Florida Power
CORPORATION

April 30, 1990
3F0490-13

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

Subject: Crystal River Unit 3
Docket No. 50-302
Operating License No. DPR-72
Licensee Event Report No. 89-039-01

Dear Sir:

Enclosed is a supplement to Licensee Event Report (LER) 89-039 which was previously submitted in accordance with 10 CFR 50.73. This supplement is being submitted to document four nonconformances considered to be additional examples of the deficiency identified in the previous Licensee Event Report.

Should there be any questions, please contact this office.

Sincerely,

Gary Boldt

G. L. Boldt
Vice President, Nuclear Production

WLR:mag

Enclosure

xc: Regional Administrator, Region II
Senior Resident Inspector

IE22
1/1

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A Florida Progress Company

NRC FORM 366 (6-89) U.S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
CRYSTAL RIVER UNIT 3	0 5 0 0 0 3 0 2	1 OF 1 1

TITLE (4) Personnel Error Results in Operation Outside the 10 CFR 50 Appendix R Separation Design Basis

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 9	9 0	8 9	0 3 9	0 1	0 4	3 0	9 0	N/A	0 5 0 0 0
									N/A	0 5 0 0 0

OPERATING MODE (9): 5

POWER LEVEL (10): 0 0 0

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more of the following) (11)

[ ] 20.402(b)	[ ] 20.405(c)	[ ] 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)	[ ] 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)	[X] 50.73(a)(2)(ii)	[ ] 50.73(a)(2)(viii)(B)	
[ ] 20.405(a)(1)(v)	[ ] 50.73(a)(2)(iii)	[ ] 50.73(a)(2)(x)	

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (AREA CODE)
L. W. MOFFATT, NUCLEAR SAFETY SUPERVISOR	9 0 4 7 9 5 - 6 4 8 6

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 November 7, 1989, Crystal River Unit 3 was in MODE 1, POWER OPERATION. At 1515, it was determined control circuits for two Make-up valves did not meet the separation criteria required by 10CFR50 Appendix R. This event was caused by cognitive personnel error. The valves and control circuits were operable at the time of the event and no equipment failures occurred. A roving fire watch patrol was confirmed in effect. The two deficient circuits will be modified to meet separation criteria. A review of modifications subsequent to the original 10CFR50 Appendix R design will be performed to assure the separation criteria are being met. On March 29, 1990, Crystal River Unit 3 was in MODE 5, COLD SHUTDOWN, for a scheduled refueling and maintenance outage. At about 1400, four 10CFR50 Appendix R non-conformances were identified to the shift supervisor. These four non-conformances are considered to be additional examples of the deficiency identified in the previous Licensee Event Report. All the affected equipment was operable at the time of the discovery, or out of service for routine, scheduled maintenance as part of the outage. No equipment failures occurred as a result of this event. Relevant Engineering Procedures have been revised to provide additional guidance to the design engineers. The Nuclear Engineering organization will develop a training program for personnel writing or reviewing modifications that invoke 10CFR50 Appendix R design criteria.

EVENT DESCRIPTION

On November 7, 1989, Crystal River Unit 3 was in MODE 1, POWER OPERATION, at 855 MWe. The control circuits for two valves [ISV] in the Make-up System [CB] (MUV-40 and MUV-41) that isolate Reactor Coolant System (RCS) [AB] letdown are located on the 124' elevation of the Control Complex [NA]. These two valves, both inside the Reactor Building [NH], comprise one train of letdown isolation (see Figure 1). The valves are part of the equipment required for safe shutdown of CR-3 in the 10CFR50 Appendix R design for a fire on the 124' elevation of the Control Complex. The redundant train is the single isolation valve outside the Reactor Building. For the design fire on the 124' elevation, the outside isolation valve control circuitry is assumed failed. At 1515, it was determined the control circuits for the two inside valves did not meet 10CFR50 Appendix R separation criteria from the outside valve. The valves and control circuits were operable at the time of the event and no equipment failures occurred.

