from G. Zech, dated 4/3/90. Subject: Lessons Learned: Vogtle Site Area Emergency

152. Memo from M. S. Briney to G. Bockhold, Jr., dated 4/3/90. Subject: Calcon Temperature Switches

153. PROPRIETARY DOCUMENTS (INPO)*

Operating Experience Close Out Packages for the following:

SOER 85-01
SER 17-88
SER 73-83
SER 42-84
O&MR 272
SOER 85-04
O&MR 365
SOER 88-03
SER 5-89
SER 26-89
SER 36-88
SER 2-87
SER 35-86
SER 31-86
SER 23-86
SER 17-86

154. Calcon Switch Information

155. Quarantined Equipment List, Rev. 4 - 4/2/90

156. Meeting Attendance Record - Diesel Generator and IIT Exit - 4/2/90

157. Procedure 14980-1, Rev. 18 - Diesel Generator Operability Test (2/5/90)

158. Letter to Director, I&E, NRC, from B. C. Guntrum, Mgr., QA, IMO. Subject: Notification of a Potential Defect in a component of a DSR or DSRV Standby D/G ([illegible]/2[illegible]/88)

*NOT TO BE RELEASED

159. VHS Video Tape Recording: RHR Mid-Loop OPS

160. Photographs

160-1 Roll 6 - (3/31/90)

160-1-1,2	TSC
160-1-3-7	D/G 1A CK Control Panel
160-1-8,9	D/G 1A CK Annunciator Panel
160-1-10	PSMS Codes
160-1-11	Plasma Display and Keyboard of PSMS
160-1-12,13	Proteus Computer
160-1-14-16	NSCW, CCW
160-1-17,18	NSCW: Return Temp, Basin Lvl. & Hdr. Press.
160-1-19	NSCW A Flow: Supply and Return
160-1-20,21	Misc. Status Lights for NSCW
160-1-22,23	CCW PMP-1, 3
160-1-24,25	Rx MW Wtr. to CCW
160-1-26,27	CCW Train A Surge Tank
160-1-28,29	CCW Cnmt. Spray, & SI Panels
160-1-30,31	RHR & CVCS, & Accum. Panels
160-1-32-33	RWST Level
160-1-34	Page 1 of D/G Procedure (13145-1)

160-2 Roll 7 - (3/31/90)

160-2-1-5	Jacket Water Pressure Sensor
160-2-6-9	Pressure Sensor
160-2-10,11	Lube Oil Pressure Sensors (3)
160-2-12-14	Lube Oil Temperature Sensor
160-2-15-25	Jacket Water Temperature Sensors (3)
160-2-26-31	Turbo Lube Oil Sensor
160-2-32-34	Emergency Break Glass Start
160-2-35	Break Glass Hammer vs Flashlight Size

161. List of Vogtle 1 Event Headquarters Operations Center Participants

162. History of Quarantined Items

163. Quarantine Sign

164. Unit 1 Second Refueling Outage Charts:

164-1 Electrical System Work
164-2 1R-1 Target vs. Actual Critical Path

— * 164-3 First Drain to Mid-loop
164-4 IR-2 Target vs. Actual Critical Path
164-5 D/G Activities/TRN B
164-6 D/G Activities/TRN A

165. Unit 1 First Refueling Outage Charts:

165-1 Mid-loop Work and Support
165-2 Electrical System Work
165-3 D/G Activities/TRN B
165-4 D/G Activities/TRN A

166. Photographs

166-1 Roll #8 - (3/31/90)

166-1-1	D/G Control Panel
166-1-2	D/G Control & Annunciator Panels
166-1-3,4	D/G Annunciator Panel
166-1-5	SI Signal
166-1-6,7	Emergency Start
166-1-8	Procedure Sec. 4.4.3.1
166-1-9	Procedure Sec. 4.4.3
166-1-10	Jacket Water Lv. In.
166-1-11,12	Jacket Water In
166-1-13,14	Generator Bearing Temp.
166-1-15-17	Day Tank Level
166-1-18	D/G Control Panel
166-1-19-21	Local Remote Control Switch
166-1-22-25	D/G Back Panel (Inside)
166-1-26	Gaitronics Phone
166-1-27	Commercial Phone
166-1-28,29	Sound-Powered Phone
166-1-30	P. 1 of Procedure 13145-1
166-1-31	Lube Oil Press

166-2 Roll #9 - (3/30/90)

166-2-1-7	Sensor Points on Blackboard. in Large Conference Rm (LCR)
166-2-8-10	Electrical Configuration of Vogtle Plant
166-2-11	Blackboard Note to IIT
166-2-12-19	Sensor Points on Blackboard in LCR
166-2-20-34	Sensor Points on Blackboard in LCR showing steps to de-energize Sequencer on Panel

166-3 Roll #10 - (3/26/90)

166-3-1,2	New Restricted Area Sign and Rope
166-3-3,4	D/G Panel
166-3-5	D/G Annunciator Panel

167. Procedure No. 00053-C D/G 1A - Temperature Monitoring and Recording (4/4/90)

168. Meeting Transcripts:

* 168-1	Telecon between IIT, Licensee Personnel, and RII Personnel.
* 168-2	D/G Meeting (4/2/90)

169. Technical Specifications - Vogtle Unit 1 - SectionS 1-5

170. 1985 Maintenance Work Orders (MWOs) - History, Unit 1, D/G-Sensor Switches

171. 1986 MWOs - History, Unit 1, D/G-Sensor Switches

172. 1987 MWOs - History, Unit 1, D/G-Sensor Switches

173. 1988 MWOs - History, Unit 1, D/G-Sensor Switches

174. 1989 MWOs - History, Unit 1, D/G-Sensor Switches

175. 1990 MWOs - History, Unit 1, D/G-Sensor Switches

176. 1988 MWOs - History, Unit 2, D/G-Sensor Switches

177. ERF Computer Points

178. D/G Temperature Switch Calibration Data Received from Licensee - 4/6/90

179. PROPRIETARY DOCUMENTS - WESTINGHOUSE*

179-1	Core Map - Instrument Locations
179-2	Reference Loading Pattern

180. D/G (1A/1B) Start Logs

181. Ltr to R. Newton, WOG, fm A. Thadani, NRC, dtd 12/11/89. Subject: Loss of RHRS Cooling While the RCS is Partially Filled, WCAP-11916, July 1988, and Other Related WOG Activities

*NOT TO BE RELEASED

182. IE Bulletins

182-1 No. 80-12 (5/9/80)
182-2 No. 86.01 (5/23/86)

183. Information Notices

183-1 No. 80-20
183-2 No. 80-41
183-3 No. 81-09
183-4 No. 81-10
183-5 No. 86-39
183-6 No. 86-74
183-7 No. 86-101
183-8 No. 87-01
183-9 No. 87-23
183-10 No. 87-59
183-11 No. 89-41
183-12 No. 89-67

184. Memorandum for T. Murley, NRC, and E. Beckjord, NRC, fm E. Jordan, NRC, dtd 5/18/87. Subject: Loss of Decay Heat Removal Function at Pressurized Water Reactors with Partially Drained RCSs

185. NUREG/CR-5015; BNL-NUREG-52121, Improved Reliability of Residual Heat Removal Capability in PWRs as Related to Resolution of GI-99

186. Case Study Report AEOD/C503, Decay Heat Removal Problems at U.S. PWRs, 12/85

187. Documentation of Special Test of RHR Lineup to be Used with Path to Loops 1&2 only

188. MWOs on S/G Manway and Nozzle Dams

188-1 No. 18906589
188-2 No. 18906581
188-3 No. 18906579
188-4 No. 18906580
188-5 No. 18906582
188-6 No. 18906588
188-7 No. 18906590
188-8 No. 18906587

189. Operator Aids - Local Control Stations

190. Ltr to USNRC Document Control Desk fm W. G. Hairston, III, GPC, dtd 8/15/89. SUBJECT: VEGP LER 50-425/1989-023

191. Selected Licensed Operator RHR Training Material

192. Selected Licensed Operator Annunciator Training Material

193. Selected PEO, D/G Training Material; Lessons Plans & Response to Annunciators

194. Procedure No. 13431-1, Rev. 4, 120V AC IE Vital Instrument Distribution System (4/20/90)

195. Presentation to Region II NRC on Vogtle Site Area Emergency, March 20, 1990 (4/9/90 - by Licensee)

196. Generic Letters

196-1 No. 87-12, Loss of RHR While the RCS is Partially Filled (7/9/87)

196-2 No. 88-17, Loss of Decay Heat Removal (10/17/88)

197. Memo for J. P. Stohr, NRC, fm A. T. Boland, NRC, dtd 4/3/90, Subject: Critique of RII's Response to the Vogtle Incident

198. Appendix A to NRC TI 2575/103, Supplemental Information, Containment Closure and Reactor Coolant System Level Instrumentation (2/26/90)

199. Letter to R. C. Jones, NRC, from L. A. Walsh, WOG, dated 4/6/90; Subject: WOG Transmittal of Information Copy for Loss of RHR While Operating at Mid-Loop Conditions Guideline and Background Documentation

→ * 200. Transcript: Telephone Conversation w/IIT, Licensee, R-II (4/5/90)

201. NRC Personnel Interviews: 4/5/90

201-1 A. Thadani, NRR
201-2 W. Lyon, NRR

202. Unit 1A Train D/G - Air Receiver Dew Point Measurements

→ * 203. Transcript: Telephone Conference w/IIT, Licensee, R-II (4/6/90)

