NRC Form 366
(9-83)

U.S. NUCLEAR REGULATORY COMMISSION
APPROVED OMB NO. 3150-0104
EXPIRES 8/31/85

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Browns Ferry - Unit 3	0 5 0 0 0 2 9 6	1 OF 0 2

TITLE (4) Reactor Manual Scram Due to Condensate Booster Pumps Failure to Start During Unit Startup

EVENT DATE (5) MONTH	DAY	YEAR	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	REPORT DATE (7) MONTH	DAY	YEAR	OTHER FACILITIES INVOLVED (8) FACILITY NAMES	DOCKET NUMBER(S)
1 1	2 0	8 4	8 4	0 1 2	0 0	1 2	2 0	8 4		0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9): N

POWER LEVEL (10): 0 0 5

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)	[ ] OTHER (Specify in Abstract below and in Text, NRC Form 366A)
[ ] 20.405(a)(1)(iii)	[X] 50.73(a)(2)(i)	[ ] 50.73(a)(2)(viii)(A)	
[ ] 20.405(a)(1)(iv)	[ ] 50.73(a)(2)(ii)	[ ] 50.73(a)(2)(viii)(B)	
[ ] 20.405(a)(1)(v)	[ ] 50.73(a)(2)(iii)	[ ] 50.73(a)(2)(x)	

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER AREA CODE	
Jimmy B. Walker	2 0 5	7 2 9 - 2 5 3 6

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

CAUSE	SYSTEM	COMPONENT	MANUFAC TURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFAC TURER	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)

Unit 3 was manually scrammed by the licensed reactor operator when control room indication showed that the vessel low water level scram set point had been exceeded. Master Refueling Test Instruction requires functional testing of the main steam relief valves per approved Surveillance Instruction (SI).

During performance of the SI, the reactor water level decreased close to the Technical Specification limit and the unit was manually scrammed.

No safety limits were exceeded. Procedure inadequacy was the root cause for the event.

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

Unit 1 was operating at 100 percent power, unit 2 was in a refueling outage, and unit 3 was in startup mode at approximately 4.5 percent power. Unit 3 was the only unit affected by this event.

On November 20, 1984, during preparation for performance of relief valve (RV) functional surveillance testing the licensed reactor operator manually scrammed the unit. Master refueling test instruction requires performance of the functional surveillance instruction. In preparation for the test, the pressure set point controller was lowered to 270 psig thereby opening approximately 1.5 bypass valves (V). The reactor vessel water level began decreasing due to the increased steam flow through the bypass valves. Condensate booster pumps (P) 3B and 3C were attempted to be placed in service to increase feedwater flow without success. The pumps local start switches were found to be set in the SAFE/STOP position. Plant procedure did not have a checkoff verification for proper position for the start switches. The operating procedures did not allow for insertion of reactor control rods (R). The operator felt that raising the pressure set point by closing the bypass valves would cause a pressure spike and drive the reactor water level lower.

When the reactor pressure vessel level indication by the control room level instruments decreased below plus 11 inches without an automatic scram, the shift engineer directed the unit operator to manually scram the reactor.

The contol room level instruments do not input to the reactor protection system (JE). Independent scram switches are used for the reactor protection system automatic scram for reactor vessel low level. Reactor level instrumentation is calibrated for rated temperature and pressure. At less than rated conditions, indicated level and actual level may not agree exactly. Nuclear Engineering calculations show that the actual level at the time of the scram was 12 inches and not 8 inches as indicated by the control room instruments (see attached). The scram instrumentation had been calibrated the morning prior to the event and would have actuated to provide the required low water level scram. These instruments were again checked following the event and vertified to be within calibration.

Procedural inadequacy relating to standby lineup of the condensate booster pumps were the root cause of the event. Necessary procedures for startup have been revised.

No further corrective action is planned.