On March 12, 1990, Florida Power Corporation (FPC) was notified by Gilbert-Commonwealth Engineers that four additional items have been identified by the partially completed review initiated as corrective action for the initial nonconformances. FPC engineers reviewed the Gilbert findings and identified the deficiencies as a Non-Conformance on March 29, 1990. All of the equipment was operable at the time of the event and no equipment failures occurred.

On March 29, 1990, Crystal River Unit 3 was in MODE 5, COLD SHUTDOWN, for a scheduled refueling and maintenance outage. At about 1400, four 10CFR50 Appendix R non-conformances were identified to the shift supervisor. These four non-conformances are considered to be additional examples of the deficiency identified in the previous Licensee Event Report. The failure to meet the 10CFR50 Appendix R separation criteria is considered to be an operation outside the design basis for the plant and reportable in accordance with 10CFR50.72(b)(2)(i). A four hour verbal report was made at 1415 on March 29th and assigned NRC event number 18098. This report is being submitted in accordance with 10CFR50.73(a)(2)(ii)(B).

CAUSE

These six circuits were design modified to comply with the requirements of 10CFR50, Appendix R separation criteria. Despite this effort, the required separation was not achieved by the design change. The failure to implement the required separation was caused by cognitive personnel error by utility and contract engineers. The designer and reviewers failed to assure the design met its intended function.

EVENT EVALUATION

If a fire had disabled these circuits, then letdown flow could not have been isolated. The 10CFR50 Appendix R design assumed letdown flow would be isolated. The consequences of the failure would be that the valves remain open and letdown flow would be higher than assumed. This would require the make up flow to be higher, but still well within the capacity of the pump.

The greatest potential for impact on the ability to safely shut down the plant would have been the inability to isolate the flow of water into the Make-up System [CB] tankage, (i.e. Make-up Tank [CB,TK], R.C. Bleed Tanks [WD,TK] in the Auxiliary Building. The make up capacity, coming from the Borated Water Storage Tank [BP,TK] (capacity 420,000 gal.), would have been sufficient to maintain cooling water to the core [AC]. However, the letdown flow would not be recirculating to the core and it would quickly overfill the Make-up Tank (capacity 4,488 gal.) or more slowly overfill the R.C. Bleed Tank (capacity 75,000 gal). This would then cause relief valves to lift, spilling untreated reactor coolant liquid onto the Auxiliary Building floor. This condition would create a radiological hazard in the Auxiliary Building and complicate shutting down the unit, even after the fire was extinguished. The possibility of this occurring is considered extremely remote. The postulated fire which would disable the control circuits for MUV-40, MUV-41 and MUV-49, would be located in the control complex while the valves are located in other buildings. Letdown flow could be isolated by manual operation of these valves.

Each of the four newly discovered deficient items is described below and an evaluation of its impact on the plant is presented.

Item 1

Loss of SWP-1A, SWP-1B and RWP-2B

Each of the listed components (SWP-1A[CC,P], SWP-1B [CC,P] and RWP-2B [KG,P]) utilizes contacts from relays located in the CRD/Relay Room in their breaker closing control circuits. There are several fire areas on the 124' elevation of the control complex, shown on Figure 2. Each of the switchgear rooms and each of the Emergency Feedwater Initiation and Control (EFIC) [BA] rooms constitute a separate fire area. The remaining area, the corridors and CRD/Relay Room, shown shaded on Figure 2, constitute the seventh area. The Fire Study assumes that a fire in this shaded area would disable all equipment in that area. The relay circuits are presently listed as "Fail Safe" and thus not impact the pump operability. However, the current review revealed that certain hot shorts in these circuits could blow the fuse in the breaker closing control circuit rendering the pumps inoperable from the Main Control Room.

In the event of a concurrent loss of offsite power, RWP-2B [KG,P] and SWP-1B [CC,P] are considered the only reliable pumps because there is a potential for losing the "A" EDG [EK,DG] due to fire damage to relays and circuits.

Control of equipment required to shut down the plant might not be available from the main control room for a fire in the shaded area of Figure 2. This does not put the plant into an unrecoverable condition because the pumps could be started locally from the 4160 volt ESB switchgear.