204. Personnel Interviews: 4/6/90

204-1	L. Walsh, Seabrook (telecon)
204-2	D. Marksberry, NRC/AEOD
204-3	J. MacKinnon, NRC/AEOD
204-4	P. Ray, NRC/AEOD
204-5	D. Ross, NRC/AEOD

205. Transcript: Telephone Conference w/IIT, Licensee, R-II (4/7/90)

206. Transcript: Telephone Conference w/IIT, Licensee (4/9/90)

207. NRC Personnel Interviews: 4/9/90

207-1	W. Hodges, R-I
207-2	G. Zech, AEOD

208. IE Information Notices

208-1	No. 83-56
208-2	No. 84-42

209. Preliminary D/G Instrument Test Outline (4/11/90)

210. Failures of Calcon Temperature & Pressure Sensors at Vogtle 1 & 2

211. NRC Information Notice No. 88-36

212. Transcript: Telephone Conference w/IIT, Licensee, R-II (4/10/90)

213. NRC Personnel Interviews: 4/10/90

213-1	R. Jones, DST
213-2	R. Eckenrode, HFAB
213-3	D. Tondi, ESB

214. Graph: Times To Core Uncovery Following Loss of RHR - During Mid-Loop Operations (4 loop W Plants Including Vogtle) - H. Ornstein (3/29/90)

215. Memorandum for M. W. Hodges, NRR, From J. E. Rosenthal, AEOD, dated 6/29/87; Subject: Additional Information on Loss of DHR at PWRs with Partially Drained RCSs

216. Memorandum for K. Kniel, RES, From J. E. Rosenthal, AEOD, dated 8/28/87; Subject: BNLs 8/18/87 Presentation of the Results of Draft Report "Improved Reliability of RHR Capability in PWRs as Related to Resolution of GI-99"

217. Memorandum for T. E. Murley, NRR, from E. S. Beckjord, RES, dated 4/4/88; Subject: GI-99 Risk Assessment Results

218. Event Report #1-90-003, Loss of Off-Site and On-Site AC Power (1E); Date of Event: 3/20/90

219. Data Sheets - MWO 19000092

220. Data Sheets - MWO 19000093

221. Commitments and Commitment Change Reports for D/G Preventive Maintenance: Report Nos. 9023, 9034 through 9038, 9052, 9082, 9086, and 13764

222. Data Sheets - MWO 19000094

223. Data Sheets - MWO 19000095

224. Letter to Director, IE/NRC, from L. R. Block, QA Engineer, IMO, dated 5/12/88; Subject: Additional Information to Supplement Letter of 4/29/88

225. Operation and Maintenance Manual, Appendix VII, Alarms and Safety Shutdowns (pp. 8-8, 8-8A)

226. Procedure No. 29101-C, Rev. 1, Emergency Lighting Surveillance (FSAR Fire Protection Surveillance), 12/5/89

227. Procedure No. 00414-C, Rev. 7, Operating Experience Program, 11/29/89

228. Procedure No. 11885-C, Rev. 10, D/G Operating Log, 5/11/89

    228-1 Data Sheets, DG1A - 3/20/90
    228-2 Data Sheet, DG1B - 3/23/90

229. Videocassettes - GPC, Vogtle, March 1990 D/G Tests

230. Bechtel Drawings:

    230-1 Lube Oil Piping Schematic 1X4AK01-27-11
    230-2 Starting Air Piping Schematic 1X4AK01-2? 12

230-3 Jacket Water Piping Schematic 1X4AK01-26-11
230-4 4160V INCM.BRKR 152-1BA0319 (DG 1B) 1X3D-BA-D03D
230-5 4160V INCM.BRKR 152-1AA0219 (DG 1A) 1X3D-BA-D02D

231. Operating Experience Close-Out Packages for the following:

IN 88-36
IN 86-101
IN 87-23
IN 80-20
IN 84-42
IN 89-64
GL 87-12
IN 89-67
IE Bulletin No. 80-12

232. PROPRIETARY DOCUMENTS (INPO):*

SER 60-83
SER 79-84

233. Transcript: Telephone Conference w/IIT, Licensee (4/11/90)

234. NRC Personnel Interviews: 4/11/90

234-1 S. Shankman, NRR
234-2 C. McCracken, NRR
234-3 H. Ornstein, AEOD

235. Maintenance Work Request (MWO 19001563) indicating when malfunction (communications failure) was recognized

236. Procedure to Test Annunciator Including First-Out (VEGP-1,1-3KJ,03,Rev.1)

237. VEGP Plant Review Board Meeting Minutes Which Include Discussion of Plant Condition on 3/20/90

238. Note to A. Chaffee, IIT, from R. Jones, RSB, dated 4/11/90; Subject: Documents Requested by IIT

239. MWO No. 19000909 (Data Concentrator) 2/22/90

240. Drawings:

240-1 Field Change Requests; Reference Drawing Numbers:

2X4AK01-369 R/8 - Eng. Cont. Pnl. Schm.
2X4AK01-498 R/8 - Elec. Schm.
2X4AK01-361 R/8
2X4AK01-369 R/8

*NOT TO BE RELEASED

240-2 VEGP Procedure No. 50009-C, As Built Notice No. 89-V1MO70A001 (1/18/90)

240-3 Bechtel Drawings:

1X4AK01-42-11 Engine Control Panel Installation
1X4AK01-52-9 Engine Control Panel Schematic
1X4AK01-357-9 A.C. Schematic
1X4AK01-358-8 Control Schematic
1X4AK01-443-4 Engine Pneumatic Schematic
1X4AK01-458-7 Instrument Identification Schedule
2X4AK01-367-8 Engine Control Panel Schematics (2)
2X4AK01-428-5 Engine Pneumatic Schematic
2X4AK01-459-6 Instrument Identification Schedule

241. Receiver Air Pressures Observed by Plant Personnel on 3/30/90

242. History (Unit 1 - From First RHR During Second Refueling Outage to Initiation of 3/20/90 Event) Defining "Windows" of Operational Configurations Potentially Affecting the NSSS, Containment, and Supporting Systems

243. Licensing Document Change Request Pkg. No. FS 88-99-Rev. 1

244. Description of Electronic Data Reading ERF, Proteus and Fault Recorder

*245. Transcript: Telephone Conference w/IIT, Licensee, R-II (4/12/90)

*246. NRC Personnel Interviews: 4/12/90

246-1 K. Brockman, R-II
246-2 S. Ebneter, R-II
246-3 F. Rosa, NRR

247. Outage Scheduling Information

248. RAT Scheduling Information

249. PRT, RCS, RWST, SG, TS Information

250. Letter to C. K. McCoy, GPC, from J. L. Tain, Westinghouse, dated 3/16/90, w/RHR B Pump Vibration Data

251. MWO No. 19001432, 3/20/90 (D/G 1B)

252. MWO No. 19001433, 3/20/90 (D/G 1A)

253. MWO No. 19001537, 3/25/90 (D/G 1B)

254. MWO No. 19001576, 3/28/90 (D/G 1A)

255. MWO No. 19001684, 3/31/90 (D/G)

256. D/G Function Checkout Data

257. Transcript (unmonitored tape recording): Teleconference w/IIT, Licensee, R-II (4/3/90)

258. Transcript: Telephone Conference w/IIT, Licensee, R-II (4/13/90)

259. Transcript: Telephone Conference w/IIT, Licensee, R-II (4/16/90)

260. GPC Personnel Interview: R. Odom, G. McCarley, M. Sheibani (4/17/90)

261. VHS Audiocassette: Calibration Tests on Temperature Sensor for D/G (4/18/90)

262. Meeting Attendance Records: IIT Entrance Briefing (4/17/90); IIT Exit (4/18/90)

263. MWOs:

263-1 No. 18906592 - Containment Equipment Hatch (12/23/89)

263-2 No. 18906593 - Personnel Hatch (12/23/90)

264. Calcon Temperature Switch Response Test on 4/4/90

265. Note to D. Gustafson and K. Burr from R. Jones (GPC), 4/13/90; Subject: Vibration Readings on Temperature Switch Piping - 1A Diesel

266. Drawings: As Built Notices

No. 00517
No. 00121
No. 00122

267. Bechtel Drawings: One Line Diagrams

1X3D-AA-E17A, Rev. 6 - 480V Switchgear 1BB16
1X3D-AA-E07A, Rev. 7 - 480V Switchgear 1BB07
1X3D-AA-E06A, Rev. 6 - 480V Switchgear 1BB06
1X3D-AA-E10A, Rev. 9 - 480V Switchgear 1NB10
1X3D-AA-D03B, Rev. 9 - 4160V Switchgear 1BA03
1X3D-AA-D03A, Rev. 8 - 4160V Switchgear 1BA03
1X3D-AA-E01A, Rev. 11 - 480V Switchgear 1NB01
1X3D-AA-E04A, Rev. 6 - 480V Switchgear 1AB04
1X3D-AA-E05A, Rev. 8 - 480V Switchgear 1AB05
1X3D-AA-D02B, Rev. 6 - 4160V Switchgear 1AA02
1X3D-AA-D02A, Rev. 8 - 4160V Switchgear 1AA02
1X3D-AA-E16A, Rev. 5 - 480V Switchgear 1AB15
1X3D-AA-A01A, Rev. 16 - Unit 1

268. Deficiency Tracking System

269. Containment Penetration Control Package (Procedures and Surveillance Task Sheets)

270. Letter to All Holders of RO and SRO Licences for PWRs from T. E. Murley, NRC/NRR, dated 11/3/88; Subject: Operator Diligence While in Shutdown Conditions

271. Letter to H. P. Allen, Southern California Edison Co., from T. E. Murley, NRC/NRR, dated 12/2/88; Subject: Loss of Decay Heat Removal

