September 25, 1996

TO: Secretary of the Commission
U.S. Nuclear Regulatory Commission
Washington D.C. 20555

Dear Sir:

Per the letter dated September 12, 1996 I am requesting a hearing IAW 10 CFR 2.103(b)(2).

Enclosed you will find:

1. A copy of the original request and documentation.
2. A copy of the denial dated September 12, 1996.
3. My reply to the above dated letter.

Sincerely,

Ralph L. Tetrick

Ralph L. Tetrick
18990 SW 270 Street
Homestead, FL 33031
Docket No. 55-20726

July 30, 1996

TO: Director

Division of Reactor Controls and Human Factors
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission, Washington, DC 20555

Dear Sir:

Per the letter sent to me on July 19, 1996 I am requesting an informal review of my written examination. Enclosed you will find copies of the questions I wish to be reviewed along with supporting documentation.

Sincerely,

Ralph L. Tetrick
18990 SW 270 Street
Homestead, FL 33031
Docket No. 55-20726

SRO QUESTION 24

Which ONE of the following describes the Spent Fuel Pool Cooling (SFPC) system basic operation and connections to the Spent Fuel Pool (SFP)?

The SFPC pumps normally take a suction on the:

ANSWER:

a. - "High" line near the top of the SFP and discharge through a line 1 foot below the top with a 1/2 inch siphon break hole 6 inches below the water level.

REFERENCE:

SD-041, Fuel Pool Cooling, Purification and Ventilation System' page 16. E.O. OF LP 6902141

COMMENT:

Answer (A) is partically incorrect because, (1) the discharge line is routed 10 inches below nominal water level and extends to the middle of the pool (ie 20 feet from top and bottom) and (2) the siphon break is 14 inches below nominal water level not 6 inches.
Answer (C) is partically incorrect because, (1) the "High" suction line is approximately 3 1/2 feet below the nominal water level and (2) the discharge line is as stated above.
Both answers A and C are equally correct because they indicate the suction is from the high line and that there is a siphon break in the top of the discharge line. An answer of A or C indicates the operator is aware of the design requirement to prevent inadvertant draining of the SFP.

RECOMMENDATION:

Accept answer c as an additional correct answer sence both a and c are partically incorrect.

C. - "High" line 1 foot below the top of the SFP and discharge through a line at thr bottom of the SFP with a 1/2 inch siphon break hole 6 inches below the water level.

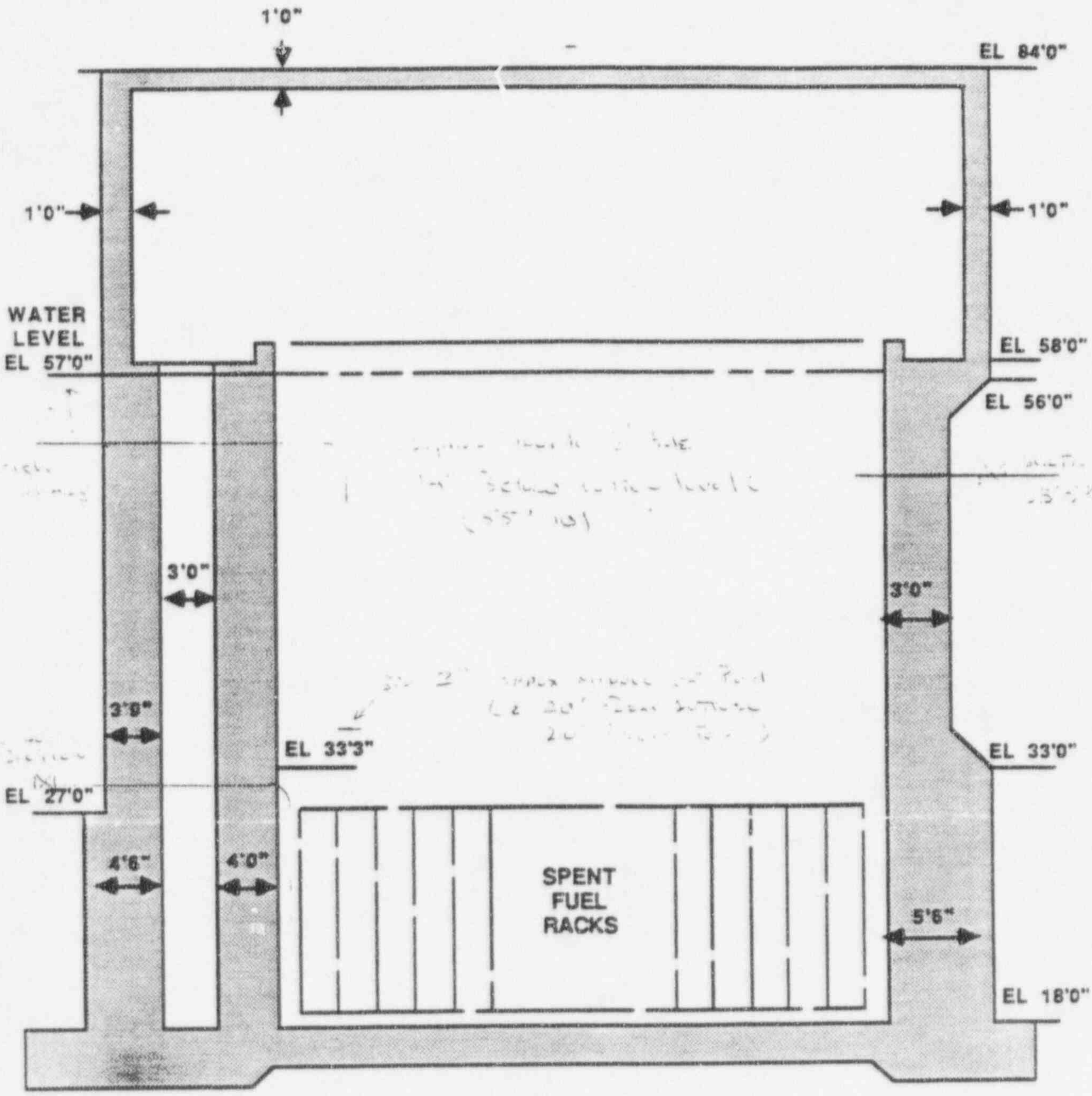
There is a thermal expansion loop in the piping on the discharge of the Goulds SFP cooling pump to accommodate thermal stresses due to pool boiling at 212°F.

Spent Fuel Pool Cooling Pumps

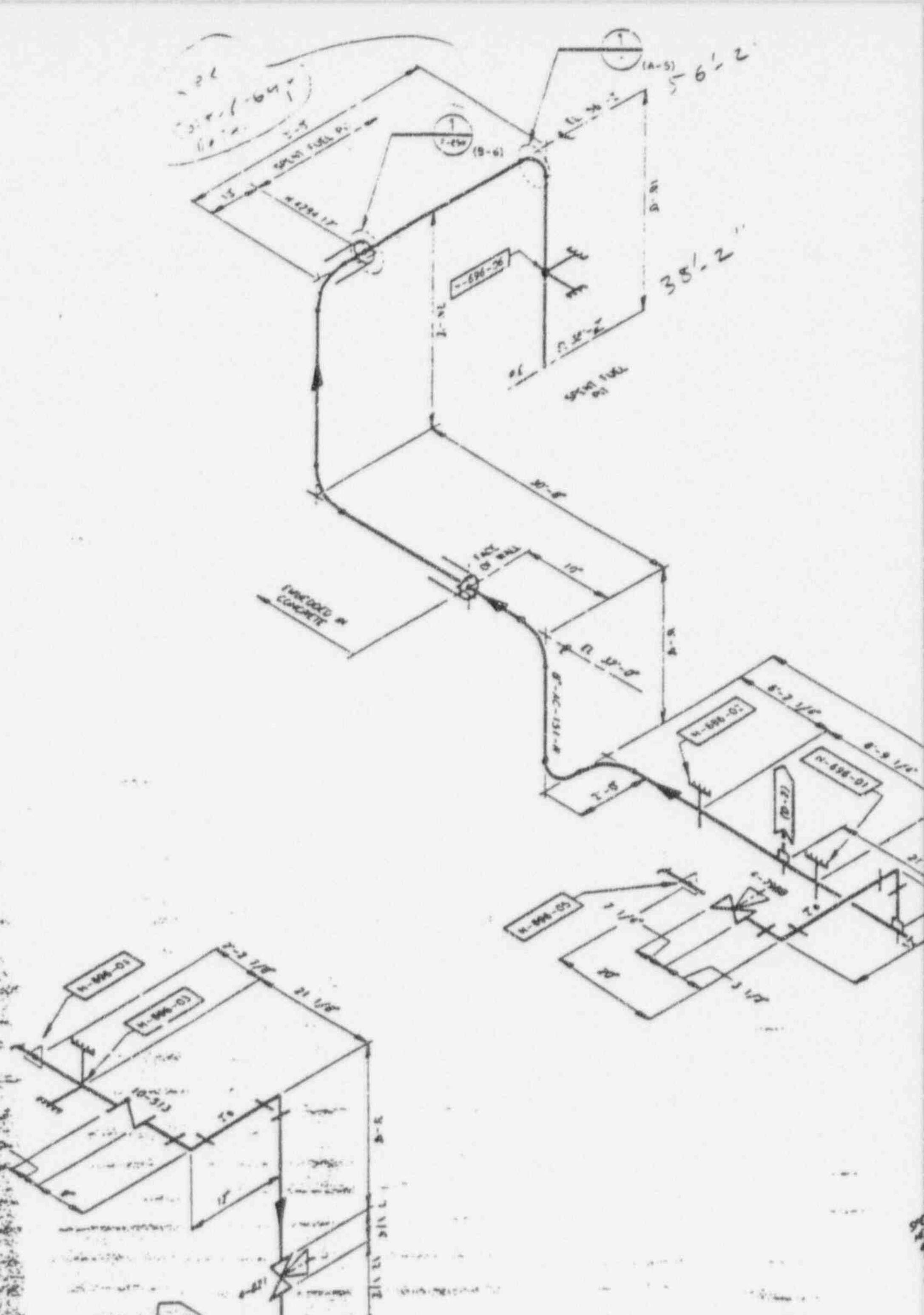
Three SFP cooling pumps are provided, A, B and emergency. Refer to Figure 10. A & B pumps are horizontal centrifugal pumps rated for 2300 gpm at 125 ft. TDH. Additionally, both pumps are powered from LC C (breaker 0309) via manual transfer panel P-16. There are two switches located on this panel; One switch is for pump A and the other switch is for pump B. They are interlocked such that only one switch can be closed at a time. The pumps are located in the SFP heat exchanger room and are controlled locally.

The third pump, emergency SFP cooling pump is also provided. It is used only when the SFP cooling pumps are not available. Power for this pump is provided by a receptacle in the cask wash area new fuel room. When the emergency pump is used, the SFP purification loop is bypassed. The Emergency Spent Fuel Pit Pump Motor is not normally connected to a permanent 480V AC power source. If its use is required the temporary local motor starter/disconnect stand and attached cables needs to be moved to outside of the Spent Fuel Pump Room and the load side cable connected to the Emergency Spent Fuel Pump Motor. The temporary motor starter/disconnect stand is normally stored in the new fuel storage room when not in use. The temporary motor/starter disconnect may be connected to provide standby operations at PS-N discretion.

The SFP cooling pumps can take a suction on the SFP through the high suction valve (796) or the low suction valve (797). The high suction line penetrates the SFP near the top and terminates. The low suction line penetrates at a level 6' above the top of the fuel assemblies and extends downward to almost the bottom of the SFP. Complete siphon draining of the pit by a break in this line is prevented by a normally locked closed valve located at the same elevation as the penetration (797). There are no other connections provided on the SFP. The cooling loop discharge line penetrates the SFP at approximately 1' below the top and extends straight down towards the stored fuel. A 1/2" hole is drilled in the discharge line at approximately 6" below the water surface, it acts as a siphon breaker.