Responsible Plant Section - OP

Previous Similar Events - None

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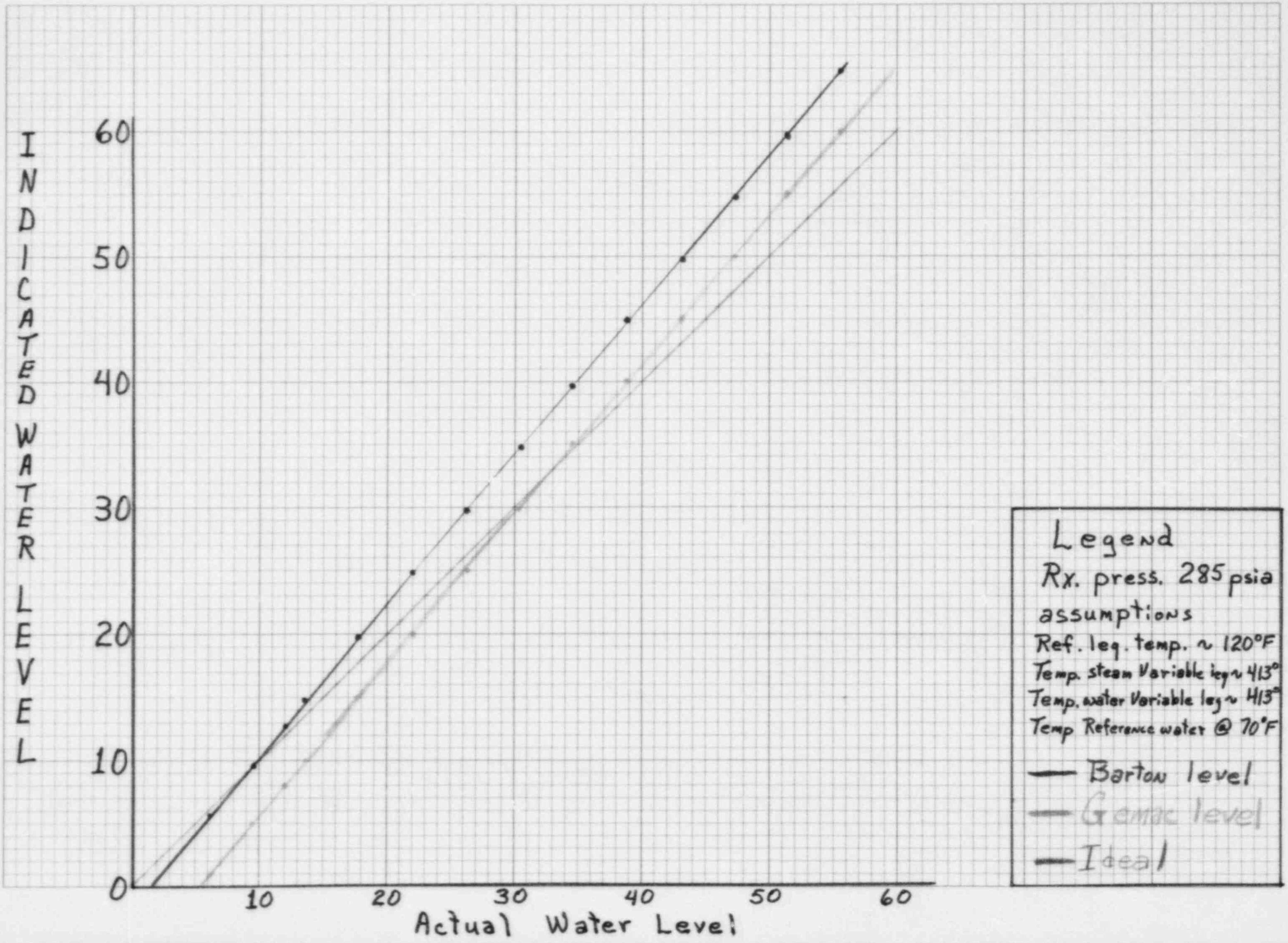
ATTACHMENT 2 - LER 50-296/84012



ATTACHMENT 3 - LER 50-296/84012

Problem: Determine the actual level and the level sensed by the scram Barton switches based on an indicated level of +8" on the GEMACS.