272. Calibration Data Sheets for Pressure Switch

273. Annunciator Response Procedures for ALB-09

273-1 Panel 1C1 on MCB
273-2 Panel 2C1 on MCB

274. ERF Computer Points (continued)

275. Training Student Handout - RHR System

276. Annunciator Response Procedures for ALB 36 on EAB Panel

277. Conoseal Status and Description at Time of Site Area Emergency on 3/20/90 (Note: Drawing No. 1X6AB02-288-1 contains proprietary data*)

278. Procedure No. 00250-C, Rev. 5 - Hazardous Substance and Waste Control (10/24/89)

*NOT TO BE RELEASED

279. Containment Building Penetrations Verification - Refueling

279-1 Procedure No. 14210-1, Rev. 4
279-2 Procedure No. 14210-2, Rev. 2

280. Bechtel Drawings: Containment Building Piping Areas

281. Westinghouse Drawing: Safeguard Actuation System - PROPRIETARY*

282. Drawings: Electrical, Control Systems, Mechanical, Nuclear

Pressurizer Press Control
Steam Dump Control
Rod Controls
Steam Generator Trip Signals

283. Security Vehicle Log -3/20/90

284. NRC Personnel Interview: J. Calvo, NRR

285. MWOs:

285-1 No. 19001482 - D/G 1B
285-2 No. 19001542 - D/G B
285-3 No. 19001677 - D/G A

286. Deficiency Cards:

286-1 No. 1-88-3016 - D/G A
286-2 No. 1-88-3083 - D/G B
286-3 No. 1-88-3453 - D/G 1A
286-4 No. 1-90-0182 - D/G 1A

287. Information Notice No. 90-25: Loss of Vital AC Power with Subsequent RCS Heat-Up

288. Interoffice Memorandum to C. K. McCoy, GPC, from M. J. Ajluni, GPC, dated 10/16/89; Subject: Vogtle's Operations Experience Program Assessment

289. INPO Database KEYWORD; DIESEL

290. Vogtle Status of NRC and INPO Operating Experience Doicument Reviews as of 4/18/90

291. Bechtel Drawings Related to Lighting

*NOT TO BE RELEASED

292. Bechtel One-Line Diagrams:

Non-Class 1E Distr. Panels
480V Motor Control Center

293. Bechtel Drawings:

1K4-1208-486-01, Reactor Head Vent
1K4-1201-064-02, RCS
2X6AB02-66-4, Reactor Vessel General Arrangements

294. VHS Audiocassette Tape: D/G Local, Control Room, Sequencer, TSC Local; Counter Nos.:

0000-363 Procedural Steps to Reset Sequencer (3/29/90)
0364-969 DG1A Sensors (3/31/90) - Lube Oil High Temperature Switch, Lube Oil Low Pressure Switches, Jacket Water Temperature Switches, Turbo Oil Pressure Switch, Jacket Water Pressure Sensor and Governor
0970-1004 Back of DG1A Control Station
1005-1380 Tested Emergency Lighting
1381-1507 Scanned DG1A Control Panel
-1508 Headset and Extension Cord
-1536 Normal and Emergency Jacks for Cord
1550-1614 Overview of DG1A Room
1615-1768 Inside of DG1A Control Panel Showing Sensor lines
1769-1810 Front of DG1A Control Panel Showing Emergency Start Button Using Break Glass Instrument (or Unscrew Glass)
1811-2207 Walkthrough of Emergency Start Procedure, SOP No. 13145-1, Section 4.4.3
(2043) Operator Aid Needed: Trips on Front Panel
2208-2837 First Out Walkthrough, Noting Annunciator Trips Provided
(2289) Operator Aid: Stand for Procedures
(2423) Required by ARP to Log First Out and Report to Control Room
(2544) Tested Annunciators Using Annunciator Response Control Buttons
(2740) Must Report All Major Functions and Malfunctions (DG Trips, First Out According to Admin. 10000 Procedure; also 13145, 14980); If Tripped, Logged and Reported to Engineering

2838-2925 Front of DG1A Panel; Magnetized Procedure Book; Ties on First Out Discussed (Two Valid Trips Simultaneously); one Reactor Trip is Eliminated or Might Not Work

2926-2993 Communication Phones: Gaitronics, Commercial Headphones, Sound Powered, and Extension Cords (25 feet)

2994-3311 Technical Support Center

3312-3430 DG1A Annunciator and Control Panels

- (3329) Sticker on C-9 for MWO and Another Sticker
- (3400) First Out on Reactor Trip Shown
- (3429) First Out Response Control Buttons for Reactor System

3431-3500 Safety Parameter Display System (SPDS)

-3505 PROTEUS

-3517 ERF

3526-3804 Walkthrough of Control Boards

- (3535) Plant Status Monitoring System (RVLIS, Incores, etc.) on Plasma Screen
- (3621) Relevant Indications
- (3636) Reactor Vessel Level and RVLIS, Scanned
- (3680) NCSW System
- (3703) CCW System
- (3732) RHR System
- (3800) Tower Platform in Control Room

3805-3844 Jacket Water Temperature Sensor in I/C Shop

3845-4179 Unit 1 Control Room (4/2/90)

- (3855) Description of Response Control Buttons in Control Room for DG1A: ACK, RESET, TEST, Including Contrast of What Happens at DG1A Local Control Station
- (4130) Reactor Trip; First Out Described and Shown

41[illegible] Turbine Trip (on Back Panel); First Out Explained, Including Response Button Controls; Electrical Hydraulic Control Cabinet Shown

- (4290) First Hit and Electrical Malfunction Reset Explained

4440-4934 DG1A Control Station

- (4625) How to Clear F6 (Switch Not in Auto); Clears Itself

4680-4878 Summary of What Operator Has to Do to Clear Audible Response Condition; Alarm Goes Solid (If in series, Silence, Acknowledge, Reset)

- (4878) How Operator Knows Condition is Cleared

4934 END

295. Spare Temperature Sensor Calibration Data

296. Test Procedure No. 17133

297. Security Access Printout for M. Lackey and S. Chessnut (3/20/90)

298. Letter to C. K. McCoy, GPC, from J. L. Tain, Westinghouse, dated 4/20/90; Subject: Vogtle, Unit 1, Vent Rate Through Reactor Vessel Head, Thermocouple Assemblies

299. Calcon Temperature and Pressure Switch Data (3 Volumes)

300. Letters to NRC Document Control Desk from W. G. Hairston, III, GPC, dated 4/19/90; Subjects:

300-1 VEGP LER, Loss of Offsite Power Leads to Site Area Emergency

300-2 VEGP LER, Unit 2 Reactor Trip from Unit 1 Reserve Auxiliary Transformer Feeder Line Fault

301. Security Logs for Entering and Exiting Control Room, Containment, D/G Room (3/20/90)

302. PROPRIETARY DOCUMENTS (INPO)*

Operating Experience Closeout Packages for the Following:

IN-81-09
SER-5-83
SER-38-83
SER-76-84
SER-36-87
SER-5-89
SER-74-81
SER-78-81
SER-87-8
O&MR-295
SER-15-87
SER-26-89

303. Wyle Laboratories Test Results - Temperature Sensors (submitted 4/26/90)

304. Response to Questions Regarding Containment LCO Logging, RCS Level, Gravity Feed

*NOT TO BE RELEASED.

305. Event Data Collection System - Equipment Locations Sketch

306. Wyle Laboratories Test Results - Temperature Sensors (submitted 4/27/90)

307. Unit 1 Control Log for 3/19/90 through 3/21/90 (pp. 5281-5287)

308. VEGP Plant Review Board Meeting Minutes Prior to the Event (3/2/90 through 3/19/90)

309. Wyle Laboratories Test Results - Temperature Sensors (submitted 4/30/90)

310. MWO for Head Removal/Replacement

311. Core-Exit Thermocouple Channel Calibration

312. Steam Generator Drawing

313. RHR Pump B Modification Data

314. Wyle Laboratories Test Results - Temperature Sensors (submitted 5/1/90)

315. Test Procedure for As-Received Testing and Calibration of Seven Calcon Model A3500-W3 Temperature Sensors (4/30/90) - by Wyle Laboratories for GPC

316. VEGP Procedure No. 93240-C, Rev. 8T, Reactor Vessel Assembly/Disassembly Instructions

317. MWO No. 18905286 - #036 Check Valve

318. Letter to GPC, Attn. K. S.Burr, from Wyle Laboratories, dated 5/2/90; Subject: Reliability Evaluation Testing of Calcon Model A3500-W3 Temperature Sensors

319. Information Notices

319-1	No. 82-20
319-2	No. 86-09
319-3	No. 83-17
319-4	No. 83-51
319-5	No. 85-28
319-6	No. 85-73
319-7	No. 85-91
319-8	No. 84-69
319-9	No. 84-69, Supplement 1

319-10   No. 86-73
319-11   No. 88-75
319-12   No. 89-87

320. IE Bulletin 77-01, Pneumatic Time Delay Relay Setpoint Drift

321. IE Circular 77-16, Emergency Diesel Generator Electrical Trip Lock-out Features

322. MWO for Accumulator Isolation Valve

323. Note to W. Lyon, IIT, from J. F. D'Amico, GPC, dated 5/2/90; Re: Information on Seal Table

324. Pressurizer Detail Drawing

325. MWO for Charging Line Check Valve 036 and 035

326. Wyle Laboratories Test Results - Temperature Sensors (submitted 5/4/90)

327. Bechtel Drawings of Accumulator Isolation Valve

328. Clearance Sheet for Reactor Head Vent

329. Instruction & Operating Manual Series X12 & X16 Models Solid State Annunciator Systems

330. PROPRIETARY DOCUMENTS (INPO)*

Operating Experience Closeout Packages for the Following:

SOER 83-01
SOER 81-10
SOER 83-06
O&MR 97
O&MR 110
O&MR 334
SER 82-78
SER 82-079
SER 09-85
SER 25-85
SOER 86-003
SER 89-028
SOER 85-001 (also includes 82-008 and 84-007)
SER 84-72
SER 88-031

*NOT TO BE RELEASED

331. Operating Experience Closeout Packages for the Following:

IN 77-01
IN 80-41
IN 82-20
IN 83-17
IN 83-51
IN 84-69 (and Supplement 1)
IN 85-28
IN 85-73
IN 85-91
IN 86-70
IN 86-73
IN 88-75
IN 89-87

332. Wyle Laboratories Test Results - Temperature Sensors (submitted 5/7/90)

333. Operating Experience Closeout Package for IMO Report

334. Deficiency Card No. 1900125 - S/G Manway

335. Bechtel Drawings: Swing Check Valve 3-C88; Motor Op. Gate Valve Mod. 10001GMS9FNH010

* 336. Draft "Corrective Actions for Site Area Emergency" and Unit 1 Status Report from 3-18 to 4-1-90 (submitted by licensee)

337. Memorandum to H. Wyckoff, EPRI, from J. O'Brien, EPRI, dated 5/11/90; Subject: Nuclear Plant Worker Capabilities Under Extreme Environmental Conditions

* 338. Letter to NRC Document Control Desk, from W. G. Hairston, III, GPC, dated 5/14/90; Subject: Vogtle Electric Generating Plant Corrective Actions for Site Area Emergency

339. Cooper Industries Test Results - Pressure Sensors and Shutdown Logic Board (submitted 5/15/90)

340. Motor Control Center Load Lists for 1NBS, 1NBI, and 1ABA to 1ABF

341. Wyle Laboratories Test Report (Preliminary), dated 5/12/90, "Reliability Evaluation Testing of Ten Calcon Model A3500-W3 Temperature Sensors"

* 342. List of Personnel On Site on March 20, 1990

343. Procedure No. 00150-C, Rev. 10, Deficiency Control (5/10/90)

344. Photographs - Switchyard Where Incident Occurred (Affected Pole, Fuel Truck, Fallen Insulator)

345. D/G 1B Troubleshooting Plan; Procedure 22981-C, Calcon Pneumatic Temperature Sensor Calibration (telecopy received 5/24/90)

346. Unit One "B" D/G Sensor Testing Sequence of Events - Preliminary (telecopy received 5/24/90)

347. Unit One "B" D/G Sequence of Events (telecopy received 5/29/90)

348. Telephone Conversations at NRC Headquarters Operations Center on March 20, 1990

348-1	Unmonitored Transcript
348-2	Memorandum from E. Weiss, AEOD to C. Siegel, IIT, dated May 31, 1990; Subject: Transcript of Telephone Conversaticns Relevant to the Vogtle Event

349. GPC Interoffice Correspondence from J. T. Swartzwelder to Department Heads dated 1/13/89; Re: Deficiency Control and QA Audit and Surveillance Finding Trend Report for Vogtle Unit 1

350. Test Report No. 17133-1, "Reliability Evaluation Testing of Ten Calcon Model A3500-W3 Temperature Sensors," by Wyle Laboratories for GPC (May 12, 1990)

351. Letter to M. W. Hodges, NRC, from R. A. Newton, WOG, dated November 21, 1988, with enclosure, WCAP-11916

352. NRC Inspection Manual, Temporary Instruction 2515/101, "Loss of Decay Heat Removal" (GL 88-17)

353. NRC Inspection Manual, Temporary Instruction 2515/103, "Loss of Decay Heat Removal" (GL 88-17) "Programmed Enhancements (Long Term) Review"

Hendt   D. Mat... Steve Varga ...

OFFICE OF THE SECRETARY
CORRESPONDENCE CONTROL TICKET

PAPER NUMBER: CRC-90-1000   LOGGING DATE: Sep 17 90

ACTION OFFICE: EDO

AUTHOR: Michael D. Kohn
AFFILIATION: DC (DISTRICT OF COLUMBIA)

LETTER DATE: Sep 11 90   FILE CODE:

SUBJECT: Request for proceedings and imposition of civil penalties

ACTION: Appropriate

DISTRIBUTION: Records, Chilk

SPECIAL HANDLING: 2.206 PETITION

NOTES: TREAT AS A PETITION UNDER 10 CFR 2.206 AND FOR ACKNOWLEDGEMENT PER D&SB THRU MR. CHILK AND OGC

DATE DUE:

SIGNATURE: DATE SIGNED:
AFFILIATION:

AH9
f 1137

OFFICE OF THE SECRETARY
CORRESPONDENCE CONTROL TICKET

PAPER NUMBER: CRC-90-1000 LOGGING DATE: Sep 17 90

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AFFILIATION: DC (DISTRICT OF COLUMBIA)

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DISTRIBUTION: Records, Chilk

SPECIAL HANDLING: 2.206 PETITION

NOTES: TREAT AS A PETITION UNDER 10 CFR 2.206 AND FOR ACKNOWLEDGEMENT PER D&SB THRU MR. CHILK AND OGC

DATE DUE:

SIGNATURE: DATE SIGNED:
AFFILIATION:

EDO --- 005836
90-11663 --1

Memo to file 9/26/90 0834

Talked to M.W. Horton concerning DG LER preparation and Safety Seq. Operation. Mike has no personnel notes or files. Other than the initial D.G. error on number of starts he believes there were no false statements in Prop. LER's and cover letters. On safety seq. he provided some new information related to T.S. interpretation and problem as Operations had applied the D.G. tech spec for sequencer downpowering in reality AFW & LOSP actuation logic was also affected. This item was identified by OSI & GPC agrees and concurs.

Mark J Cylinic

A/138

A/156

Release

pu 0'S

BULL & ASSOCIATES

Certified Shorthand Court Reporters
Computerized Reporting
since 1966

October 1, 1990

Jesse P. Schaudies, Jr., Esq.
127 Peachtree Street, N.E.
1400 The Candler Building
Atlanta, GA. 30303

RE: UNITED STATES DEPARTMENT OF LABOR
MOSBAUGH VS. GA. POWER CO.
C. A. FILE NO. 90-ERA-58

DEPOSITION OF: ROBERT PATRICK MC DONALD TAKEN ON 9/17/90

Enclosed is your copy of the deposition, which was stenographically reported in the above-captioned matter.

Also, enclosed is the jurat page from the original deposition.

It is requested that the deponent read the deposition for accuracy, making certain the court reporter correctly reported the testimony.

If there are corrections to typing or spelling, et cetera, these should be noted on a separate sheet of paper, indicating the page and line numbers on which these appear.

In addition to the foregoing, it is requested that the personally subscribed jurat page, together with the errata sheet, if any, be returned to the office of Bull & Associates, within the next thirty-day period. We shall, upon receipt, include same with the original deposition.

Your assistance and cooperation in this matter is appreciated.

Sincerely,

BULL & ASSOCIATES
Bunnie J. Taylor
Bunnie J. Taylor
Administrative Assistant

Enclosure-Copy of deposition
Original jurat page
A/139

CC: Michael D. Kohn, Esq.

Release

☐ 4651 Roswell Road, N.E.
Suite F504
Atlanta, Georgia 30342
(404) 256-2886

FAX (404) 843-3616
WATS 1 (800) 447-2855

☐ ✓ For Return Information

☐ 125 Habersham Drive
Suite D
Fayetteville, Georgia 30214
(404) 460-9774

A/H

Ajluni

UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
101 MARIETTA STREET, N.W.
ATLANTA, GEORGIA 30323

OCT 0 1 1990

Docket Nos. 50-424, 50-425
License Nos. NPF-68, NPF-81
EA 90-129

OCT 1990
Received
Sear-Vogtle

Georgia Power Company
ATTN: Mr. W. G. Hairston, III
Senior Vice President -
Nuclear Operations
P. O. Box 1295
Birmingham, AL 35201

Gentlemen:

SUBJECT: ENFORCEMENT CONFERENCE SUM[illegible]
(NUREG-1410 AND NRC INSPECTION REPORT NOS. 50-424/90-16
AND 50-425/90-16)

This letter refers to the Enforcement Conference held at our request on September 5, 1990, concerning activities authorized for your Vogtle facility. This meeting was requested to discuss numerous items identified by the Incident Investigation Team which was chartered in response to the Site Area Emergency event of March 20, 1990.

The circumstances, root causes, corrective actions, and safety significance of three areas of concern were discussed at this meeting. These three issues included the site's failure to make the required emergency notifications to state and local government agencies in a timely manner, the inability of site personnel to establish containment integrity within the required time limits, and the failure of the emergency diesel generator to provide AC power as intended (inadequate root cause analysis program).

Your presentation was effective in clarifying the reasons for the apparent violations and in delineating your corrective actions. This meeting was also beneficial in the fostering of open communications. A list of attendees and a copy of your handout are enclosed. We are continuing our review of these issues to determine the appropriate enforcement action.

In accordance with Section 2.790 of the NRC's "Rules of Practice," Part 2, Title 10, Code of Federal Regulations, a copy of this letter and its enclosures will be placed in the NRC Public Document Room.