Item 2

Loss of "B" EDG

Figure 3 shows the Auxiliary Building 95' elevation. Most of this elevation is treated as a single fire area. This area includes the seawater room that contains both the A and B trains of the Emergency Nuclear Services Closed Cycle Cooling Water Pumps (SW pumps [CC,P]) and the Nuclear Services Seawater Pumps (RW pumps [KG,P]). It was assumed that one train would be operable, opposite the end of the room that the fire was in. This was acceptable because a fire on the 95' elevation of the Auxiliary Building would not affect the Emergency Diesel Generators (EDG's). The current review found a circuit for the power feed to ES MCC 3B1, which powers the fuel oil transfer pump for the "B" EDG, being routed through the seawater room. The fire could also disable the "B" train battery charger, making the DC powered backup fuel oil transfer pump unreliable.

This creates a situation where it is possible to postulate the loss of the "A" SW and RW pumps and the "B" EDG, a condition where it is not completely assured that at least one train of safe shutdown equipment is available in a common fire area.

Compensatory operator actions can prevent this situation from degenerating to an unrecoverable condition. The amount of fuel in the day tank for the "B" EDG allows enough time for the operators to cross-tie the ES 4160 volt busses. This evolution involves stripping all non-safe shutdown loads from the busses. This would allow the "B" SW and RW pumps to be powered by the "A" EDG.

Item 3

Loss of Pressurizer Level Indication, Reactor Coolant System Temperature Indication and Steam Generator Pressure Indication

This concern is not applicable to the present plant configuration. During the time interval from the completion of the original Appendix R implementation until the installation of the Regulatory Guide 1.97 indicators a condition existed where the listed indications could be lost.

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

The circuits for the Train 1 indicators, RC-1-LT1, RC-4A-TE1 and SP-6A-PT1, are routed through the 4160 volt and 480 volt "B" ES switchgear rooms. See Figures 4 and 2 for layout showing the ES switchgear rooms. Since the indicators listed are Train 1 and the fire is in the "B" ES switchgear rooms, the Train 2 indicators must be assumed lost due to loss of all "B" power. Thus, both redundant trains of indication are lost.

The Regulatory Guide 1.97 indicators of these parameters are not affected by a fire in the "B" ES switchgear rooms. The operators would not lose all indication in the Main Control Room since the Regulatory Guide 1.97 indications were installed.

Item 4

Spurious Actuation of Breakers 3207, 3208, 3211 and 3212

The control circuits for these breakers are not provided with isolation contacts from the Remote Shutdown Isolation Relays to isolate the Main Control Room and Cable Spreading Room circuits in the event of a fire in the Control Room/Cable Spreading Room fire area.

Spurious closure of these breakers could result in loss of the diesel generators or the normal off-site power source. This would complicate the shutdown while power sources were re-aligned.

CORRECTIVE ACTION

It was immediately confirmed that a roving fire watch patrol was in effect. This was considered to be an effective fire protection measure which would reduce the potential for a major fire and the actions required by the 10CFR50 Appendix R fire study. The roving fire watch and the action listed for each item were considered adequate compensatory measures until the deficient circuits could be modified. The six deficient circuits will be modified to meet the 10CFR50 Appendix R separation criteria.

The Crystal River Unit 3 10CFR50 Appendix R design development was completed in 1985. Since then, relevant Engineering Procedures have been revised to provide additional guidance, regarding 10CFR50 Appendix R criteria, to the design engineers. A review of all modifications developed subsequent to installation of 10CFR50 Appendix R modifications in 1985 will be performed to assure all the 10CFR50 Appendix R separation criteria are being met.

In addition to these actions, the company is taking additional actions in response to an internal audit. The Nuclear Engineering organization will develop a training program for personnel writing or reviewing modifications that invoke 10CFR50 Appendix R design criteria. The additional information in the Engineering procedures, along with the training program, should preclude a repetition of this event.

PREVIOUS SIMILAR EVENTS

There have been eight previous events caused by cognitive personnel error that led to an operation outside design basis. However these were all discovered subsequent to the design of this circuitry. The corrective action to those events would not have prevented this error.