Reference: Reactor Water Level Calculations (June 1979) BFNP File Document 364 L20

BF SCI 234

Given: Indicated level on GEMACS = 8"
Reference leg temperature = 120°F (ASSUMED)
RPV pressure = 270 psig (285 psia)

Assumptions:

Avg. Temperature of reference leg = 120°F D_1 = 61.71 lbm/ft³
Temperature of steam in variable leg = 413°F D_2 = .6159 lbm/ft³
Avg. Temperature of water in variable leg = 413°F D_3 = 53.12 lbm/ft³
Temperature of reference water = 70°F D_4 = 62.30 lbm/ft³

Determine ma output of 6-72 A,B

$$\frac{40\,\text{ma}}{60''} \times \underline{8''} + 10\,\text{ma} = \underline{15.33\,\text{ma}}$$

Determine Ma input #2 to 6-72 A,B

$$\underline{270\,\text{psig}} \times \frac{40\,\text{ma}}{1500\,\text{psig}} + 10\,\text{ma} = \underline{17.2\,\text{ma}}$$

Output of prop. Amps 6-72A,B is given by:

$[K_1(\text{level}) + K_2(\text{Pressure}) - IB] \cdot G + OB = \text{Output}$

where:

$K_1 = 0.8$ scaling factor (level)
$K_2 = 0.125$ scaling factor (pressure)
$IB = 12.75$ (Input Bias)
$OB = 10.00$ (Output Bias)
$G = 1.25$ (Gain)

$[.8 X_L + .125(17.2) - 12.75]\, 1.25 + 10 = 15.33$ ma
$[.8 X_L + 2.15 - 12.75]\, 1.25 + 10 = 15.33$

$X_L = \underline{18.58 \text{ ma}}$

Determine inches H_2O input to 6-52A,B level sensors to yield output of 18.58 ma.

$\frac{60''}{40\text{ma}}(18.58 \text{ ma} - 10\text{ma}) = \boxed{12.87 \text{ inches}}$

This is level scram Bartons saw.

Determine the actual level in RPV:

Convert sensed level to ΔP in inches H_2O

$\underline{12.87}$ inches $\times \frac{[-29.16 - (-71.59)]}{60 \text{ inches}} + (-71.59) = \underline{-62.49}$ "H_2O ΔP

Determine actual level

$$\Delta P = (Act - 517)\frac{D_3}{D_4} + (600.5 - Act)\frac{D_2}{D_4} - (600.5 - 517)\frac{D_1}{D_4}$$

$$= (Act - 517)\frac{53.12}{62.30} + (600.5 - Act)\frac{-6159}{62.30} - (83.5)\frac{61.71}{62.30}$$