Sincerely,

For Luis A. Reyes, Director
Division of Reactor Projects

Enclosures: (See page 2)

A114

~~9010150222~~ 19 pp

Enclosures:
1. List of Attendees
2. Conference Handout

cc w/encls:
R. P. McDonald
Executive Vice President-Nuclear
Operations
Georgia Power Company
P. O. Box 1295
Birmingham, AL 35201

C. K. McCoy
Vice President-Nuclear
Georgia Power Company
P. O. 1295
Birmingham, AL 35201

G. Bockhold, Jr.
General Manager, Nuclear Operations
Georgia Power Company
P. O. 1600
Waynesboro, GA 30830

J. A. Bailey
Manager-Licensing
Georgia Power Company
P. O. Box 1295
Birmingham, AL 35201

D. Kirkland, III, Counsel
Office of the Consumer's
Utility Council
Suite 225, 32 Peachtree Street, NE
Atlanta, GA 30302

Office of Planning and Budget
Room 615B
270 Washington Street, SW
Atlanta, GA 30334

Office of the County Commissioner
Burke County Commission
Waynesboro, GA 30830

Lonice Barrett, Commissioner
Department of Natural Resources
205 Butler Street, SE, Suite 1252
Atlanta, GA 30334

(cc w/encls cont'd - see page 3)

cc w/encls: (Continued)
Thomas Hill, Manager
Radioactive Materials Program
Department of Natural Resources
878 Peachtree St., NE., Room 600
Atlanta, GA 30309

Attorney General
Law Department
132 Judicial Building
Atlanta, GA 30334

State of Georgia

ENCLOSURE 1

LIST OF ATTENDEES

U.S. Nuclear Regulatory Commission

S. D. Ebneter, Regional Administrator, Region II (RII)
J. L. Milhoan, Deputy Regional Administrator, RII
L. A. Reyes, Director, Division of Reactor Projects (DRP), RII
A. F. Gibson, Director, Division of Reactor Safety, RII
J. P. Stohr, Director, Division of Radiation Safety and Safeguards (DRSS), RII
G. R. Jenkins, Director, Enforcement and Investigation Coordination Staff (EICS)
A. R. Herdt, Chief, Reactor Projects Branch 3, DRP, RII
K. E. Brockman, Chief, Reactor Projects Section 3B, DRP, RII
W. H. Rankin, Chief, Emergency Preparedness Section, Emergency Preparedness and Radiological Protection (EPRP) Branch, DRSS, RII
E. D. Testa, Senior Radiation Specialist, EPRP, DRSS, RII
B. R. Bonser, Senior Resident Inspector, Vogtle, DRP, RII
R. D. Starkey, Resident Inspector, Vogtle, DRP, RII
P. A. Balmain, Resident Inspector, Vogtle, DRP, RII
W. H. Miller, Jr., Project Engineer, DRP, RII
R. W. Borchardt, Regional Coordinator, Office of the Executive Director of Operations, NRC
T. A. Reed, Project Manager, McGuire, Office of Nuclear Reactor Regulation (NRR)
D. Hood, Project Manager, Vogtle, NRR
B. Uryc, Jr., Senior Enforcement Coordinator, EICS, RII
A. D. McQueen, Office of Enforcement

Georgia Power Company

R. P. McDonald, Executive Vice President
W. G. Hairston, III, Senior Vice President - Nuclear Operations
C. K. McCoy, Vice President - Vogtle Project
G. Bockhold, General Manager - Vogtle
P. D. Rushton, Manager, Engineering and Licensing
J. A. Bailey, Manager, Licensing
L. A. Ward, Manager, Maintenance and Support
K. R. Holmes, Manager, Training and Emergency Preparedness

ENCLOSURE 2

AGENDA

OPENING . . . . R. P. MCDONALD

INTRODUCTION . . . . C. K. MCCOY

A. EMERGENCY NOTIFICATION . . . . KEN HOLMES

1. STATEMENT OF WHAT OCCURRED (TIME LINE), EQUIPMENT
2. ROOT CAUSE
3. CORRECTIVE ACTION
4. SIGNIFICANCE

B. EQUIPMENT HATCH CLOSURE . . . . PAUL RUSHTON

1. DISCUSSION OF HATCH CLOSURE
2. ANALYSIS OF LOSS OF RHR
3. ACTIONS FOR UPCOMING OUTAGE AND FUTURE CONSIDERATION
4. SIGNIFICANCE OF MARCH 20 EVENT

C. D/G FAILURE . . . . LEWIS WARD

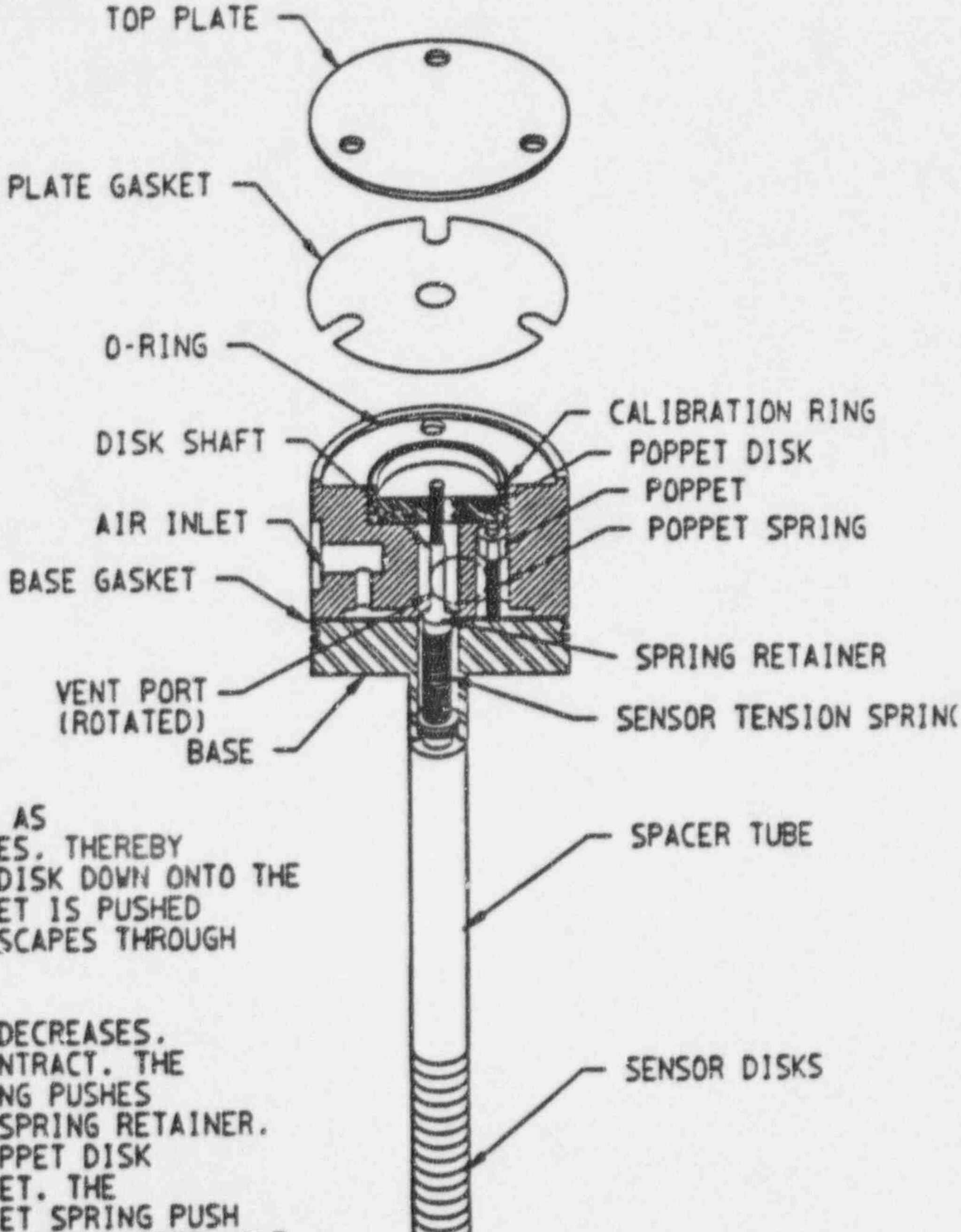
SCNOPCC

1. STATEMENT OF FACTS
2. ROOT CAUSE
3. CORRECTIVE ACTION
4. SIGNIFICANCE

HISTORY OF DG PNEUMATIC CONTROLS PROBLEMS

- IT HAS BEEN STATED THAT VOGTLE HAS LONG HISTORY OF DG PROBLEMS.
- NUREG-1410 LISTS 69 CALCON SENSOR FAILURES.
- PRESENTATION WILL COVER 3 AREAS:
  - CALCON TEMPERATURE SWITCHES
  - CALCON PRESSURE SWITCHES
- IN ADDITION, WE HAVE BEEN REQUESTED TO ADDRESS RECENT PROBLEMS WITH:
  - COOPER STARTING AIR ADMISSION VALVES
- FOLLOWING A PRESENTATION OF THE HISTORY AND TECHNICAL FACTS RELATED TO THESE 3 SEPARATE PROBLEMS, OUR CONCLUSION IS THAT:
  - NO CLEAR PRECURSORS TO THE 3/20/90 EVENT ARE EVIDENT FROM A REVIEW OF THE PREVIOUS SENSOR WORK HISTORY.
  - GPC DID DO THOROUGH TESTING AND IDENTIFIED 3 GENERIC COMPONENT DEFICIENCIES WITH TDI SUPPLIED COMPONENTS.
  - GPC DID NOTIFY OTHER OWNERS OF THESE PROBLEMS AND SOLUTIONS.

CALCON TEMPERATURE SENSOR
MODEL A3500-W3



SENSOR DISKS EXPAND AS TEMPERATURE INCREASES. THEREBY PULLING THE POPPET DISK DOWN ONTO THE POPPET. AS THE POPPET IS PUSHED DOWNWARD. THE AIR ESCAPES THROUGH THE VENT.