NRC FORM 366A
(6-89)

U.S. NUCLEAR REGULATORY COMMISSION

APPROVED OMB NO. 3150-0104
EXPIRES 4/30/92

ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS INFORMATION COLLECTION REQUEST: 50.0 HRS. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE RECORDS AND REPORTS MANAGEMENT BRANCH (P-530), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555, AND TO THE PAPERWORK REDUCTION PROJECT (3150-0104), OFFICE OF MANAGEMENT AND BUDGET, WASHINGTON, DC 20503

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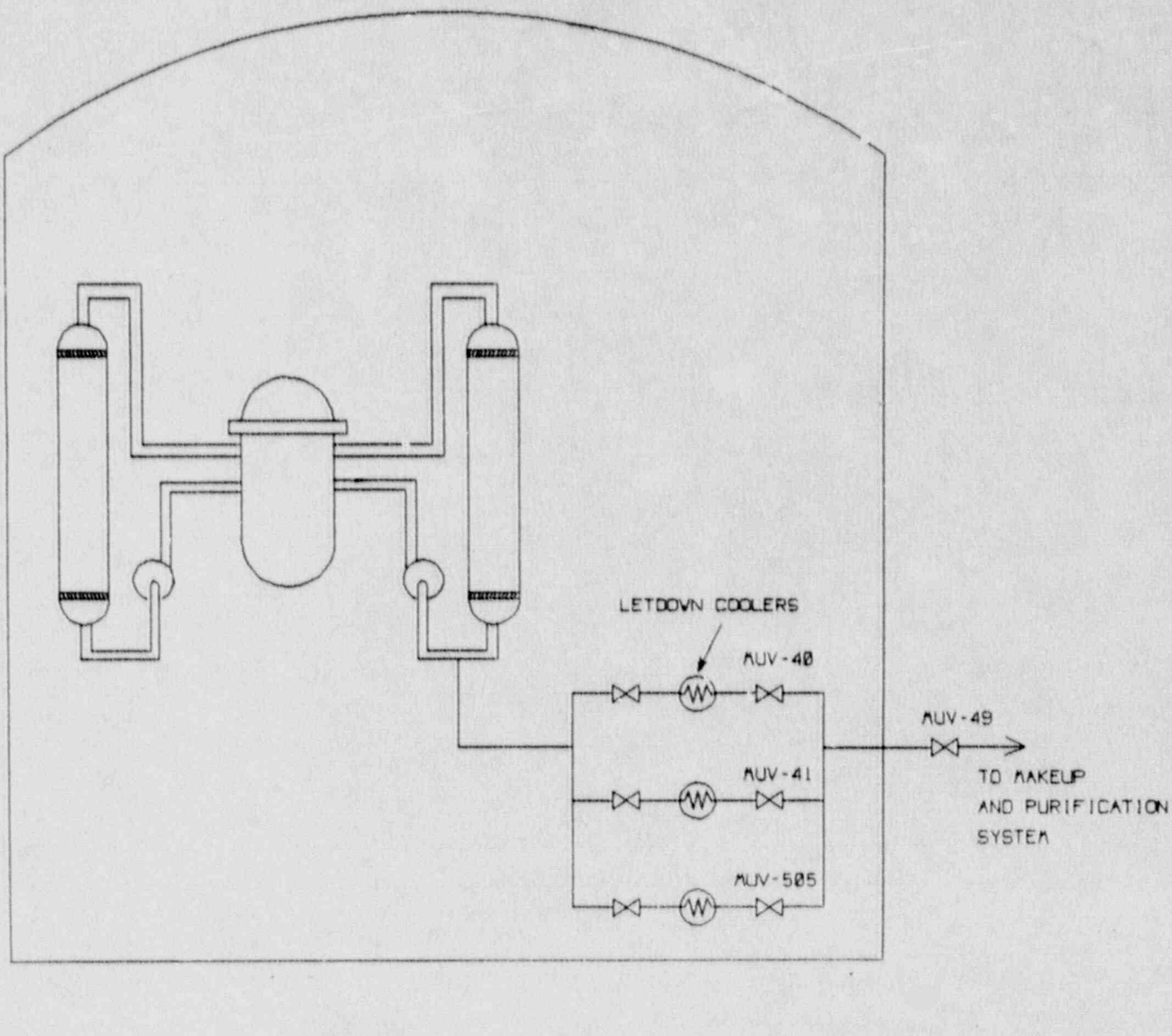
TEXT (If more space is required, use additional NRC Form 366A's) (17)