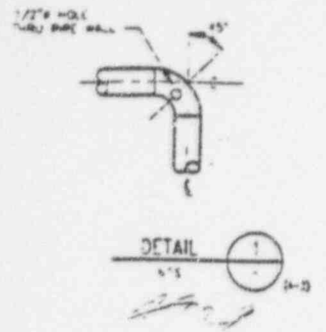


SPENT FUEL POOL



REV	DATE	REVISION	BY	CH	APP	APP	REV	DATE	
							D		ISSUED FOR PC/M 85-148 AND

1 2 3 4



		DIRECTION OF STEM
SNUBBER	DOUBLE ACTING	SPRING HANGER
ANCHOR	GROUTED ANCHOR	WELDED STANCHION
	SUPPORT MARK NO.	ANALYSIS NODE NO.

GLOBAL COORDINATE SYSTEM

NORTH

REFERENCE OPERATING DIAGRAM 5610-T-E-4515, SH. 1

ANSTEC APERTURE CARD

Also Available on Aperture Card

PIPE MATERIAL	LINE SPEC	LINE SIZE	INSULATION
A312 TP 304	8"-AC-151-R	8"SCH 10S SMLS	NONE
A312 TP 304		2"SCH 10S SMLS	NONE

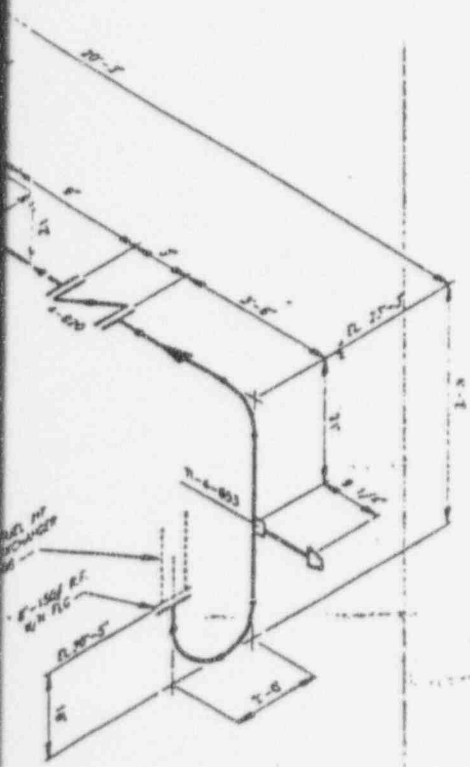
SUC 40 [illegible]

VALVE WEIGHTS:

VALVE TAG NO.	WT	VALVE TAG NO.	WT
4-798B	15#	4-871	15#
4-820	75#	10-313	16#

NOTES:

1) ALL ELBOWS ASSUMED LONG RADIUS UNLESS OTHERWISE SPECIFIED
2) FOR PIPE SUPPORT DETAILS SEE DRAWING SERIES 5614-H-696
3) SYSTEM DESIGN PRESSURE 150 PSIG
   DESIGN TEMPERATURE 200°F
   PEAK PRESSURE 70 PSIG.
   OPERATING TEMPERATURE 112°F
   MAX. OPERATING TEMPERATURE 212°F



...TION ONLY

NUCLEAR SAFETY RELATED

NOTE: THIS DWG IS MADE FROM
VENDOR DWG. NO. [illegible]
A/E DWG. NO. [illegible]

TURKEY POINT NUCLEAR UNIT 4

PIPING ISOMETRIC

AUXILIARY BUILDING
SPENT FUEL PIT COOLING SYSTEM

FPL

BECHTEL

DRAWING NUMBER 5614-P-696

SYS 033

REV 0

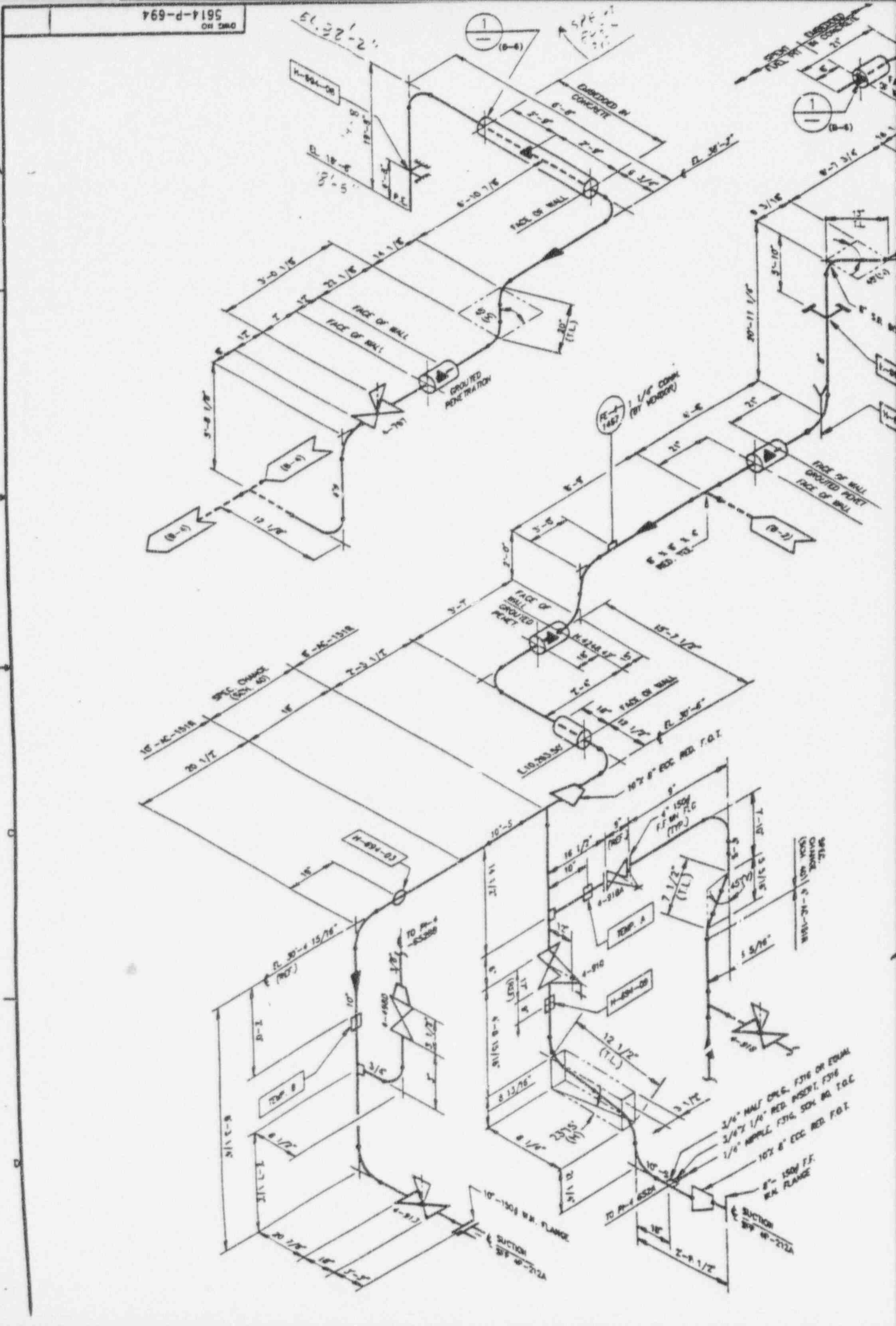
...ED INTO THE FPL SYSTEM

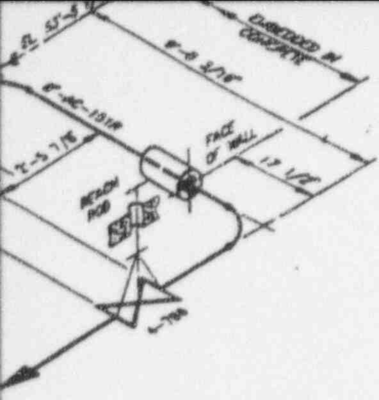
REVISION

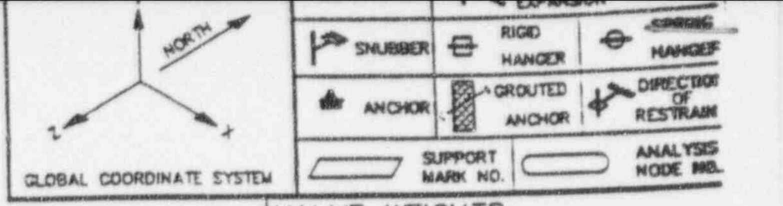
BY | CH | APP | APP

9-3-97

9610250070-01

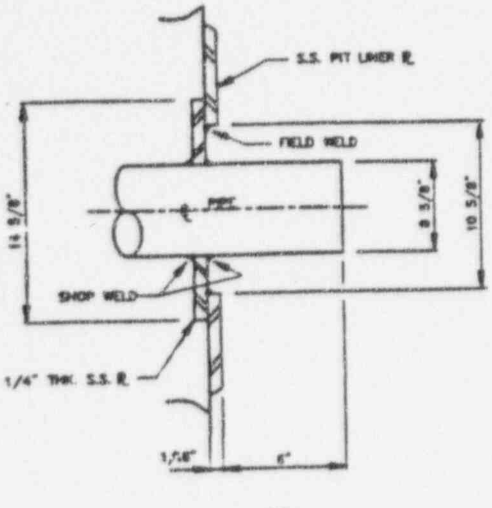

DWG NO
5614-P-694
EMBEDDED IN CONCRETE
FACE OF WALL
GROUTED PENETRATION
FE-4 1467
1 1/2" CONN. (BY VENDOR)
FACE OF WALL GROUTED PICKET
H-694-08
H-694-03
H-694-09
H-4246.47
SPEC. CHANGE (SCH. 40)
10"-HC-1518
8"-HC-1518
10"x 8" ECC RED. F.O.T.
4"-150# F.F. WN FLG (TYP.)
TEMP. A
TEMP. B
TO PH-4 -6529B
TO PH-4 6526
EL 30'-4 15/16" (REF.)
EL 30'-6"
EL 38'-3"
3/4" HALF CPLG. F316 OR EQUAL
3/4"x 1/4" RED. INSERT, F316
1/4" NIPPLE, F316, SCH. 80, T.O.E.
10"x 8" ECC RED. F.O.T.
8"- 150# F.F. W.N. FLANGE
10"-150# W.N. FLANGE
℄ SUCTION SFP 4P-212A
4-910
4-918A
4-912
4-4980
4-797





VALVE WEIGHTS:

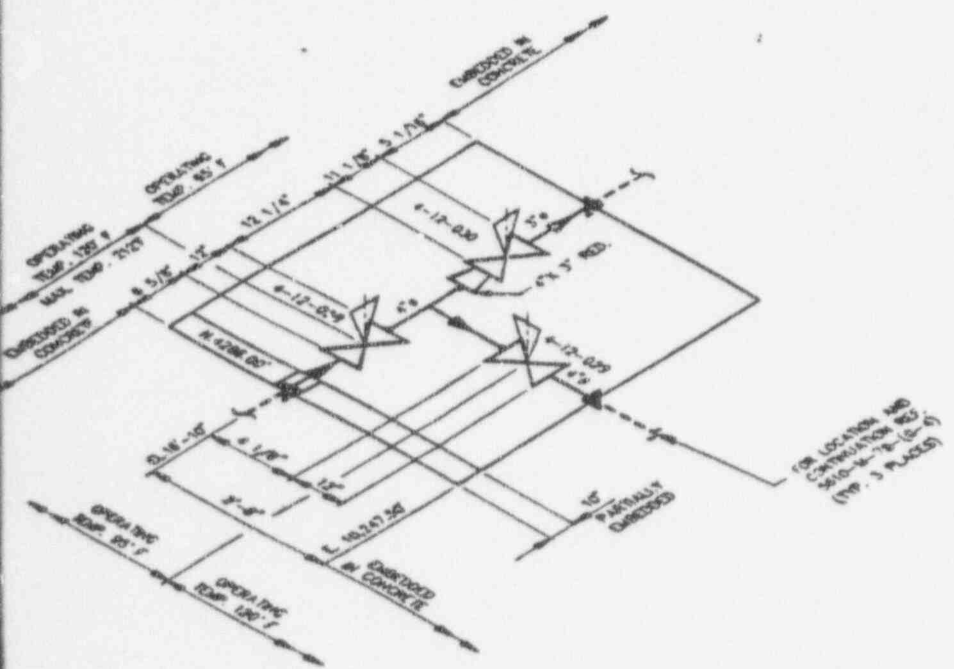
VALVE TAG NO.	WT.	VALVE TAG NO.	WT.
4-796	4309	4-913	3369
4-797	1023	4-918	918
4-910	7859	4-920	59
		4-918A	84



DETAIL 1
(NTS)

NOTES:

1. THIS DRAWING IS MADE FROM ISOMETRIC NO. E-2369-IC-309 & E-2369-IC-318.
2. INSULATION: NONE
3. SYSTEM DESIGN PRESSURE 150 PSIG.
   DESIGN TEMPERATURE 200° F
   PEAK PRESSURE 70 PSIG.
   MAXIMUM OPERATING TEMPERATURE 212° F (EXCEPT AS NOTED)
   MAXIMUM TEMPERATURE 212° F
4. FOR P&ID SEE DRAWING 5613-M-3033, SHEET 1.
5. PIPING: 10" SCH. 10S SMLS A312 TYPE 304
   8" SCH. 10S SMLS A312 TYPE 304
   4" SCH. 10S SMLS A312 TYPE 304
6. LOCATION: AUX BUILDING
7. FOR PIPE SUPPORT DETAILS SEE DRAWINGS 5614-M-684-03,09 & PTN-C-90-389-001, 002.
8. THE PIPING AND SUPPORTS SHOWN ON THIS DRAWING ARE SAFETY RELATED.
9. THERMAL MOVEMENTS AT EQUIPMENT NOZZLES ARE NEGLIGIBLE UNLESS NOTED OTHERWISE.
10. FOR PIPING AND INSTRUMENT DIAGRAM OF THIS SYSTEM, SEE DRAWING 5614-M-430-54.
11. FOR SYSTEM SPECIFICATIONS OF PIPE, VALVES AND FITTINGS, SEE SPEC. 5177-M-52 AND WESTINGHOUSE SPEC. G-369866 (5610-M-410-14 SH.4).
12. ALL COMMODITIES SHALL BE TAGGED FOR IDENTIFICATION PER SPECIFICATION 5177-M-51 (6.14).
13. A-193 GR. B7 BOLTING AND A-194 GR. 2H NUTS MAY BE SUBSTITUTED FOR A-307 BOLTING LISTED IN 5177-PS-11.
14. FOR STRESS ANALYSIS, SEE CALCULATION M08-585-05 REV. 4.
15. UNLESS OTHERWISE INDICATED, NO ALLOWANCE HAS BEEN MADE FOR WELDED JOINTS OR GASKETS.



MISC. RADWASTE DRAIN DETAIL

NUCLEAR SAFETY RELATED

REV	DATE	REVISION	BY	CH	APP
2	6/20/95	ISSUED AS-BUILT FOR PC/M 95-120.	WJ	POB	CP
1	7/26/93	ISSUED AS-BUILT FOR DCR-TPM 93-079.	SP/HJF	INITIALS ON FILE	
0	12/24/91	THIS DRAWING SUPERSEDES DWG. 5610-P-694 SH.1. ISSUED AS-BUILT FOR PC/M 90-389. INCORP. CRN M-3257, CRN M-3276, CRN M-3284, AND CRN M-3336 AND ISSUED INTO THE FPL SYSTEM.	AJC	INITIALS ON FILE	

FPL

TURKEY POINT NUCLEAR UNIT 4

PIPING ISOMETRIC

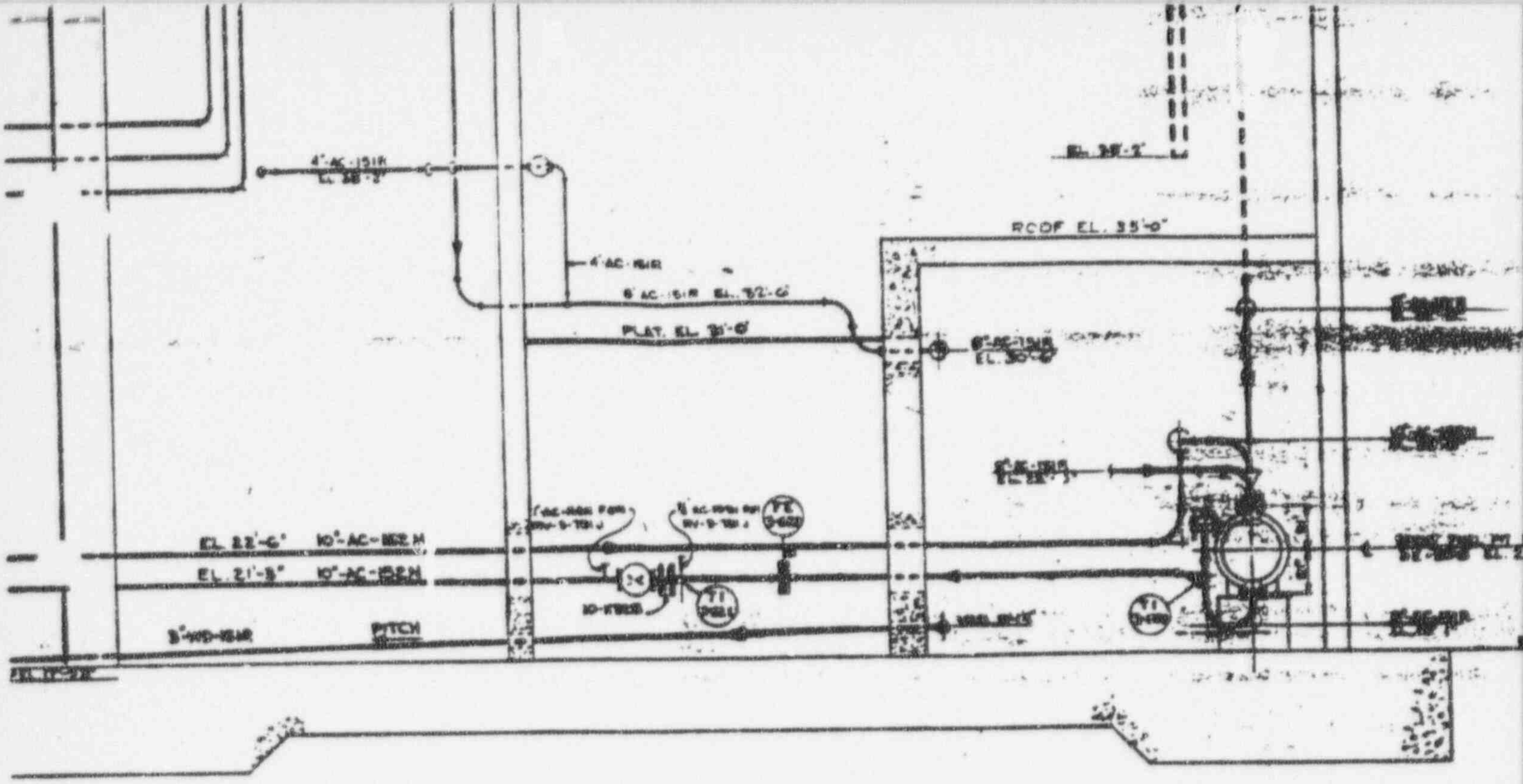
SPENT FUEL PIT COOLING SYSTEM
SUCTION PIPING TO SPENT FUEL PIT PUMPS
AND MISCELLANEOUS DETAILS