AS THE TEMPERATURE DECREASES. THE SENSOR DISKS CONTRACT. THE SENSOR TENSION SPRING PUSHES UPWARD AGAINST THE SPRING RETAINER. WHICH PUSHES THE POPPET DISK UPWARD OFF THE POPPET. THE AIR SUPPLY AND POPPET SPRING PUSH THE POPPET BACK INTO THE POPPET HOLE.

SUMMARY OF DIESEL GENERATOR

TEMPERATURE SENSOR PROBLEMS AT VOGTLE

ISTRUMENT UNCTION	PROBLEM	UNIT 1 CONSTRUCTION (8/85-12/86)	UNIT 2 CONSTRUCTION (1/88-12/88)	UNIT 1 1R1 (9/88-10/88)	UNIT 1 CYCLE 2 PRE-OUTAGE	UNIT 1 CYCLE 2 OUTAGE PRE-EVENT	POST EVENT 3/20-3/25/90
. W. Temp.	Out-of-cal.	8	7	4	4		2
. W. Temp.	Defective	1		6			1
. O. Temp.	Out-of-cal.		1		1		1
rg. Temp.	Defective			10		1	
isc.							
ib. Snsrs.	Defective		3				

JACKET WATER SWITCH PROGRAM RESULTS

- INSUFFICIENT TEMPERATURE STABILIZATION PERIOD PRIOR TO CALIBRATION.
- CONTAMINANTS ON THE TEMPERATURE SENSOR (TIP).
- CALIBRATION BATH HEATUP RATE.
- THERMOWELL SETSCREW TIGHTNESS.
- SPACER-TUBE TIGHTNESS.
- INTERNAL CONTAMINANTS.

JACKET WATER TEMPERATURE SWITCH PROBLEMS

-- ROOT CAUSE

- INTERNAL CONTAMINATION CAN CAUSE SWITCH TO CONTINUOUSLY VENT.
- CALIBRATION OF THE SWITCHES WAS INADEQUATE.

CORRECTIVE ACTION

- CALIBRATION PROCEDURE THAT CLEANS AND PROPERLY CONTROLS THE CALIBRATION REQUIREMENTS.
- RELIABILITY OF THE BASIC SWITCH COMPONENT WAS ESTABLISHED.
- SWITCHES HAVE BEEN DEFEATED IN THE EMERGENCY START MODE.
- CURRENTLY EVALUATING REPLACEMENT.

SIGNIFICANCE OF PROBLEM

- INADEQUATE CALIBRATION PROCEDURES AND TECHNIQUES RESULTED IN REDUCED RELIABILITY OF THE EMERGENCY DIESEL GENERATORS.

CONCLUSION

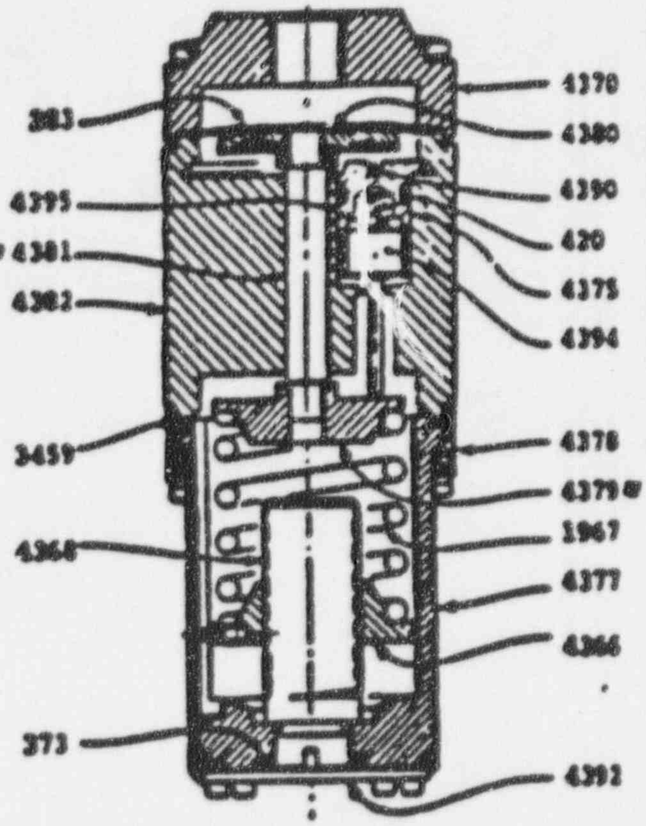
- BEGINNING WITH PROBLEMS ENCOUNTERED DURING THE FIRST UNIT 1 REFUELING IN 1988, GPC HAS TAKEN A SERIES OF ACTIONS TO RESOLVE DG TEMPERATURE SWITCH MALFUNCTIONS.

NOTES:

1. A — PART AVAILABLE
   FOR FIELD REPAIR.

2. LUBRICATE MOVING PARTS
   WITH DOW CORNING G-322-L.

3. *LOCTITE 4381/4379



SENSING HEAD-4347			
NO	DESCR	QTY	REF
283	DIAPHRAGM	1	A
4370	HEAD	1	A
4380	PRESSURE PLATE	1	A

VALVE SECTION-4338			
NO	DESCR	QTY	REF
420	O-RING	1	A
3681	VENT SEAL (NOT SHOWN)	1	A
4273	LABEL (NOT SHOWN)	1	
4375	VALVE GUIDE	1	A
4381	SHAFT	1	A
4382	BOOT	1	
4390	RETAINER CLIP	1	A
4394	VALVE POPPET ASSEMBLY	1	A
4395	SPRING	1	A

SPRING ADJUST GROUP-4337			
NO	DESCR	QTY	REF
373	O-RING	1	A
1967	SPRING	1	A
3459	O-RING	1	A
4366	ADJUSTER SEAT ASSEMBLY	1	A
4368	ADJUSTER SCREW	1	A
4377	SPRING HOUSING	1	A
4378	HOUSING SLEEVE	1	A
4379	SPRING SEAT	1	A
4392	NAMEPLATE	1	

B 4400 PRESSURE SENSOR SUB-ASSEMBLIES

CHG NO.

SUMMARY OF DIESEL GENERATOR

PRESSURE SENSOR PROBLEMS AT VOGTLE

NSTRUMENT UNCTION	PROBLEM	UNIT 1 CONSTRUCTION (8/85-12/86)	UNIT 2 CONSTRUCTION (1/88-12/88)	UNIT 1 1R1 (9/88-10/88)	UNIT 1 1R2 PRE-OUTAGE	OUTAGE PRE-EVENT	POST EVENT 3/20-3/25/90
. O. Pres.	Out-of-cal.	4		2			
. O. Pres.	Defective	1			1		
. O. Pres.	Out-of-cal.	1	1				
. O. Pres.	Defective				1		
. P. (P-3)	Out-of-cal.			1			
. P. (P-3)	Defective			1			1
. W. Pres.	Out-of-cal.	1		1		1	

DIESEL GENERATOR CALCON PRESSURE SWITCHES

10CFR PART 21 (MAY 1988)

"DEVICES THAT ARE ALREADY INSTALLED AND OPERATING AFTER SEVERAL HOURS BETWEEN TESTS HAVE DEMONSTRATED THEIR RELIABILITY. IMO DELAVAL RECOMMENDS THAT ALL DEVICES NOT INSTALLED, OR THAT ARE INSTALLED BUT HAVE NOT OPERATED FOR SEVERAL HOURS BETWEEN TESTS, BE RETURNED TO IMO DELAVAL FOR REMACHINING, INSPECTION AND TESTING."

10CFR PART 21 (JUNE 8, 1990, ADDENDUM 3)

"OUR RECOMMENDATION OF MAY 12, 1988 MAY HAVE BEEN CONFUSING AND IN LIGHT OF THIS FAILURE AFTER 9 YEARS, IT IS APPROPRIATE TO RESTATE OUR RECOMMENDATION. COOPER INDUSTRIES RECOMMENDS THAT ALL PRESSURE SENSOR DEVICES, COOPER P/N F-573-156, BE MODIFIED OR REPLACED BY DEVICES IDENTIFIED AS CALCON P/N B4400B."

PRESSURE SWITCH FAILURE ANALYSIS RESULTS

<u>ROOT CAUSE</u>

- MIS-INTERPRETATION OF PART 21 (MAY 1988).

<u>CORRECTIVE ACTION</u>

- ALL CALCON PRESSURE SWITCHES HAVE BEEN REPLACED.