FIGURE 1



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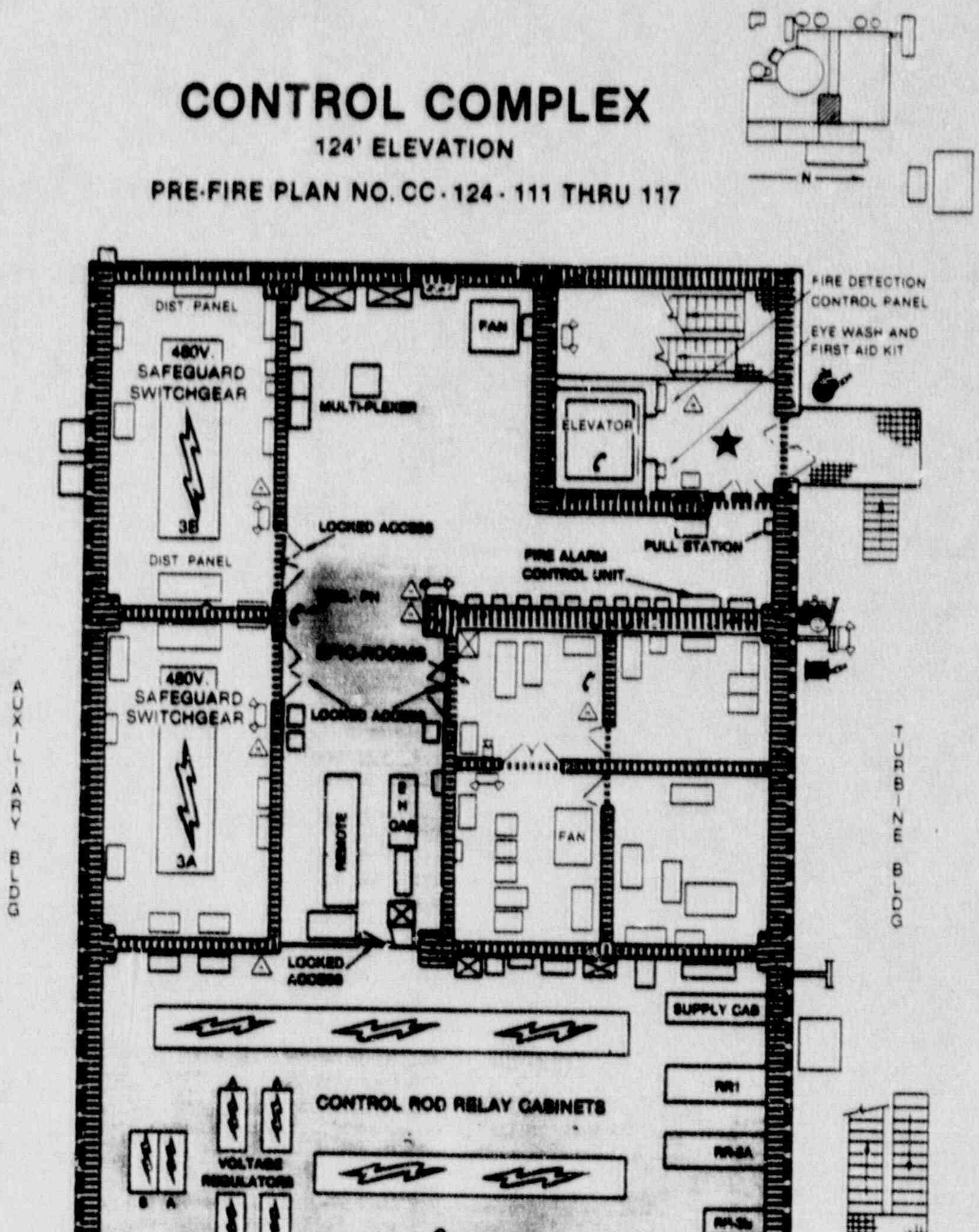
TEXT (If more space is required, use additional NRC Form 366A's) (17)