BECHTEL

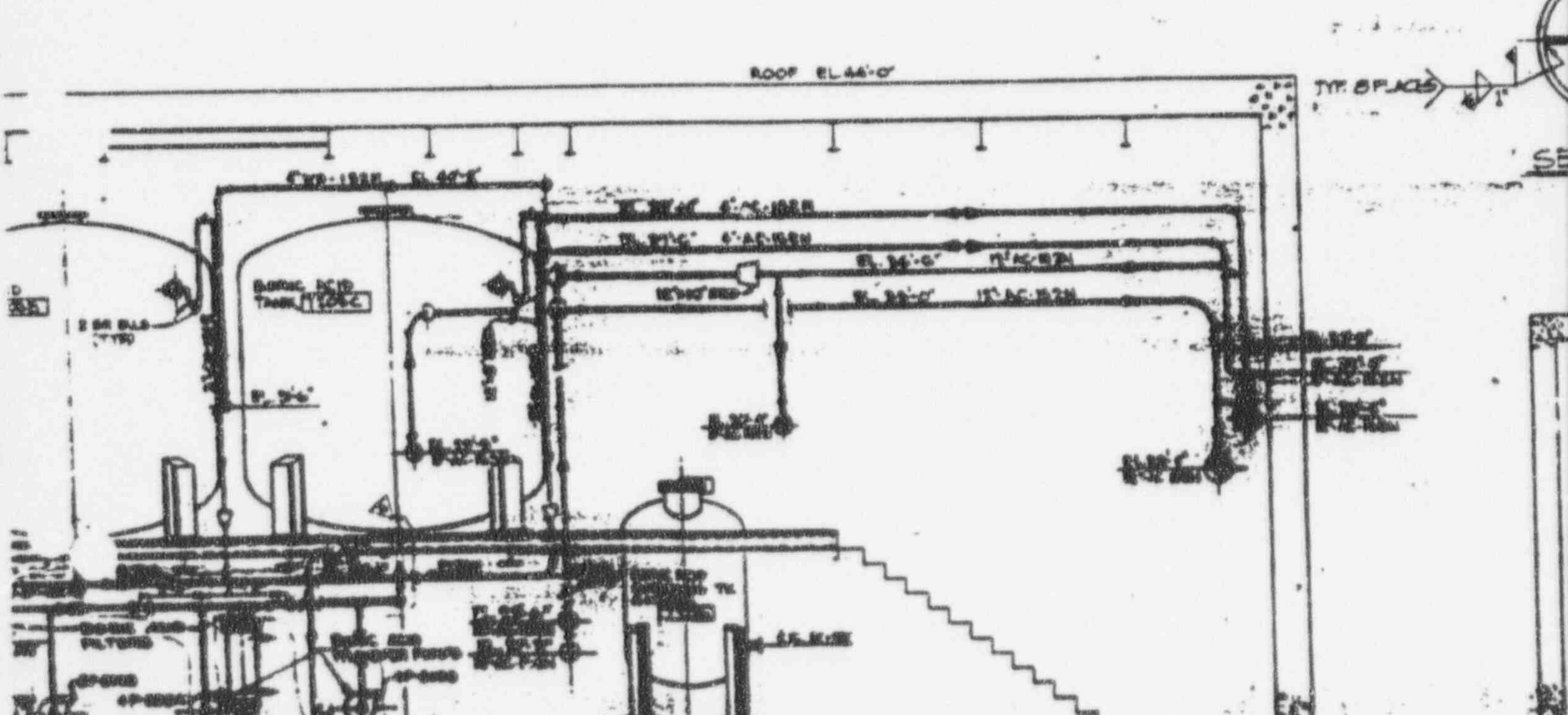
DRAWING 5614-P-694

033

9610250070-02

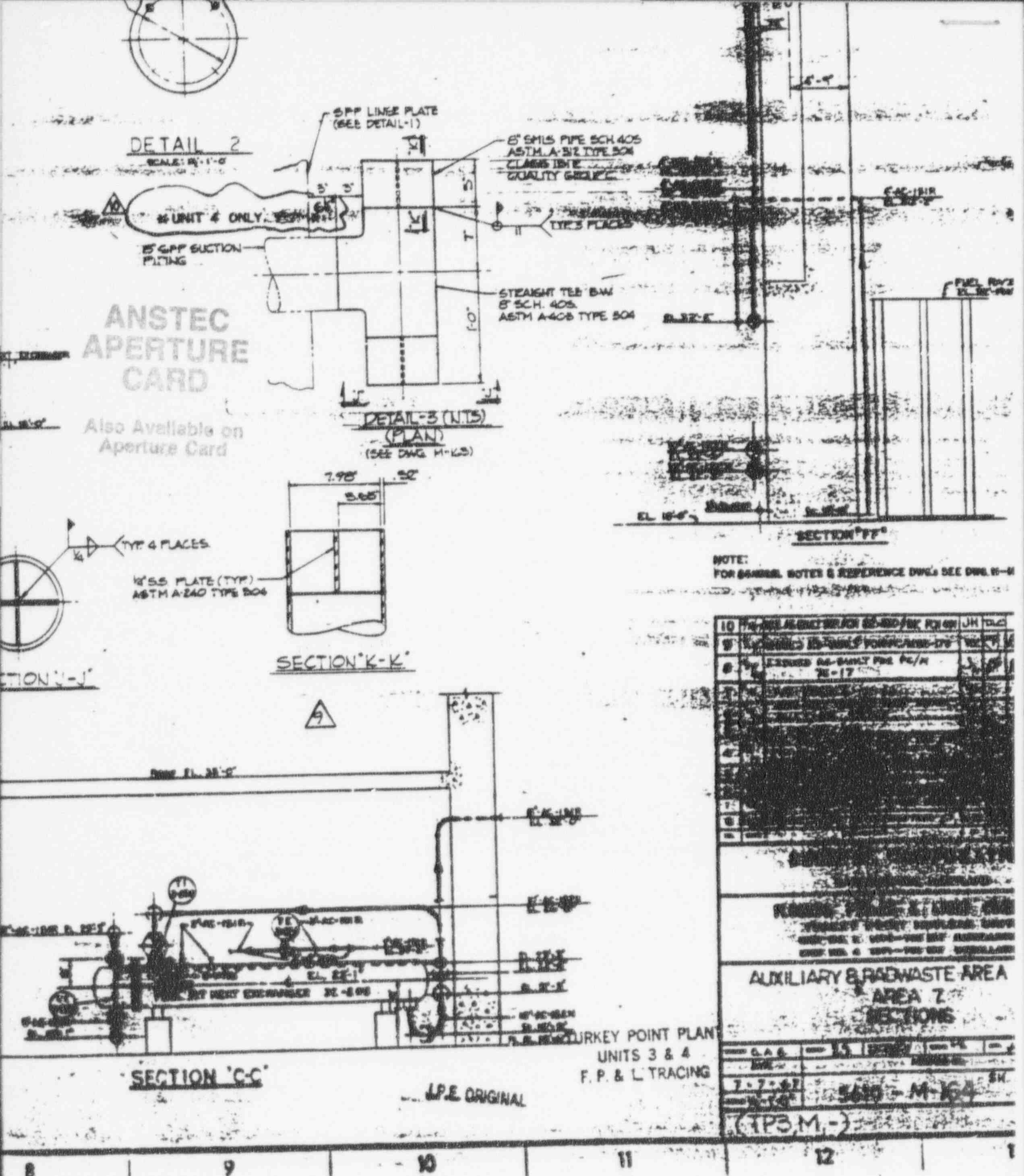


SECTION "A-A"

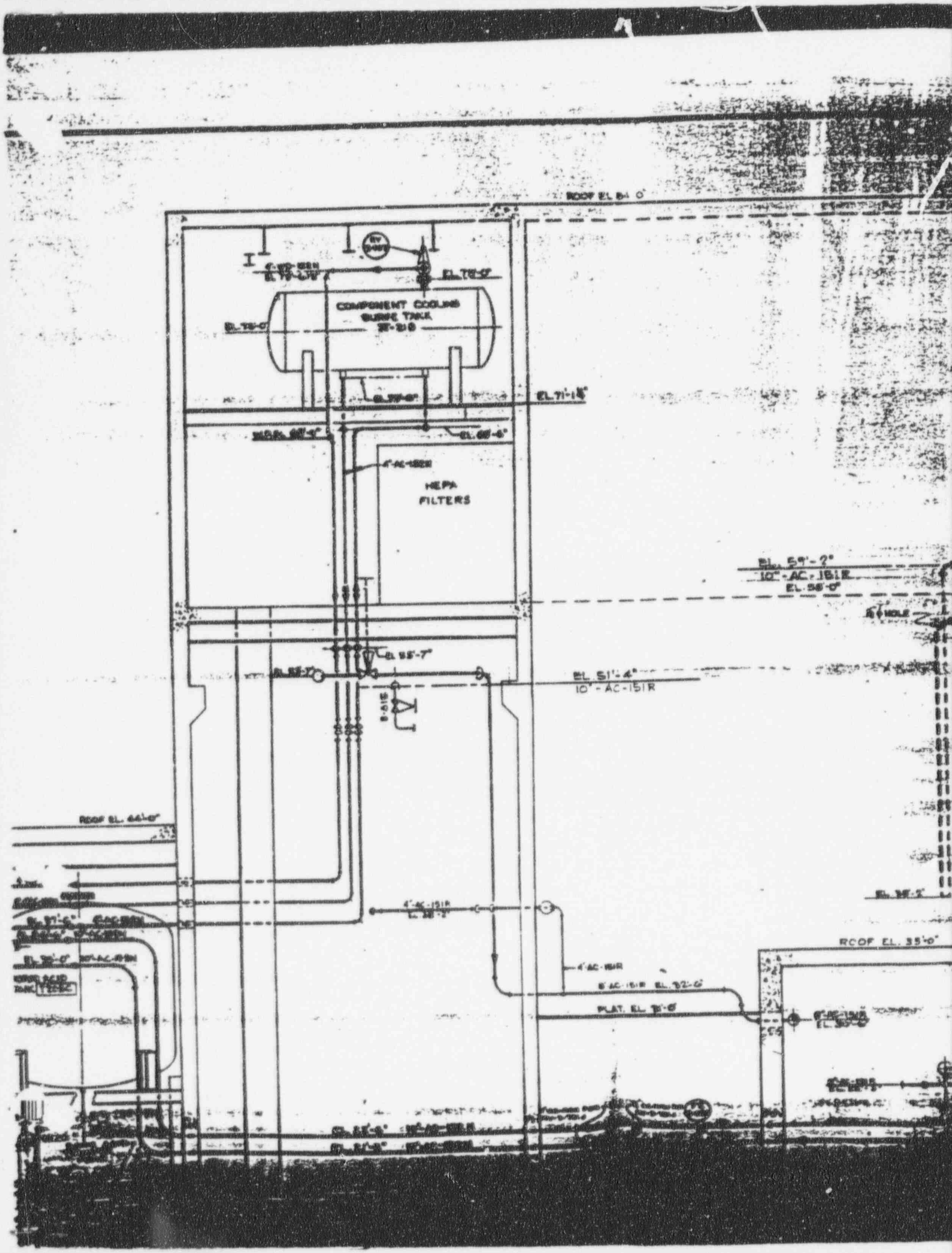


SECTION D-D

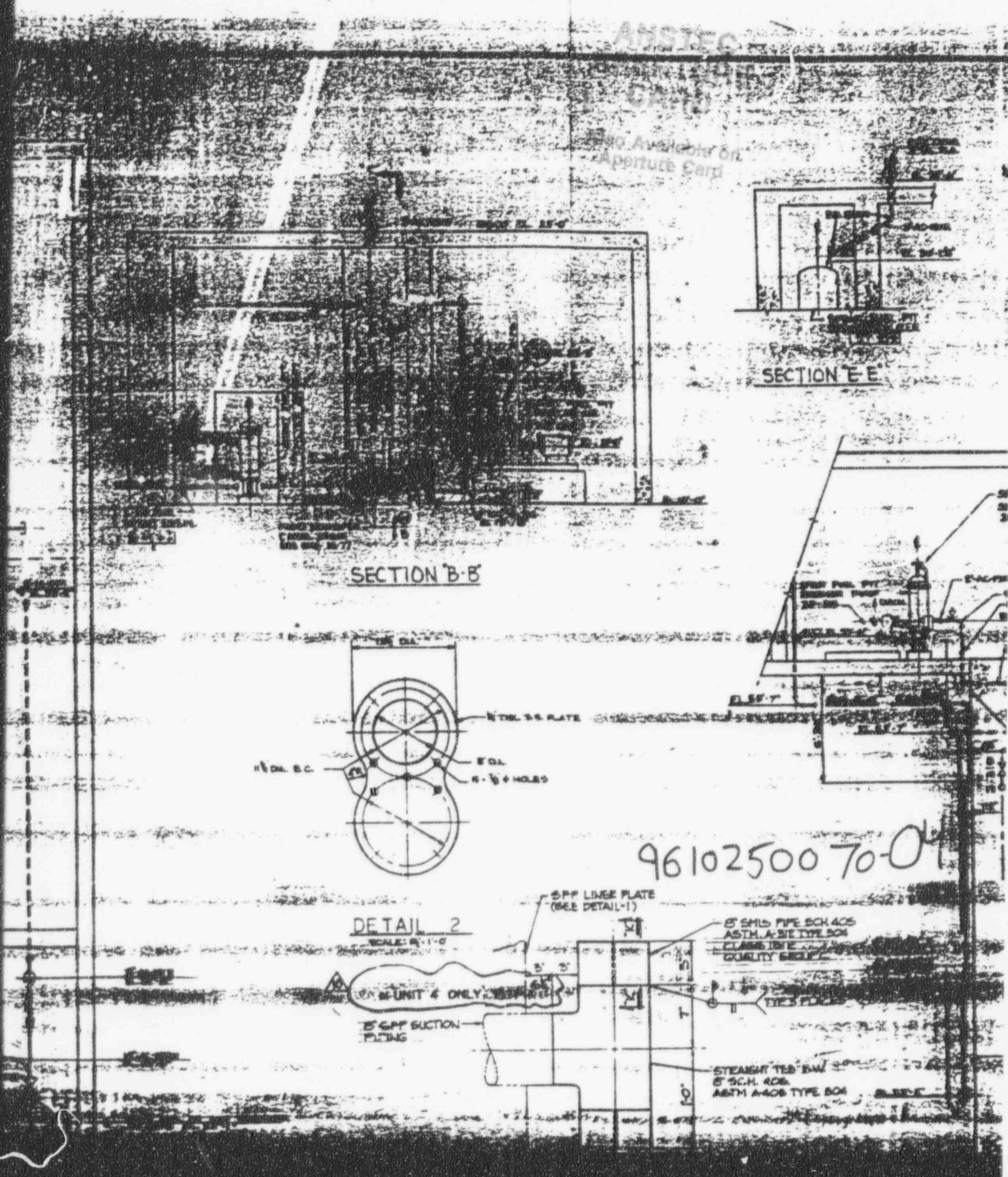
4 5 6 7



9610250070-03



ROOF EL. 84' 0"
COMPONENT COOLING
SURGE TANK
EL. 71'-1 1/2"
HEPA
FILTERS
EL. 59'-2"
10"-AC-181R
EL. 51'-4"
10"-AC-151R
ROOF EL. 44'-0"
4"-AC-151R
ROOF EL. 35'-0"


ANSTEC
APERTURE
CARD
Also Available on
Aperture Card
SECTION E-E
SECTION B-B
DETAIL 2
SPP LINER PLATE
(SEE DETAIL-1)
UNIT 4 ONLY
SPP SUCTION
PIPING
9610250070-04

SRO QUESTION 63

Plant conditions:

- Preparations are being made for refueling operations
- The refueling cavity is filled with the transfer gate valve open.
- Alarm annunciators H-1/1, SFP LO LEVEL and G-9/5, CNTMT SUMP HI LEVEL are in alarm.