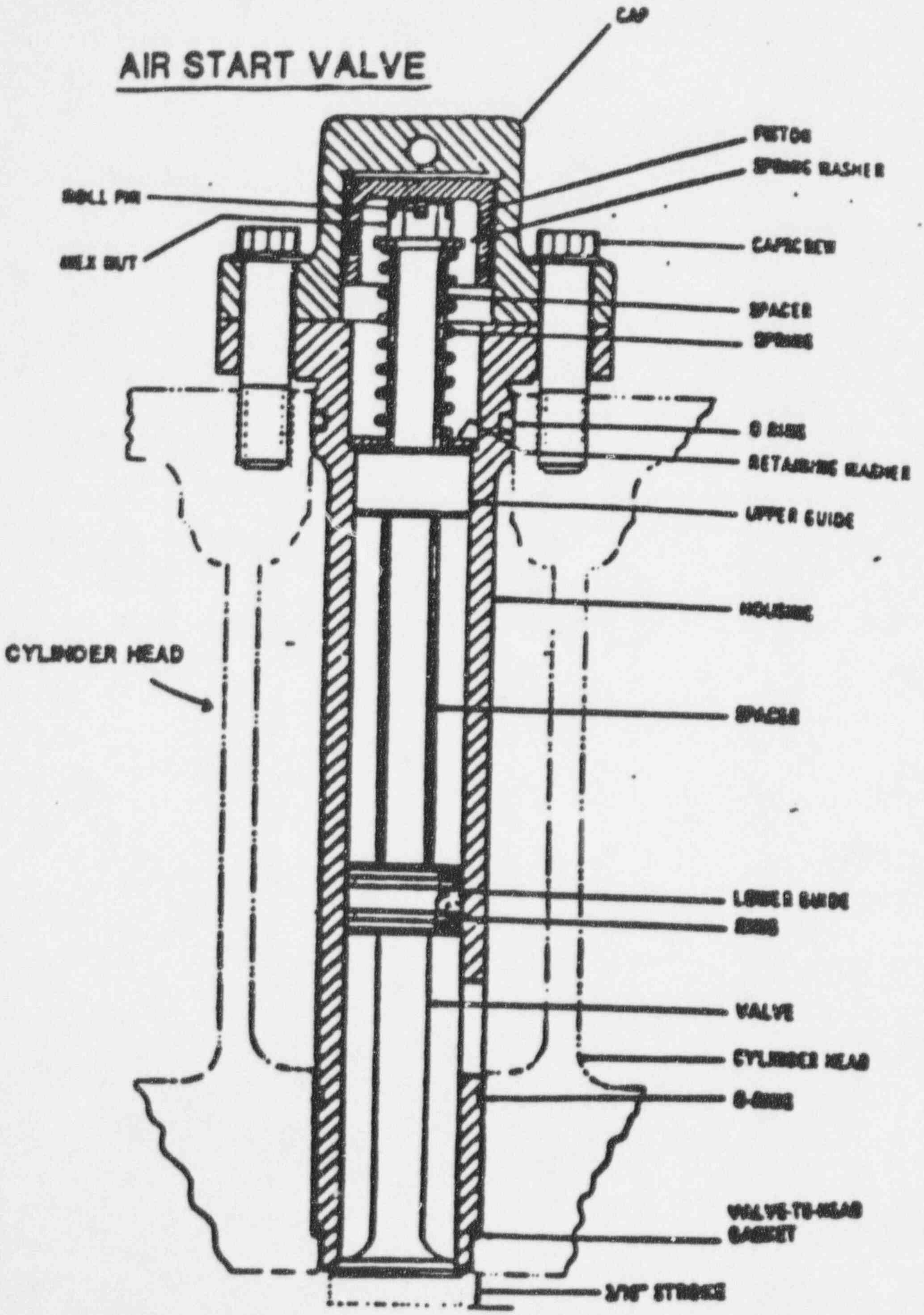
<u>SIGNIFICANCE OF PROBLEM</u>

- STICKING PRESSURE SWITCHES COULD HAVE RESULTED IN REDUCED RELIABILITY OF THE EMERGENCY DIESEL GENERATORS.

<u>CONCLUSION</u>

- AS A RESULT OF THE GPC ROOT CAUSE EVALUATION OF A PRESSURE SWITCH PROBLEM IN 1990, THE VENDOR CLARIFIED AND REISSUED THE 10CFR21 NOTICE.

AIR START VALVE



DIESEL GENERATOR STARTING AIR ADMISSION VALVE PROBLEM

- <u>INDICATION OF AIR START PROBLEMS</u>

  - 1/24/90 - DG 2A ROLLED BUT FAILED TO START DURING ROUTINE SURVEILLANCE TESTING.

  - 1/25/90 - DG 2A START WAS ATTEMPTED. THE ENGINE SLOW ROLLED BUT DID NOT START.

  - 4/12/90 - DG 2A ROLLED BUT FAILED TO START DURING ROUTINE SURVEILLANCE TESTING.

  - 7/5/90 - A SIMILAR EVENT OCCURRED ON DG 1B DURING SURVEILLANCE TESTING.

  - 7/11/90 - DG 2A AGAIN SLOW ROLLED AND FAILED TO START.

STARTING AIR ADMISSION VALVE

RESULTS OF INVESTIGATION

- SEVERAL INDIVIDUAL CYLINDER AIR START VALVES WERE DETERMINED TO BE STICKING IN THE OPEN POSITION.
- SEVERAL VALVE CAPS WERE MACHINED WITH THE BORE SLIGHTLY OVAL-SHAPED AND TAPERED BY SEVERAL MILS.
- SOLUTION WAS TO POLISH EACH PISTON TO PROVIDE APPROXIMATELY 2 - 3 MIL CLEARANCE WITH ITS MATCHED CYLINDER CAP.
- COOPER INDUSTRIES ISSUED A PART 21 REPORT TO THE NRC ON JULY 19, 1990.

Root Cause of Events

- Manufacturing defect in individual air start valves was discovered and resolved by Georgia Power Company.

Corrective Action

- Increase clearance in valves to prevent binding.
- Notify vendor for Part 21 issue.
- Long-term correction not yet determined by vendor.

Significance of Problem

- DG starting reliability was adversely affected by this manufacturing defect.

Root Cause of Events

- Manufacturing defect in individual air start valves was discovered and resolved by Georgia Power Company.

Corrective Action

- Increase clearance in valves to prevent binding.
- Notify vendor for Part 21 issue.
- Long-term correction not yet determined by vendor.

Significance of Problem

- DG starting reliability was adversely affected by this manufacturing defect.

UNITED STATES
U.S. NRC NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555

1990 SEP 13

EDO Principal Correspondence Control

Dir:
DD:
Policy:
Ops:
OI:RI
OI:RII
OI:RIII
OI:RIV
OI:RV
Support
FOIA:
FILE:

FROM: DUE: 10/09/90 EDO CONTROL: 0005838
DOC DT: 09/11/90
FINAL REPLY:

Michael D. Kohn
Counsel to Marvin B. Hobby and
Allen L. Mosbaugh

TO:

Chairman Carr

FOR SIGNATURE OF: ** GRN ** CRC NO: 90-1000

DESC: ROUTING:

2.206 - REQUEST FOR PROCEEDINGS & IMPOSITION OF CIVIL PENALTIES FOR IMPROPERLY TRANSFERRING CONTROL OF GEORGIA POWER CO.'S LICENSES TO THE SONOPCO PROJ. & FOR THE UNSAFE & IMPROPER OPERATION OF GEORGIA POWER CO. LICENSED FACILITIES

Taylor
Sniezek
Thompson
Blaha
Murley, NRR
Lieberman, OE

DATE: 09/17/90

ASSIGNED TO: OGC CONTACT: Scinto

SPECIAL INSTRUCTIONS OR REMARKS:
NRR TO COORDINATE WITH OE.

Copy to Ben Hayes

A1/1

UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
101 MARIETTA STREET, N.W.
ATLANTA, GEORGIA 30323

NOV 0 1 1990

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Request No. RII-90-12

TO: James Y. Vorse, Director
Office of Investigations
Region II Field Office

FROM: Stewart D. Ebneter
Regional Administrator

REQUEST FOR INVESTIGATION

Georgia Power Company	50-424, 50-425
Licensee	Docket Nos.
Vogtle Electric Generating Plant	RII-90-A-0092
Facility	Allegation No.

A. Request

What is the matter that is being requested for investigation? (Be as specific as possible regarding the underlying incident.)

By letter dated September 11, 1990, the law firm of Kohn, Kohn, and Colapinto, 517 Florida Avenue, N.W., Washington, D.C., petitioned the Nuclear Regulatory Commission on behalf of Messrs. Marvin B. Hobby and Allen L. Mosbaugh (petitioners), former employees of Georgia Power Company, "...for proceedings and imposition of civil penalties." A petition entitled "REQUEST FOR PROCEEDINGS AND IMPOSITION OF CIVIL PENALTIES FOR IMPROPERLY TRANSFERRING CONTROL OF GEORGIA POWER COMPANY'S LICENSES TO THE SONOPCO PROJECT AND FOR THE UNSAFE AND IMPROPER OPERATION OF GEORGIA POWER COMPANY LICENSED FACILITIES" was attached to the above referenced letter. This document was received in the Office of the Secretary of the Commission on September 11, 1990, and instructions were provided to process the petition under 10 CFR 2.206.

Contained within the petition under Part 3, Section ^, "SONOPCO intentionally mislead the NRC about the condition of the Vogtle Plant after a Site Area Emergency in order to hasten the restart of the reactor," the petitioners assert that the licensee knowingly provided "false statements intended to mislead the NRC with false assurances about the reliability of the diesel generator whose failure resulted in the Site Area Emergency [on March 20, 1990]." The petitioners further alleged that SONOPCO knew "the

A/142

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diesel generator had actually continued to experience an excessive number of trips, failures and problems similar in nature to the failure which lead to the March 20, 1990 station blackout." The petitioners provided additional information to support their allegation in the petition.

B. Purpose of Investigation

1. What is the basis for the belief that the violation of a regulatory requirement is more likely to have been intentional or to have resulted from careless disregard or reckless indifference than from error or oversight? (Be as specific as possible.)

   There have been several inspections related specifically to the diesel generator start issue performed by Region II inspectors. Those inspections, although not specifically focusing on possible intentional actions by the licensee to mislead the NRC with regard to diesel generator starts, did develop enough information to indicate that there may, in fact, have been a "counting problem" with respect to enumerating the number of starts and defining what actually constituted a valid start for counting purposes. The entire issue was clouded with differing views and opinions among the licensee's staff, enough so that an accurate assessment of the issue could not be established. The staff is concerned that because of the lack of clarity regarding this matter that there may be sufficient confusion, intentional or otherwise, involved in the issue to conceal possible efforts on the part of the licensee to obtain the most favorable position with respect to the NRC. The information provided by the petitioners adds to the confusion surrounding the matter and cannot be discounted because one of the petitioners claims to have first-hand knowledge and actual involvement in the issue.