FIGURE 2

ITEM 1
SWP-1A
SWP-1B
RWP-2B
EDG 'A'



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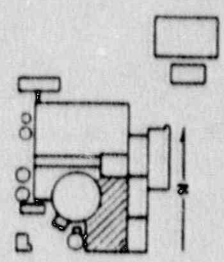
TEXT (If more space is required, use additional NRC Form 366A's) (17)

FIGURE 3

REV. 02
12-31-88

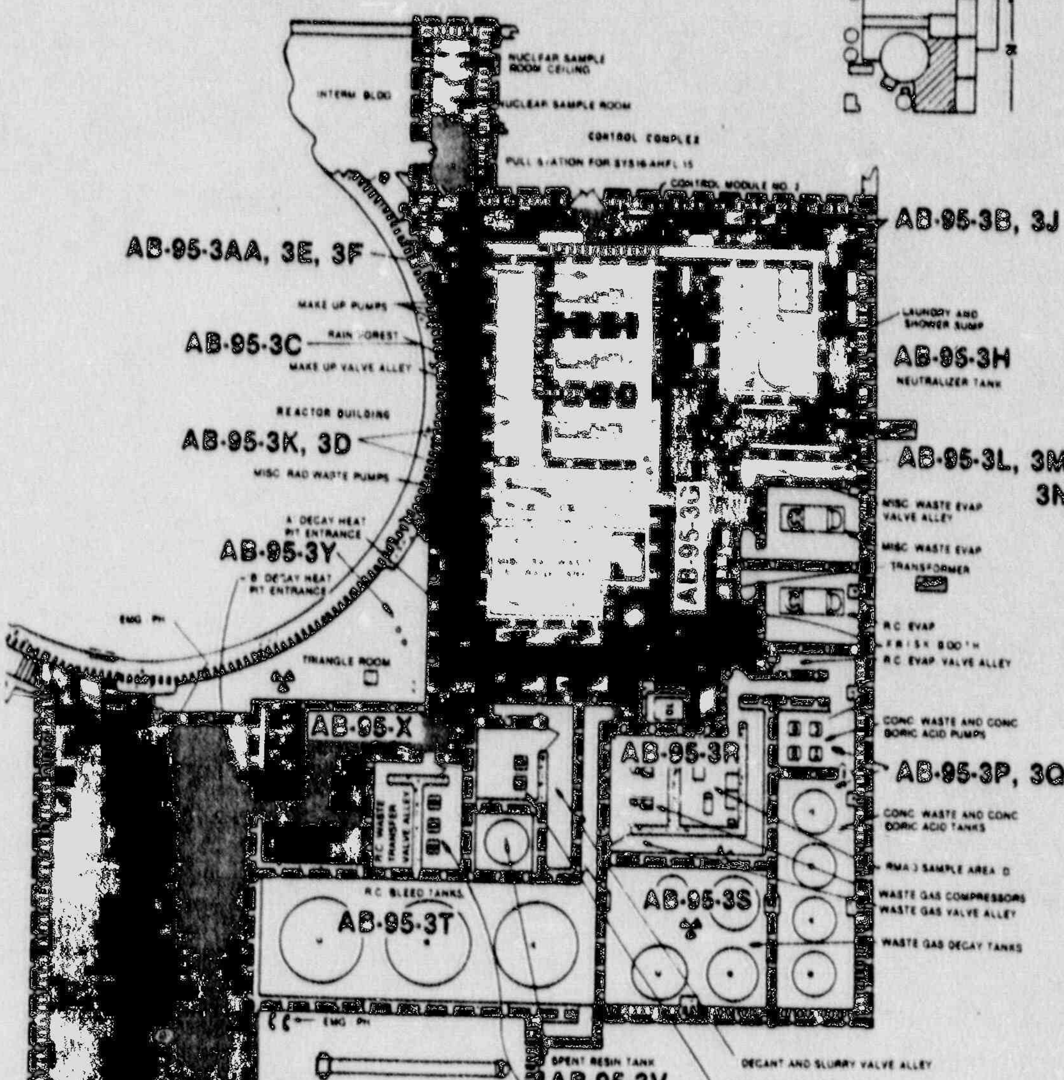


AUXILIARY BUILDING

95' ELEVATION

DEVELOPED BY THE CR-3 FIRE PROTECTION STAFF

ITEM 2
EDG 3B



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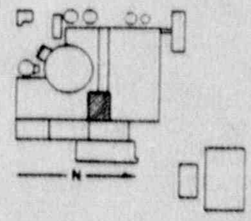
TEXT (If more space is required, use additional NRC Form 366A's) (17)