Which ONE of the following is the required IMMEDIATE ACTION in response to these conditions?

ANSWER:

b. - Sound the containment evacuation alarm.

REFERENCE:

3-ONOP-033.2, Refueling Cavity Seal Failure, page 5
E.O. 6 of LP-6902144

COMMENT:

Annunciator H-1/1 is a entry condition for 3-ONOP-033.1 (Attachment 1). The immediate actions for 3-ONOP-033.1 is to verify the alarm is valid.
Additionally, the RCO is required to respond to alarms per 0-ADM-219 (Attachment 2). Alarms H-1/1 and G-9/5 are classified as priority 3 (BLUE) alarms requiring prompt (not immediate) action. The specified operator actions for both alarms per ARP-097.CR is to verify the alarms (ie containment sump level recorder and spent fuel pit level indication). See Attachment 3 for the ARP actions.

RECOMMENDATION:

Accept answer A as an additional correct answer.

A. - Verify alarms by checking containment sump level recorder and spent fuel level indication.

ATTACHMENT 1 (PAGE 1 OF 1)

1.0 PURPOSE

1.1 This procedure provides instructions for response to off-normal conditions of the Spent Fuel Pit (SFP) and the SFP Cooling System including SFP High/Low Level and High Temperature.

2.0 SYMPTOMS

2.1 Annunciators

2.1.1 H 1/1, SFP LO LEVEL

2.1.2 H 1/2, SFP HI TEMP

2.1.3 H 1/3, SFP HI LEVEL

2.2 Indications

2.2.1 High/low SFP, as indicated on LI-3-651, (VPB) or by local visual inspection using level placard located at the Southwest corner of the SFP (normal level is 56'- 10" - 57'2")

2.2.2 Low SFP Cooling Pump Discharge Pressure (PI-3-651B; 651A)

2.2.3 SFP Filters High ΔP (DPI-3-151 A, B, C),
ΔP across filters should be < 10 psid

2.2.4 Low SFP Skimmer Pump Discharge Pressure (PI-3-671A)

2.2.5 SFP Skimmer Pump Filters High ΔP (DPI-3-150A, B, C),
ΔP across filters should be < 10 psid

2.2.6 SFP Demineralizers High ΔP (INLET PI-3-655A - OUTLET PI-3-655B)
(N/A if RWST is on Recirc through the SFP Demin)
ΔP across Demin should be < 35 psid

3.0 AUTOMATIC ACTIONS

NOTE

Bkrs for SFP pps are located inside Unit 3 SFP Hx Room.

a. 3A SFP Pp Bkr 3P212A

b. 3B SFP Pp Bkr 3P212B (Power supply to 3NP212 panel is fed from Bkr 30309)

3.1 Possible SFP Cooling Pump Breaker trip on overload.

3.2 Possible SFP Skimmer Pump Breaker trip on overload. (Bkr 30777)

4.0 IMMEDIATE ACTIONS

⟹ 4.1 Verify annunciated alarm is valid.

ATTACHMENT 2 (PAGE 1 OF 2)

3.0 **RESPONSIBILITIES**

3.1 Nuclear Plant Supervisor (NPS) - The NPS shall provide technical guidance for event mitigation when ARPs are in effect.

3.2 Assistant Nuclear Plant Supervisor (ANPS) - The ANPS should direct the detailed event mitigation strategy for the affected unit unless otherwise directed by the NPS when ARPs are in effect.

3.3 Nuclear Watch Engineer (NWE) - The NWE should direct non-licensed operators, if necessary, to determine the cause of the alarm condition and the performance of corrective actions when ARPs are in effect.

3.4 Affected Unit Reactor Control Operator (RCO) - The affected unit RCO is responsible for the following when ARPs are in effect:

3.4.1 Respond to alarms based on color code priority and plant conditions.

3.4.2 Reading the ARP in effect and performing the event mitigation strategy for alarms received in the Control Room.

3.4.3 Transition to the appropriate procedures if required by the ARP.

3.4.4 Inform the unit ANPS of abnormal alarm conditions.

3.4.5 Coordinate actions with non-licensed operators when the alarm condition occurs at local annunciator panel in the field.

3.5 Non-affected Unit Reactor Control Operator - The non-affected unit RCO should maintain the non-affected unit in a safe condition which does not threaten the event mitigation strategy on the affected unit when ARPs are in effect.

3.6 Third Licensed Operator - The third licensed operator should assist the affected unit(s) RCO in performance of the event mitigation strategy when ARPs are in effect.

3.7 Non-Licensed Operator (NLO) - The NLO is responsible for the following when ARPs are in effect on either unit:

3.7.1 Read the ARP in effect and perform the event mitigation strategy for alarms received at local annunciator panels.

3.7.2 Inform the affected unit RCO of the alarm condition.

3.7.3 Performing actions requested from the affected unit RCO or NWE to correct the alarm conditions in the Control Room.

ATTACHMENT 2 (2 OF 2)

4.0 DEFINITIONS

4.1 Annunciator Response Procedures (ARPs)

Plant procedures that specify the operator actions required to mitigate the consequences of transients that cause plant parameters to exceed alarm setpoints.

4.2 Local (Locally)

An action performed by an operator outside the Control Room.

4.3 Manual (Manually)

An action performed by the operator in the Control Room. This does not include automatic actions which take place without operator intervention.

4.4 Priority 1 (Red): Nuclear Safety

These alarms require immediate response and reflect a potential challenge to plant safety and require protective systems to activate. These alarms include: SI and Reactor/Turbine/Gen Trips.

4.5 Priority 2 (Yellow): Power Production Availability

These alarms require immediate response and reflect a challenge to plant equipment or systems that may affect continued plant availability or timely recovery. Immediate response to these alarms would be deferred only if action were required by a Priority 1 alarm. Failure to properly respond to a Priority 2 condition may lead to or contribute to a higher level condition.

4.6 Priority 3 (Blue): Investment Protection

These alarms require prompt response and provide information that, if unattended, may result in a threat to higher level actions. Prompt action to this level of alarm may reduce the consequences of the problem by minimizing equipment damage or material waste.

4.7 Priority 4 (White): Status/Information

These alarms require non-priority response and reflect equipment status, transitions, or conditions to be corrected, but do not threaten the unit availability. Because they are not strictly "Information Only" items, they may warrant operator action. Priority 4 items are deferred in the face of higher priority items.

4.8 Transition

A change from one place to another in the procedures, either from one step to another step or from one procedure to another procedure.

BLUE	INVESTMENT PROTECTION	H 1.1

H1

	1	2	3	4	5	6	7	8	9
1	■								
2									
3									
4									
5									
6									

~~ATTACHMENT 8~~
~~Page 1 of 54~~
~~Panel H~~

ATTACHMENT 3
(PAGE 1 OF 2)

SFP
LO LEVEL

DEVICES:
Level actuator at north end of SFP

LT-3-651

SETPOINTS:
56' 10"

OPERATOR ACTIONS:

1. Verify alarm by checking the following:
   a. LI-3-651 (VPB)

2. Corrective actions:
   a. Dispatch operator to check:
      (1) Spent fuel level indication LY-3-651 (behind VPB)
      (2) Local level at the SFP.
      (3) Power to LT-3-651 (LP-50, Bkr 19 - east wall of cable spreading room.
   b. Refer to 3-ONOP-033.1, SPENT FUEL PIT (SFP) COOLING.
   c. IF in a refueling configuration with the SFP transfer tube open, THEN terminate refueling operations and refer to 3-ONOP-033.2, Refueling Cavity Seal Failure System Malfunction for cavity seal failure required actions.
   d. Refer to TS 3.9 for additional actions.

NOTES

- *If SFP cooling has to be secured, monitor SFP temperature per Attachment 3 of 3-OP-033, notify Reactor Engineering.*
- *If annunciator is OOS, refer to 0-ADM-214.*

CAUSES:

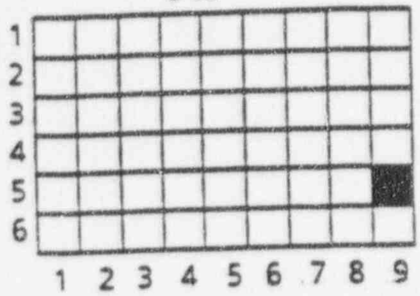
1. Actual low level in SFP (Evaporation, leakage, or SFP system valve misalignment)
2. Loss of power to LT-3-651
3. Instrumentation failure

REFERENCES:

1. FPL DWG 5613-M-3033 Sh 1
2. Tech. Spec. Section 3.9

BLUE	INVESTMENT PROTECTION	G 9/5

G45



~~ATTACHMENT~~ 7
~~Page 53 of 54~~
~~Panel G~~

ATTACHMENT 3
(PAGE 2 OF 2)

CNTMT
SUMP
HI LEVEL

DEVICES:

R-1418
(unit 4 VPA)

SETPOINTS:

30"

OPERATOR ACTIONS:

1. Verify alarm by checking the following:
   a. CNTMT sump recorders R-1418 (unit 4 VPA), R-6308A/B, and DDPS point DDPSA102-3.

2. Corrective actions:
   a. Verify proper operation of the containment sump pumps.
   b. Pump down the sump as required.
   c. Monitor RCS parameters for indications of leak, if applicable.
   d. Perform 3-OSP-041.1 to determine the RCS leak rate, if applicable.

NOTE

If annunciator is OOS, refer to 0-ADM-214.

CAUSES:

1. RCS leak.
2. Instrument malfunction.

REFERENCES:

1. FPL DWG 5610-M-12

SRO QUESTION 84

Which ONE of the following is the basis for step 1, "Verify Reactor Trip", of FR-S.1, Response to Nuclear Power Generation/ ATWS?

ANSWER:

a. - To ensure that only decay heat and reactor coolant pumps are adding heat to the RCS.

REFERENCE:

3-BD-EOP-FR-S.1, Response to Nuclear Power Generation/ATWS, page 8. E.O. 6 of LP-6902346

COMMENT:

A review of the corresponding Step 1 of 3-EOP-E-O and 3-EOP-FR-S.1 with respect to reducing reactor power indicates a difference. Rods are manually inserted in FR-S.1 but not E-0. While the basis documents for both procedures discuss decay heat and reactor coolant pump heat, only Basis Document BD-EOP-FR-S.1 discusses the need for taking further corrective action if the reacto is not tripped. (ie manually insert control rods) (see attached basis documents and procedures)

RECOMMENDATION:

Accept answer C as an additional correct answer.