2. What are the potential regulatory requirements that may have been violated?

   10 CFR 50.9

3. If no violation is suspected, what is the specific regulatory concern?

   N/A

4. Why is an investigation needed for regulatory action and what is the regulatory impact of this matter, if true?

   Due to the claims and counter-claims of the various parties involved in this matter, a formal investigation is the most expeditious means of resolution. If the allegations that the licensee failed to provide accurate and complete information regarding the diesel generator to the NRC are substantiated, legitimate concerns as to the

licensee's ability and determination to operate the facility in a manner consistent with ensuring the public health and safety would be paramount and require significant evaluation and action by the staff. Any substantive action would require a formal investigative basis.

5. In addition to the accuracy of information issues involved with the diesel generator, there are other similar issues relating to the question of the licensee providing accurate information to the NRC during a special team inspection conducted at the facility.

During the period August 6-17, 1990, a special team inspection was conducted at the Vogtle Electric Generating Plant (VEGP) to determine if the licensee operates the facility in accordance with approved procedures and within the requirements and intent of the facility's operating license. The inspection was predicated upon recent activities at VEGP which raised concerns within the NRC as to the licensee's ability and determination to operate the facility in a safe and conservative manner. An aggregation of facts and circumstances associated with operational events and unsubstantiated allegations were viewed as a possible indicator of a non-conservative attitude on the part of the licensee's operating staff which warranted immediate initiation of the special team inspection. The results of this special inspection were detailed in Draft No. 7 of NRC Inspection Report Nos. 50-424/90-19 and 50-425/90-19. The final report, which has not yet been issued, will not include any of the details for items referred to OI or for items included in the 10 CFR 2.206 petition.

In addition to a review of specific technical issues made during the special inspection, the inspection team concluded that during the inspection inaccurate information was received on several occasions from responsible managers and operators on topics well within the scope of their specific responsibility. In five instances the initial information supplied by cognizant licensee representatives was clearly incorrect or inadequately researched. The inspection team concluded that in each of these examples, that licensee officials provided inaccurate, unsworn, oral statements concerning information which concerned topics well within their specific responsibilities and expertise. In the first two cases listed below, the inaccurate information was considered significant to the outcome of the inspection process. Specifically:

a. During a Unit 1 surveillance procedure, the unit shift supervisor (USS) stated, and the operations manager later confirmed, that the containment isolation valves for the hydrogen monitor system were allowed to be opened without entering the LCO action

requirements for Technical Specification (TS) 3.6.3 because the valves received an automatic isolation signal. The inspection identified that these containment isolation valves were remotely-operated, manual valves without automatic isolation signals.

b. The operations manager stated that, after Unit 1 refueling outage 1R2, the modifications to the snubbers were done in conjunction with preplanned system outages which were required for other preventive or corrective maintenance or testing. The inspection identified that few of the snubber modifications were done jointly with pre-planned system outages.

c. The general manager stated that VEGP complied with the corporate position regard ESF actuation reportability. The inspection identified that VEGP did not follow this policy and, in fact, complied with the requirements of 10 CFR 50.72 in that all ESF actuations were reported.

d. The operations manager stated that the shift superintendents (SS) reported directly to the operations manager and that he personally prepared their performance appraisals. The inspection identified that the SSs reported to the unit superintendent (US), and that the US personally prepared the performance appraisals of the SSs.

e. The US indicated that there were no Operations Department actions which were anticipated or required within the first three hours of entering the action statement of TS 3.0.3. The inspection identified that the VEGP management policy and stated practice required preparations for a power reduction, including informing the load dispatcher within the first hour.

C. Requester's Priority

1. Is the priority of the investigation high, normal, or low?

   High

2. What example from Appendix 0517, Part III, does this incident most closely fit, if any?

   4.a.(3)

3. What is the estimated date when the results of the investigation are needed?

   December 20, 1990

4. What is the basis for the date and the impact of not meeting this date? (For example, is there an immediate safety issue that must be addressed or are the results necessary to resolve any ongoing regulatory issue and if so, what actions are dependent on the out- come of the investigation?)

   The date when the results of the investigation are needed is based on the urgent need to have multiple issues involving VEGP resolved on a timely basis. There are currently two other investigations pending that involve activity related to VEGP, as well as the special team inspection. The issue involving the diesel generator is included in the 10 CFR 2.206 petition which is currently pending before the NRC. The outcome of this investigation will directly impact the outcome of the special inspection and the staff's response to the 2.206 petition.

D. Actions by Staff

1. What actions have been taken by the staff (e.g., inspections, Notice of Violation, Enforcement Conferences, Confirmatory Actions Letters, etc.)?

   The special team inspection has been completed and its report is currently pending. The staff has begun preparation of the response to the 2.206 petition. The results of this investigation will be incorporated into the response.

2. Actions to be taken if investigation is closed without a report (based on currently available information).

   Additional inspection will have to be performed to resolve the issues through the inspection process.

E. Contact

1. Staff members: K. Brockman, Region II, Ext. 16299
   A. Herdt, Region II, Ext. 15583
   B. Uryc, Region II, Ext. 14192

2. Alleger identification with address and telephone number if not confidential. (Indicate if any confidential sources are involved and who may be contacted for the identifying details.)

   See petition for identifying data pertaining to the petitioners.

F. Other Relevant Information

Additional details regarding the special team inspection are provided in the draft special inspection report.

Stewart D. Ebneter

Stewart D. Ebneter

Enclosure:
Ltr M. Kohn to K. Carr, 9/11/90,
w/encl as indicated

cc: J. Taylor, EDO
J. Sniezek, DEDR
H. Thompson, DEDS
B. Hayes, OI
J. Lieberman, OE
J. Partlow, NRR
J. Goldberg, OGC

OI RESPONSE TO/STATUS OF REQUEST FOR INVESTIGATION

November 6, 1990

TO: Stewart D. Ebneter
Regional Administrator

FROM: James Y. Vorse, Director
Office of Investigations Field Office, Region II

REQUEST NO.: RII-90-12 DATE OF REQUEST: 11/01/90

Licensee/Vendor/Applicant: Georgia Power Company

Facility or Site Location: Vogtle Electric Genating Plant

Docket No.: 50-424, 50-425 License No.: N/A

CASE INITIATED: [X] CASE NO.: 2-90-020

Date Opened: 11/06/90 Type of Case: (I)

Assist (A)
Inquiry (Q)
Investigation ( )

ECD: 08/91

Case Priority: H

H - High N - Normal L - Low

CASE NOT INITIATED: [ ] REASON: ____________

COMMENTS:

cc: D. Murphy, OI:HQ
J. Weddle, OI:HQ
J. Lieberman, OE

Distribution:
s/f 2-90-020
c/f

OI:RII JR
LRobinson
90/11/06

OI:RII JR
JVorse
90/11/06

A/143

Release

INVESTIGATIVE PLAN

Date November 6, 1990

Case Agent L. Robinson

Case No. 2-90-020

Controlling Office/Requester RA

Review of request & applicable regulations by FOD 11-01-90

Coordination with staff

Coordination with Regional Counsel/OGC NA

Request for additional information from staff

Submission of OI response/status of request for investigation (Case Opening Paper) 11/06/90

Interview of alleger: Confidentiality granted YES(X) NO( ) CS No. OI-90-15
7/18-19/90

Interview of appropriate staff member(s)

Submission of initial monthly investigation status report (ISR) with ECD
11/06/90

Initial discussion with RA regarding ECD and priority of investigation
12-06-90

Review of case file and discussion with case agent:

12/12/90	3/26/91
1/14/91	4/25/91
2/12/91	5/29/91
2/22/91	6/25/91

Discussion of case progress with RA:

12-06-90	4-8-91	8-14-91	1-10-92	6-12-92
1-07-91	5-13-91	9-16-91	2-10-92	7-10-92
2-11-91	6-12-91	10-10-91	3-11-92	
3-[illegible]-91	7-8-91	11-15-91	4-14-92	
		12-05-91	5-15-92	

Submission of monthly ISR:

11/30/90
12-31-90
01/31/91

Date field work completed:

Date draft report recv'd by FOD:

Date final report signed by FOD and/or forwarded to HQ for review:

A/144
A/2

OIMIS DATA ENTRY SHEET

	CASE NUMBER	2-90-020
01	DOCKET NUMBER	50-424, 50-425
03	FACILITY	Vogtle Elec. Gen. Plant
04	TYPE OF CASE	
05	DATE OPENED (omit /)	11/06/90
06	ESTIMATED COMPLETION DATE	08/91
07	REQUESTED BY	RA
08	STATUS	

10 ALLEGATION (2 lines max)
ALLEGED FALSE STATEMENTS REGARDING OPERABILITY OF DIESEL GENERATOR

123456789012345678901234567890123456789012345678901234567890

11 COMMENTS/REMARKS (10 lines max)

123456789012345678901234567890123456789012345678901234567890

12	CASE AGENT	Robinson
13	CATEGORY	WR
15	STATUTE DATE	
16	CLOSED DATE	
18	ISSUED DATE	
19	DOJ REFERRAL	
20	DOJ ACTION	
22	RELATED CASES	
23	FOIA NUMBER	
26	PRIORITY	H
27	SOURCE OF ALLEGATION	A
28	VENDOR CODE	

A/145

Release