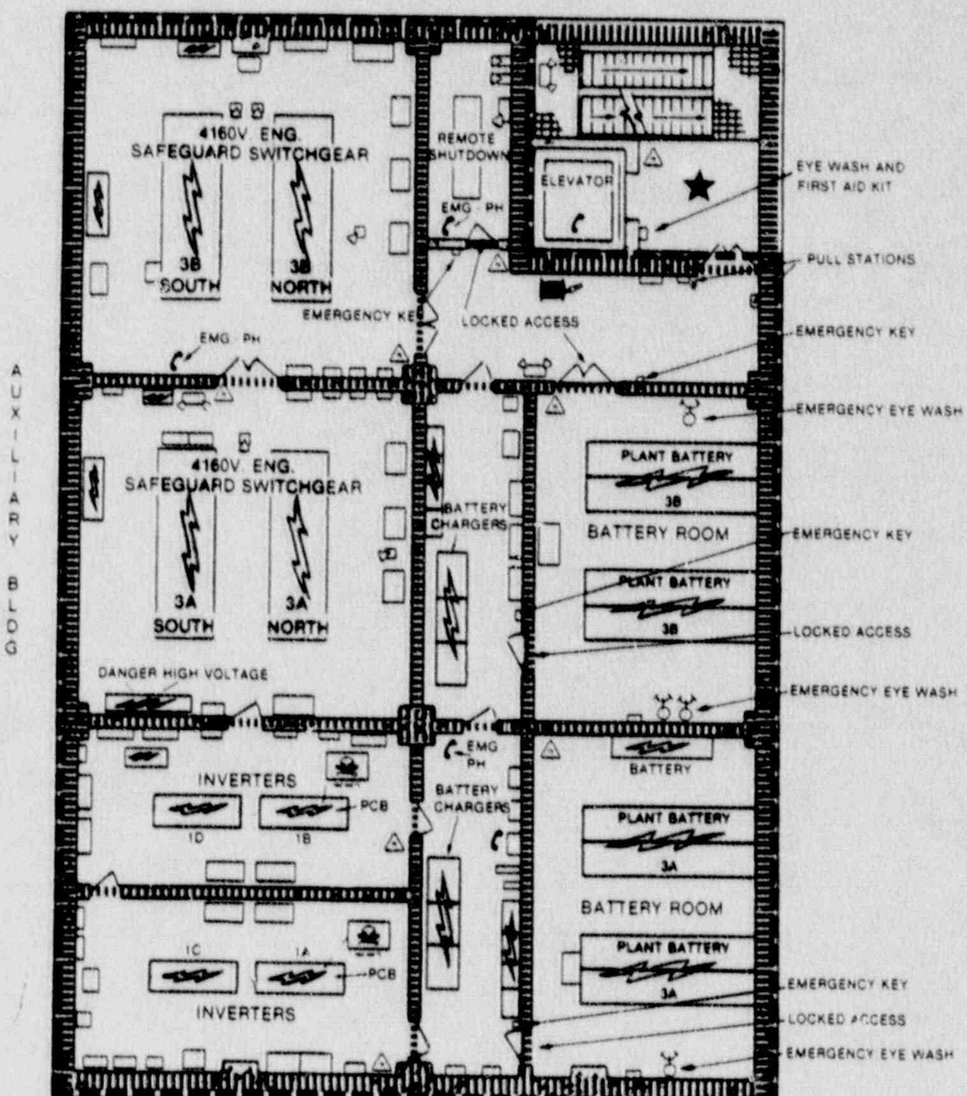
FIGURE 4

CONTROL COMPLEX

108' ELEVATION

PRE-FIRE PLAN NO. CC - 108 - 102 THRU 110





NOTE * PULL STATION TO ACTUATE HOSE REEL

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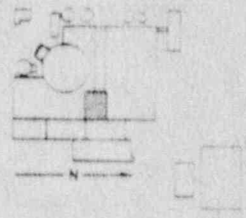
TEXT (If more space is required, use additional NRC Form 366A's) (17)