C. - To alert the operator to take further corrective action if the reactor is NOT tripped.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTES

- *Steps 1 through 14 are IMMEDIATE ACTION steps.*
- *Foldout page shall be monitored throughout this procedure.*

1	**Verify Reactor Trip:** • Rod bottom lights - ON • Reactor trip and bypass breakers - OPEN • Rod position indicators - AT ZERO • Neutron flux - DECREASING	Manually trip reactor. <u>IF</u> reactor power is greater than 5% <u>OR</u> intermediate range power is <u>NOT</u> stable or decreasing, <u>THEN</u> perform the following: a. Direct operator to monitor Critical Safety Functions using 3-EOP-F-0, CRITICAL SAFETY FUNCTION STATUS TREES. b. Go to 3-EOP-FR-S.1, RESPONSE TO NUCLEAR POWER GENERATION/ATWS, Step 1.

BASIS DOCUMENT

WOG Procedure Step: 1 PTN Procedure Step: 1

BASIS:

Reactor trip must be verified to ensure that the only heat being added to the RCS is from decay heat and reactor coolant pump heat. The safeguards systems that protect the plant during accidents are designed assuming that only decay heat and pump heat are being added to the RCS. If the reactor cannot be tripped, a transition is made to FR-S.1, RESPONSE TO NUCLEAR POWER GENERATION/ATWS, to deal with ATWS conditions.

STEP DEVIATION FROM WOG GUIDELINE:

TYPE	DESCRIPTION
8	The rod bottom lights are checked to be ON vice LIT to conform with plant specific terminology.
1	The RNO was changed so that a transition to FR-S.1, RESPONSE TO NUCLEAR POWER GENERATION/ATWS, will only be made if the criteria from the Critical Safety Function status tree for subcriticality is satisfied. This provides the operator with a clear definition of what constitutes a reactor trip, and eliminates the need for the operator to make a decision under stress. This change complies with the intent of the RNO column provided in ERG Feedback item DW-88-033.
9	The WOG guidelines require initiation of Critical Safety Function status tree monitoring whenever exiting E-0. The RNO was modified to provide procedural guidance for performance of this task so that the need to memorize User's Guide requirements is eliminated.

PLANT SPECIFIC SETPOINTS:

5% - Reactor power level just in the Power Range. (EOP Setpoint P.2)

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE

Steps 1 through 2 are IMMEDIATE ACTION steps.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1	Verify Reactor Trip: • Rod bottom lights - ON • Reactor trip and bypass breakers - OPEN • Rod position indicators - AT ZERO • Neutron flux - DECREASING	Manually trip reactor. <u>IF</u> reactor will <u>NOT</u> trip, <u>THEN</u> manually insert control rods.
2	Verify Turbine Trip:	
	a. All turbine stop valves - CLOSED	a. Manually trip turbine. <u>IF</u> turbine will <u>NOT</u> trip, <u>THEN</u> manually run back turbine. <u>IF</u> steam flow to turbine causes uncontrolled RCS cooldown, <u>THEN</u> close main steamline isolation and bypass valves.
	b. Close MSR Main Steam Supply Stop MOVs	b. Close main steamline isolation and bypass valves.
	c. Reheater timing cam - AT ZERO	c. Remove timing cam to close timing valves. <u>IF</u> any timing valve can <u>NOT</u> be closed, <u>THEN</u> close main steamline isolation and bypass valves.
	d. MSR Purge Steam Valves - CLOSED	d. Manually close MSR purge valves. <u>IF</u> any MSR purge valve can <u>NOT</u> be closed, <u>THEN</u> close main steamline isolation and bypass valves.
3	Check AFW Pumps - ALL RUNNING	Manually open steam supply valves.

BASIS DOCUMENT

WOG Procedure Step: 1 PTN Procedure Step: 1

BASIS:

Reactor trip must be verified to ensure that the only heat being added to the RCS is from decay heat and reactor coolant pump heat. The safeguards systems that protect the plant during accidents are designed assuming that only decay heat and pump heat are being added to the RCS. If the reactor cannot be tripped, then the control rods should be manually inserted into the core in order to decrease reactor power.

STEP DEVIATION FROM WOG GUIDELINE:

TYPE	DESCRIPTION
8	The rod bottom lights are checked to be ON vice LIT to conform with plant specific terminology.

PLANT SPECIFIC SETPOINTS:

N/A

SRO QUESTION 96

Which ONE of the following is the lowest level of position responsible for ensuring entries are made in the Technical Specification Related Equipment Out-Of-Service Index?

ANSWER:

b. Assistant Nuclear Plant Supervisor

REFERENCE:

0-ADM-213, page 10

COMMENT:

0-ADM-200, Conduct of Operations states that the Nuclear Watch Engineer (NWE) is responsible for "Routinely relieving the ANPS of the Control Room command function". By relieving the ANPS the NWE assumes all responsiblities of the ANPS thereby becoming the lowest level able to make entries into the EOOS index. Additionally FP&L Training Dept. recently resolved a similar test question on our Contractor Exam (Question 88, see attached) and ruled similar as this request (ie accepted ANPS and NPS since ANPS can relieve the NPS).

RECOMMENDATION:

Accept answer d as an additional correct answer.

d. - Nuclear Watch Engineer

3.4.6 Review and approve all unit Plant Work Orders, prior to work commencing and ensuring the NWE and RCO are aware of all work outside of the Control Room.

3.4.7 Maintain the equipment out-of-service book in accordance with 0-ADM-213, Technical Specification Related Equipment and Risk Significant SSC Out-of-Service Logbook

3.4.8 Coordinate the on shift training of licensed operators.

3.4.9 Maintaining a thorough knowledge and understanding of the following:

1. The duties and responsibilities of the ANPS required by the facility operating licenses.
2. Conditions and limitations contained in the facility operating licenses and Technical Specifications.
3. Operating procedures for the nuclear units.
4. Plants' status at all times. [Commitment - Step 2.3.3]

3.4.10 Notifying the NPS when any Technical Specification Limiting Condition for Operation is entered.

3.4.11 Notifying the NPS when any Risk Significant SSC is removed from service.

3.5 <u>Nuclear Watch Engineer (NWE)</u> - One Nuclear Watch Engineer will be assigned to assist the NPS in coordinating the activities of Licensed and Non-Licensed personnel during routine, complicated, or infrequent evolutions. The NWE reports to the NPS and is responsible for:

3.5.1 Performing duties assigned by the NPS or his designee for each unit.

3.5.2 Coordinating the activities of the Control Room with other operations and plant personnel to achieve safe, reliable, and efficient unit operation as directed by the NPS/ANPS.

3.5.3 Supervising and coordinating the operation of plant equipment and systems when assigned by the NPS.

3.5.4 Acting as the Fire Brigade Chief or Shift Communicator, when assigned, but not both.

3.5.5 Routinely relieving the ANPS of the Control Room command function to enable the ANPS to leave the Control Room. In an emergency, function as the NPS if required.

QUESTION 88 RO & SRO

Which one of the following is correct regarding who has control and responsibility for the issuance of ICCS keys?

a. NPS

b. NPS/ANPS

c. NPS/ANPS/NWE

d. NPS/ANPS/NWE/RCO

ANSWER: A

REFERENCE: 294001K1.05 (3.4/3.6)
ADM-205, Section 10, Key Control

ADM-205 section 5.4, Key Control, clearly states ICCS are under the control of the NPS. The question asks about issuance, nothing else. NO CHANGES REQUIRED.

QC reviewer recommendation considered and final decision made to accept A or B as correct because of numerus exceptions to the NPS issue requirement. ANPS may be acceptable for issue of ICCS keys under certain conditions. ADM-205 does not have any provisions for the RCO or NWE ever issuing ICCS keys.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

September 12, 1996

Mr. Ralph L. Tetrick
18990 SW 270 Street
Homestead, FL 33031

Dear Mr. Tetrick:

In response to your letter of July 30, 1996, we have reconsidered the proposed denial issued to you on July 19, 1996, and reviewed the grading of the written examination administered to you on June 14, 1996, in light of the information you supplied. We find that you did not pass the written examination. The result of our review is enclosed.

Consequently, the proposed denial of your license application is sustained. If you accept the proposed denial and decline to request a hearing within 20 days as discussed below, the proposed denial will become a final denial. You may then reapply for a license in accordance with 10 CFR 55.35, subject to the following conditions:

a. Because you did not pass the written examination administered to you on June 14, 1996, you will be required to retake the written examination.

b. Because you did pass your operating test administered on June 19 - 21, 1996, you may request a waiver of that portion. A waiver would be valid for up to 1 year from your test date.

c. You may reapply for a license 2 months from the date of this letter. A reexamination will be scheduled upon request by your facility management.

If you do not accept the proposed denial, you may, within 20 days of the date of this letter, request a hearing pursuant to 10 CFR 2.103(b)(2). Submit your request, in writing, to the Secretary of the Commission, U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, with a copy to the Assistant General Counsel for Hearings and Enforcement, Office of the General Counsel, at the same address.

Failure on your part to request a hearing within 20 days constitutes a waiver of your right to demand a hearing and, for the purpose of reapplication under

10 CFR 55.35, renders this letter a notice of final denial of your application, effective as of the date of this letter.

If you have any questions, please contact Mr. Stuart A. Richards, at 301-415-1031.

Sincerely,

R. L. Spessard

for Bruce A. Boger, Director
Division of Reactor Controls
and Human Factors
Office of Nuclear Reactor Regulation

Docket No. 55-20726

Enclosure: As stated

cc w/encl.: Mr. W. Lindsey, Licensed Operator
Training Supervisor, Turkey Point

NRC REVIEW FOR RALPH L. TETRICK - SRO APPLICANT - TURKEY POINT

In response to the applicant's letter of July 30, 1996, the NRC reconsidered the proposed denial issued on July 19, 1996, and reviewed the grading of the written examination administered to the applicant on June 14, 1996. Giving due consideration to the information supplied by the applicant, the NRC has determined that the applicant failed the written examination. The results of the NRC review are outlined below.

REVIEW OF WRITTEN EXAMINATION QUESTIONS

EXAMINATION QUESTION #24:

Which ONE of the following describes the Spent Fuel Pool Cooling (SFPC) system basic operation and connections to the Spent Fuel Pool (SFP)?

The SFPC pumps normally take a suction on the:

a. *"High" line near the top of the SFP and discharge through a line 1 foot below the top with a 1/2 inch siphon break hole 6 inches below the water level.*

b. *"Low" line near the bottom of the SFP and discharge through a line 6 inches below the top with a 1/2 inch siphon break hole 1 foot below the water level.*

c. *"High" line 1 foot below the top of the SFP and discharge through a line at the bottom of the SFP with a 1/2 inch siphon break hole 6 inches below the water level.*

d. *"Low" line located with a 1/2 inch siphon break hole 1 foot below the water level of the SFP and discharge through a line 1 foot below the top.*

ANSWER: *a*

Applicant's Contentions (summary): The applicant asserts that choices "a" and "c" are both partially incorrect because the discharge path enters at the top of the SFP but then extends down 20 feet to the middle of the pool, where the returning fluid is actually "discharged." The applicant also points out some other minor discrepancies in the elevations specified in the question. The applicant recommends that the NRC accept choice "c" as an additional correct answer because both choices are partially (in)correct.

NRC Analysis and Conclusion: Answers "b" and "d" are clearly incorrect because the "low" line suction is normally locked closed to prevent inadvertent draining of the SFP. The suction points specified in choices "a" and "c" are essentially the same, so the decisive difference between the choices rests in the return flow path - "high" (i.e., one foot below the top

of the SFP) in choice "a" and "low" (i.e., at the bottom of the SFP) in choice "c." Although the intent of the question was to evaluate the applicants' general knowledge of the piping penetrations through the wall of the SFP (not the specific elevations of each penetration with respect to the top of the pool or the nominal water level), the NRC agrees with the applicant that the wording of the choices (i.e., "discharge through a line") could also lead the reader to believe that the question was concerned with the location of the outlet of the SFP cooling system discharge pipe. Under this interpretation neither answer "a" or "c" is correct; the discharge pipe terminates at an elevation approximately half way between the bottom and the top of the SFP. Therefore, the NRC concludes that no answers are completely correct and the question should be deleted from the examination.