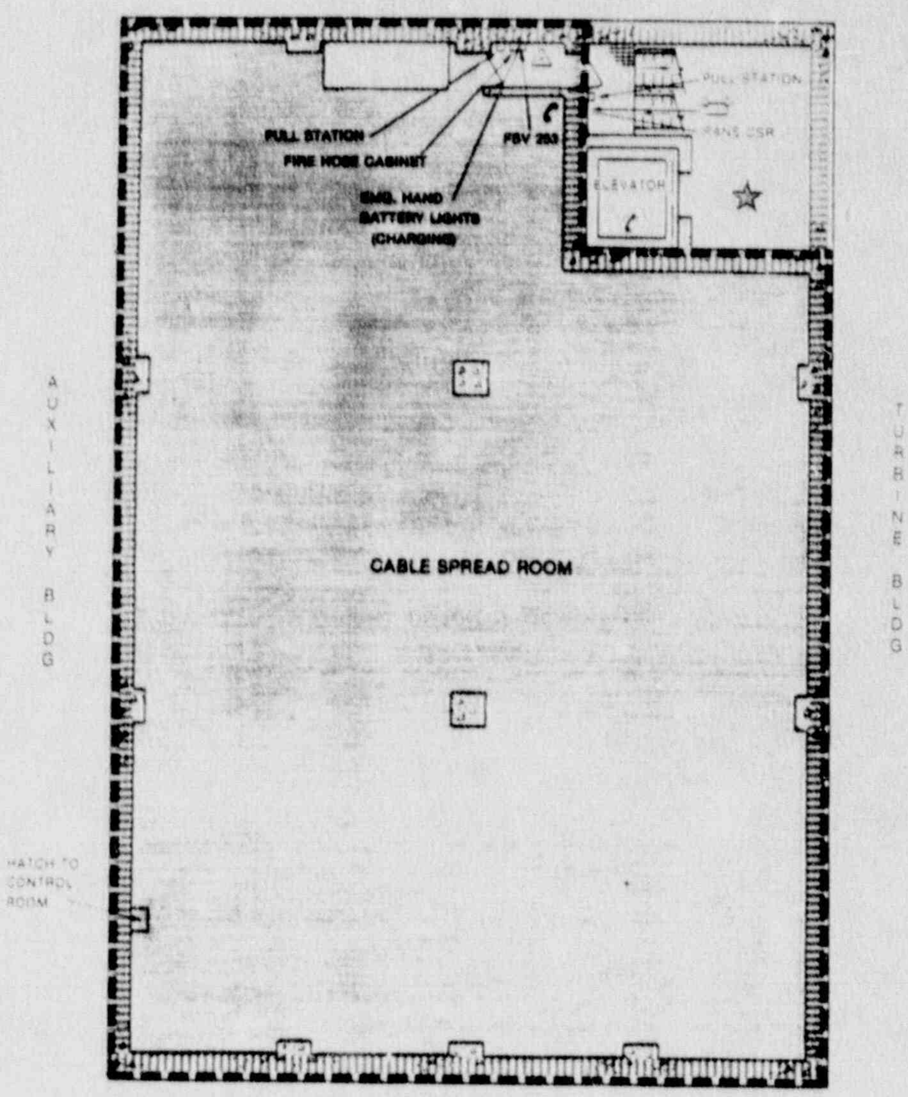
FIGURE 5

CONTROL COMPLEX

134' ELEVATION

PRE-FIRE PLAN NO. CC - 134 - 118A





NOTE

DENOTES HALON SUPPRESSION SYSTEM

COMMAND POST LOCATED ON ELEVATOR LANDING 124' ELEVATION

ALTERNATE HOSE REEL ON [illegible] EL. HEATER BAY