EXAMINATION QUESTION #63:

Plant conditions:

- *Preparations are being made for refueling operations.*
- *The refueling cavity is filled with the transfer tube gate valve open.*
- *Alarm annunciators H-1/1, SFP LO LEVEL and G-9/5, CNTMT SUMP HI LEVEL are in alarm.*

Which ONE of the following is the required IMMEDIATE ACTION in response to these conditions?

a. *Verify alarms by checking containment sump level recorder and spent fuel level indication.*

b. *Sound the containment evacuation alarm.*

c. *Initiate containment ventilation isolation.*

d. *Initiate control room ventilation isolation.*

ANSWER: b

Applicant's Contentions (summary): The applicant asserts that the reactor control operator (RCO) is required to respond to alarms per 0-ADM-219, "Annunciator Response Procedure [ARP] Usage," by reading the ARP in effect and performing the event mitigation strategy for the alarms received in the control room. The applicant also states that, individually, these alarms are priority 3, which require prompt rather than immediate action, and that the appropriate response for each alarm, per the associated ARP and off-normal operating procedures, is to verify the alarm. Therefore, answer "a" should be accepted as an additional correct answer.

NRC Analysis and Conclusion: Reactor operators and senior reactor operators are expected to analyze alarms and determine the appropriate course of action

based upon the specific plant conditions and indications. This expectation is reflected in step 3.4.1 of 0-ADM-219, which directs the RCO to respond to alarms based on color code priority **and plant conditions**. Furthermore, steps 5.1.15 and 5.6.8 of 0-ADM-200, "Conduct of Operations," direct on-shift licensed operators involved in abnormal or emergency operations to believe and respond to their instrument indications until the instruments are proven to be incorrect. The plant conditions and indications specified in this question (i.e., the refueling cavity filled and the transfer tube gate valve open with coincident SFP LOW LEVEL and CONTAINMENT SUMP HIGH LEVEL alarms) are mutually supportive, confirmatory, and sufficient to enter Off-Normal Operating Procedure 3-ONOP-033.2, "Refueling Cavity Seal Failure." That procedure has only one IMMEDIATE ACTION - to sound the containment evacuation alarm. In accordance with step 5.2.1 of 0-ADM-211, "Emergency and Off-Normal Operating Procedure Usage," operators shall be capable of performing steps identified as IMMEDIATE ACTION steps from memory, and step 3.5.1 requires the RCO to ensure that all immediate operator actions of the procedure in effect are performed.

Furthermore, step 5.13 of 0-ADM-211 states that plant operating procedures have the following order of priority: Functional Restoration Procedures, Optimal Recovery Procedures, and Off-Normal Operating Procedures. That information plus the fact that both of the existing alarms were classified by their respective ARPs as priority 3 (blue) alarms that require prompt rather than immediate response supports the argument that there is only one correct IMMEDIATE ACTION in response to the stated conditions. Therefore, the NRC concludes that choice "b" is the only correct answer to this question.

EXAMINATION QUESTION #84:

Which ONE of the following is the basis for step 1, "Verify Reactor Trip", of FR-S.1, Response to Nuclear Power Generation/ATWS?

a. To ensure that only decay heat and reactor coolant pumps are adding heat to the RCS.

b. To ensure shutdown margin is within Technical Specifications limits for HOT STANDBY.

c. To alert the operator to take further corrective action if the reactor is NOT tripped.

d. To verify that all automatic reactor protective features have functioned as designed.

Answer: a

Applicant's Contentions (summary): The applicant compares the corresponding steps and the difference in the basis documents for procedure FR-S.1, "Response to Nuclear Power Generation/ATWS," and procedure EOP-E-0, "Reactor Trip or Safety Injection." In FR-S.1, the control rods are manually inserted,

but they are not inserted in EOP-E-0. The basis documents for both procedures discuss decay heat and reactor coolant pump heat, but only the basis document for FR-S.1 discusses the need for inserting control rods if the reactor is not tripped. Therefore, accept answer "c" as an additional correct answer.

NRC Analysis and Conclusion: The basis documents for both FR-S.1 and EOP-E-0 clearly indicate that the first step of each procedure (i.e., "verify reactor trip") is necessary to ensure that the only heat being added to the reactor coolant system is from the reactor coolant pumps and the radioactive decay of fission products. The safeguards systems that protect the plant during accidents are designed on the basis of that assumption. Although the basis document for FR-S.1 goes on to state that the control rods should be manually inserted into the core to decrease reactor power if the reactor cannot be tripped, this does not change the fact that the reason (i.e., basis) for the (automatic and manual) action associated with this step of FR-S.1 (i.e., tripping the reactor or inserting control rods) is to reduce the heat load to within the capacity of the safeguards systems. Therefore, the NRC concludes that choice "a" is the only correct answer to this question.

EXAMINATION QUESTION #96:

Which ONE of the following is the lowest level position responsible for ensuring entries are made in the Technical Specification Related Equipment Out-Of-Service Index?

- *a. Nuclear Plant Supervisor*
- *b. Assistant Nuclear Plant Supervisor*
- *c. Senior Nuclear Plant Operator*
- *d. Nuclear Watch Engineer*

ANSWER: b

Applicant's Contentions (summary): The applicant recommends that choice "d" be accepted as an additional correct answer because procedure 0-ADM-200, "Conduct of Operations," makes the Nuclear Watch Engineer (NWE) responsible for routinely relieving the Assistant Nuclear Plant Supervisor (ANPS) of the control room command and control function to enable the ANPS to leave the control room. Consequently, the NWE becomes the lowest level responsible for making entries in the Technical Specification Related Equipment Out-of-Service Index.

NRC Analysis and Conclusion: Procedure 0-ADM-213, "Technical Specification Related Equipment and Risk Significant SSC Out-of-Service Logbook," states that the ANPS is the lowest level position responsible for entering inoperable equipment in the subject index. When the NWE relieves the ANPS, he becomes the ANPS; he is not authorized to make entries in the subject index unless he is acting in the capacity of the ANPS. Therefore, the NRC concludes that choice "b" is the only correct answer to this question.

SUMMARY OF NRC REVIEW

In summary, the NRC has concluded the following based on its review of the applicant's contentions:

QUESTION NO.	NRC CONCLUSIONS
24	This question has been deleted because there is no correct answer.
63	No change, choice "b" is the only correct answer.
84	No change, choice "a" is the only correct answer.
96	No change, choice "b" is the only correct answer.

APPLICANT'S ORIGINAL EXAMINATION GRADE: 78% (78 of 100)

FINAL NRC GRADE: 78.8% (78 of 99)

The applicant's final grade of 78.8% remains below the minimum passing grade of 80%. Therefore, the NRC has concluded that the applicant failed the written examination.

In response to the letter dated September 12, 1996 applicant contends the following:

EXAM QUESTION #63

Plant conditions:

- Preparations are being made for refueling operations.
- The refueling cavity is filled with the transfer tube gate valve open.
- Alarm annunciators H-1/1, SFP LO LEVEL and G-9/5 CNTMT SUMP HI LEVEL are in alarm.

Which one of the following is the required immediate action in response to these conditions?

A. Verify alarms by checking containment sump level recorder and spent fuel level indication.

B. Sound the containment evacuation alarm.

C. Initiate containment ventilation isolation.

D. Initiate control room ventilation isolation.

ANSWER: B

The NRC analysis and conclusion contends that reactor operators and senior reactor operators are expected to analyze alarms and determine the appropriate course of action based upon specific plant conditions and indications.

Applicant contends that performing an action based solely on annunciation alone is not the proper way to operate. Even though SFP LOW LEVEL and CONTAINMENT SUMP HIGH LEVEL alarms are mutually supportive and sufficient to enter 3-ONOP-033.2 "REFUELING CAVITY SEAL FAILURE" The annunciators should be verified by additional supportive information to preclude the possibility of annunciator failure. Additionally CONTROL ROOM ANNUNCIATOR RESPONSE procedure 3-ARP-097.CR states that for all alarms the ARP shall be consulted. Applicant therefore contends that answer "A" verify alarms is also a correct answer.

5.0 SUBSEQUENT ACTIONS

OTSC

NOTES

ANNUNCIATOR RESPONSE GUIDELINES

1) Unit ANPS/NPS SHALL be made fully aware/cognizant of all Annunciators at all times (whether they have cleared or are locked in).

2) RCO - Upon receipt of an annunciator, take immediate corrective actions as necessary, informing ANPS of any corrective actions.

3) Daily Annunciator Response Procedure Usage:

- *For expected alarms such as I&C working in Racks, actual opening of ARP's is not required.*
- *For common or frequent alarms (WBP, Blender Deviation) use of the ARP is required for the first annunciation on the particular shift for the day. Subsequent annunciation does not require ARP consultation.*
- *For ALL other alarms the ARP SHALL be consulted as well as any other applicable procedures.*

0520-96

5.1 Annunciator on Panel A

5.1.1 Perform Appropriate Attachment 1, Page 15

5.2 Annunciator on Panel B

5.2.1 Perform Appropriate Attachment 2, Page 69

5.3 Annunciator on Panel C

5.3.1 Perform Appropriate Attachment 3, Page 123

5.4 Annunciator on Panel D

5.4.1 Perform Appropriate Attachment 4, Page 177

5.5 Annunciator on Panel E

5.5.1 Perform Appropriate Attachment 5, Page 231

5.6 Annunciator on Panel F

5.6.1 Perform Appropriate Attachment 6, Page 285

5.7 Annunciator on Panel G

5.7.1 Perform Appropriate Attachment 7, Page 339

EXAM QUESTION #84

Which one of the following is the basis for step 1, "VERIFY REACTOR TRIP", of FR-S.1, RESPONSE TO NUCLEAR POWER GENERATION/ATWS?

A. To ensure that only decay heat and reactor coolant pumps are adding heat to the RCS.

B. To ensure shutdown margin is within technical specifications limits for hot standby.

C. To alert the operator to take further corrective action if the reactor is not tripped.

D. To verify that all automatic reactor protective features have functioned as designed.

ANSWER: A

The NRC contends that the basis for E-0 step one and FR-S.1 step one are the same and that there is only one answer.

The applicant contends that FR-S.1 is a FUNCTION RESTORATION PROCEDURE And that it gives guidance to restore CRITICAL SAFETY FUNCTIONS. SInce the reactor was verified not tripped in E-0 step one you are sent to FR-S.1 Where the operator is directed to insert rods because the reactor is not tripped. Because FR-S.1 is a FRP and gives guidance the applicant contends that the basis for FR-S.1 is twofold, (1) to ensure only decay heat is added and (2) To direct corrective actions. Therefore the applicant asks that answer "C" also be accepted as a correct answer.

4.0 DEFINITIONS

4.1 Action Verbs

All action verbs used in two-column format procedures are defined in 0-ADM-101, Procedure Writer's Guide.

4.2 Critical Safety Function

An activity which serves to protect the integrity of one or more of the physical barriers against radiation release.

4.3 Emergency Operating Procedures (EOPs)

Plant procedures that specify the operator actions required to mitigate the consequences of transients and accidents that cause plant parameters to exceed reactor protection system setpoints, engineered safety features setpoints, or other appropriate technical limits. The EOP network consists of all Optimal Recovery Procedures and Function Restoration Procedures.

4.4 Faulted

Refers to any steam generator with an unisolable leak in it's secondary pressure boundary of sufficient size to require Safety Injection.

4.5 Functional Restoration Procedures (FRPs)

Those procedures which respond to Critical Safety Function challenges. Guidance is provided to restore the Critical Safety Function to a satisfied condition. Typically, actions are based on the severity of the challenge and may not correspond to "good operational practice". These procedures are identified by the procedure identifier F or FR.

4.6 Local (Locally)

An action performed by an operator outside the Control Room.

4.7 Manual (Manually)

An action performed by the operator in the Control Room. This does not include automatic actions, which take place without operator intervention.

4.8 Optimal Recovery Procedures (ORPs)

Those procedures which provide guidance to recover the plant in the most efficient manner to a safe and stable end state. Typically, actions correspond to "good operational practice". These procedures are identified by the procedure identifiers E, ES, and ECA.

BASIS DOCUMENT

WOG Procedure Step: 1 PTN Procedure Step: 1

BASIS:

Reactor trip must be verified to ensure that the only heat being added to the RCS is from decay heat and reactor coolant pump heat. The safeguards systems that protect the plant during accidents are designed assuming that only decay heat and pump heat are being added to the RCS. If the reactor cannot be tripped, a transition is made to FR-S.1, RESPONSE TO NUCLEAR POWER GENERATION/ATWS, to deal with ATWS conditions.

STEP DEVIATION FROM WOG GUIDELINE:

TYPE	DESCRIPTION
8	The rod bottom lights are checked to be ON vice LIT to conform with plant specific terminology.
1	The RNO was changed so that a transition to FR-S.1, RESPONSE TO NUCLEAR POWER GENERATION/ATWS, will only be made if the criteria from the Critical Safety Function status tree for subcriticality is satisfied. This provides the operator with a clear definition of what constitutes a reactor trip, and eliminates the need for the operator to make a decision under stress. This change complies with the intent of the RNO column provided in ERG Feedback item DW-88-033.
9	The WOG guidelines require initiation of Critical Safety Function status tree monitoring whenever exiting E-0. The RNO was modified to provide procedural guidance for performance of this task so that the need to memorize User's Guide requirements is eliminated.

PLANT SPECIFIC SETPOINTS:

5% - Reactor power level just in the Power Range. (EOP Setpoint P.2)

BASIS DOCUMENT

WOG Procedure Step: 1

PTN Procedure Step: 1

BASIS:

Reactor trip must be verified to ensure that the only heat being added to the RCS is from decay heat and reactor coolant pump heat. The safeguards systems that protect the plant during accidents are designed assuming that only decay heat and pump heat are being added to the RCS. If the reactor cannot be tripped, then the control rods should be manually inserted into the core in order to decrease reactor power.

STEP DEVIATION FROM WOG GUIDELINE:

TYPE	DESCRIPTION
8	The rod bottom lights are checked to be ON vice LIT to conform with plant specific terminology.

PLANT SPECIFIC SETPOINTS:

N/A

EXAM QUESTION #96

Please review this question as stated in the original request.

THe following question was discovered to be wrong after the first request for review was sent. The applicant wishes for the question to also be considered.

EXAM QUESTION #90

When draining the RCS using 3-OP-041.9, REDUCED INVENTORY OPERATIONS, the reactor vessel head and pressurizer are both vented to containment atmosphere.

Which one of the following describes the effects on reactor vessel level indication if an adequate vent path is not provided? (Assume the reference leg remains full).

A. A vacuum in the rcs loops will result in level indication being lower than actual levels.

B. A vacuum in the rcs loops will result in level indication being higher than actual levels.

C. A positive pressure in the rcs loops will result in level indication being lower than actual levels.

D. The level instruments automatically compensate for positive or negative pressure.

ANSWER: A

REFERENCE: 3-OP-041.9, REDUCED INVENTORY OPERATIONS, Page 25,
Section 5.2.2.3 Caution
E.O. 3 OF LP-6902121

The assumption that the reference leg remains full makes this question invalid. At Turkey Point the drain down level indication has dry reference legs. This condition is verified by 0-PMI-041.110. Applicant requests that this question be deleted.

INITIALS
CK'D VERIF

5.2.1 (Cont'd)

6. Both RHR Pump Discharge Isolation valves have been throttled to limit maximum RHR System flow to 3200 gpm.

___ a. RHR Pump A Disch Isol, 3-754A

___ b. RHR Pump B Disch Isol, 3-754B

___ 7. RHR Hx Bypass Flow valve, FCV-3-605, has been adjusted to maintain between 3100 and 3200 gpm RHR flow.

___ 8. Verify one Source Range Nuclear Instrument audible count rate is on in the Control Room when fuel is in the Reactor Vessel.

5.2.2 Procedure Steps

___ 1. Station an operator at Drain Down Level Indicator Hose, LI-3-6422 and verify direct communication with Control Room in order to commence logging level every 15 minutes using Attachment 1. [Commitment - Steps 2.3.6 and 2.3.8]

___ 2. Commence logging reduce inventory parameters using Attachment 2.

___ 3. Place Letdown Diversion Valve, TCV-3-143 to DIVERT.

CAUTIONS

- *RCS level indication may be lower than actual level during RCS draining unless large vent paths are provided.*
- *RCS level indication is connected to Loop A intermediate leg. At high RHR flow rates, the indicated level will be different than actual level at the RHR hot leg suction. Refer to Enclosure 2 for minimum required RCS level indication.*

___ 4. Verify open RHR Letdown Stop, 3-205B.

___ 5. Open RHR LTDN to CVCS, HCV-3-142.

___ 6. Throttle Low Pressure LTDN Controller, PCV-3-145 as necessary to maintain RCS Drain Down Level indication on LIS-3-6421 within 4.0 percent of LIS-3-6423 during RCS draining. [Commitment - Step 2.3.8]

Florida Power & Light Company

Turkey Point Nuclear Plant

CONTROLLED DOCUMENT
NO. # 34

This procedure may be affected by an O T S.C. (On The Spot Change) verify information prior to use
Date verified ________ Initials ________

0-PMI-041.110

Title:

RCS Drain Down Level Calibration

Safety Related Procedure	
Responsible Department:	Maintenance
Revision Approval Date:	3/6/96
Periodic Review Due:	8/29/00

RTS 95-0620, 96-0144
OTSC 0450-95

PC/M 95-150

/OK/ev/sw/ev

INITIALS
CK'D VERIF

6.2 Level Transmitter LT-6421/23 Calibration Check

CAUTION

Care shall be taken not to break the neck seal between the sensor module and the electronics housing.

NOTES

- *Transmitter output Test terminals (local) are located inside the transmitter housing. To gain access, housing side cover identified as Terminal side (see nameplate) must be removed.*
- *Zero and Span adjustment screws are accessible externally and are located behind the transmitter name plate.*
- *The transmitter output increases with clockwise rotation of the adjustment screw.*
- *O-rings shall be replaced if housing cover is removed.*
- *Level Transmitters LT-6421/23 are located inside containment on the 14 foot level outside the bio-wall.*

LT-6421 LT-6423

____ ____ 6.2.1 Shut transmitter high, low, and equalizing valves at 3-valve manifold.

____ ____ 6.2.2 Open transmitter vent valves slowly to release any trapped pressure.

NOTES

- *Transmitter and impulse line liquid contents should be collected in accordance with RWP requirements.*
- *Equalizing valve should never be opened as low side is dry.*

____ ____ 6.2.3 Remove caps from test fittings.

____ ____ 6.2.4 Allow both sides of transmitter to drain.

____ ____ 6.2.5 Connect pressure source and test gauge to high pressure test fitting.

____ ____ 6.2.6 Close vent valve on high pressure side.

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____ ____ 6.3.7 Install cover and hand tighten.

____ ____ 6.3.8 Attach special transmitter cover tool to torque wrench.

INDEPENDENT VERIFICATION POINT

Independent Verifier shall:

- *Verify proper torque application in Step 6.3.9.*
- *Initial appropriate space on Data Sheet.*

____ ____ * 6.3.9 Torque cover to 200 in-lb.

Acceptance Criteria: 190 in-lb to 210 in-lb

____ ____ * 6.3.10 Record O-ring part number and attach QC tag to PWO.

____ ____ 6.3.11 Disconnect and remove pressure test set.

____ ____ 1. Replace defective transmitter test fittings (i.e., Swagelok), if required.

____ ____ 6.3.12 Reinstall test fitting caps and tighten properly.

____ ____ 6.3.13 Reinstall vent valve caps finger tight, do not torque at this time.

6.4 Placing Level Transmitter in Service

____ ____ 6.4.1 Verify that Operations has established a vent path through the pressurizer.

____ ____ 6.4.2 Remove the cap and connect a hose or place a poly bag to catch any fluid from the dry leg low point drain valve below the transmitter three-valve manifold.

____ ____ 6.4.3 Open the dry leg low point drain below the transmitter three-way valve.

____ ____ 6.4.4 Place a poly bag to catch any fluid from the three-valve manifold on test tee downstream of PRZR Safety Valve *-551A (*-551B) Loop Drain *-545A (*-546A). |

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____ ____ 6.4.5 Disconnect dry leg tubing at **B** valve on three-valve manifold or remove test "T" cap downstream of *-545A (*-546A).

NOTE

If test tee is used, have Operations close "B" valve on three-valve manifold.

____ ____ 6.4.6 Connect a source of dry nitrogen or instrument air to the test tee on dry leg tubing on the 58 foot level and blow down to the drain valve on the 14 foot level until all moisture is removed from the line.

____ ____ 6.4.7 If test tee was not used, disconnect the blowdown connection from the dry leg tubing.

NOTE

If test tee is used, have Operations close "B" valve on three-valve manifold.

____ ____ 6.4.8 Place a poly bag to catch any fluid below the pipe cap or drain valve downstream of *-545A (*-546A).

____ ____ 6.4.9 Remove the pipe cap or open the drain valve downstream of *-545A (*-546A) and drain any moisture.

____ ____ 6.4.10 Replace the pipe cap or close the drain valve downstream of *-545A (*-546A).

____ ____ 6.4.11 If test tee was not used, connect a source of dry nitrogen or instrument air to the **B** valve on three-valve manifold downstream of *-545A (*-546A).

____ ____ 1. Blow into the pressurizer until all moisture is removed from the line.

____ ____ 6.4.12 Disconnect the blowdown connection.

____ ____ 6.4.13 Reconnect the dry leg tubing to the **B** valve connection or replace test "T" cap on the three-valve manifold.

____ ____ 6.4.14 Verify closed equalizing valve on three-valve manifold at transmitter.

____ ____ 6.4.15 Remove cap from transmitter low side vent.

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NOTE

Dry nitrogen or instrument air may be used to assist in draining the low side of the transmitter. Direct all flow down from transmitter vent to low point drain.

___ ___ 6.4.16 Slowly open low side isolation valve and allow transmitter to drain.

___ ___ 6.4.17 Close the dry leg low point drain valve below the transmitter three-valve manifold.

___ ___ 6.4.18 Remove the hose if used, and replace the cap on the low point drain valve.

___ ___ 6.4.19 Replace cap on transmitter low side vent.

___ ___ 6.4.20 Remove cap on high side test tee above transmitter three-valve manifold. Install a length of hose from the test tee into a poly bag.

NOTE

It may be necessary to bleed several gallons of fluid before all air is removed from the line.

___ ___ 6.4.21 Slowly open the high side isolation valve and drain RCS fluid into poly bag until all air is removed from line. Open the valve fully to obtain the maximum flow rate.

___ ___ 6.4.22 Close the high side isolation valve.

___ ___ 6.4.23 Remove the hose and replace the test tee cap.

___ ___ 6.4.24 Remove the transmitter high side vent cap.

___ ___ 6.4.25 Slowly open the high side isolation valve and fill the transmitter.

___ ___ 6.4.26 Close the high side isolation valve.

___ ___ 6.4.27 Replace cap on transmitter high side vent.