$$= .8428\,Act - 440.82 + 5.94 - 82.71$$

$$-62.49 = .8428\,Act - 440.82 + 5.94 - 82.71$$

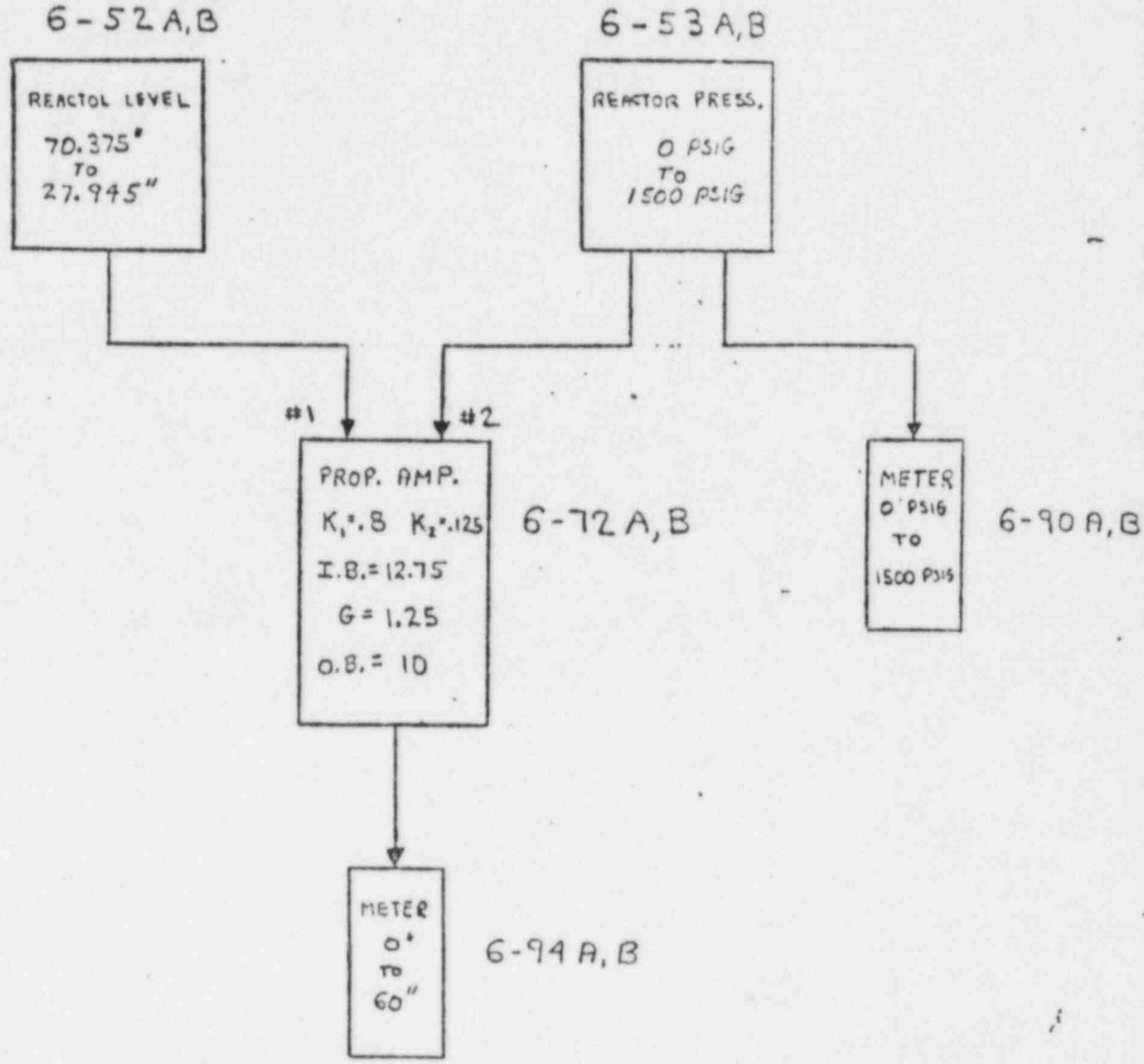
$$455.10 = .8428\,Act$$

$Act = 540''$ above vessel zero

Actual Water level relative to Instrument Zero is
$540 - 528 = \boxed{12.0}$ inches Actual Water level.

Performed by: J.D. Thompson + E.D. Nave
Reviewed by: B.T. Williamson

FEEDWATER CONTROL SYSTEM LEVEL



6-52A,B LEVEL XTMR.	71.59" TO 29.16" INPUT 10 MA TO 50 MA OUTPUT	
6-53A,B PRESS. XTMR.	0 PSIG TO 1500 PSIG INPUT 10 MA TO 50 MA OUTPUT	
6-72A,B PROP. AMP.	K_1=.8, K_2=.125 IB=12.75 G=1.25 O.B.=10	

TENNESSEE VALLEY AUTHORITY
Browns Ferry Nuclear Plant
P. O. Box 2000
Decatur, Alabama 35602

December 20, 1984

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

Dear Sir:

TENNESSEE VALLEY AUTHORITY - BROWNS FERRY NUCLEAR PLANT (BFN) UNIT 3 - DOCKET NO. 50-296 - FACILITY OPERATING LICENSE DPR-68 - REPORTABLE OCCURRENCE REPORT BFRO-50-296/84012

The enclosed report provides additional details concerning reactor manual scram due to condensate booster pumps failure to start during unit startup. This report is submitted in accordance with 10 CFR 50.73 (a)(2)(i) and 50.73 (a)(2)(iv).

Very truly yours,

TENNESSEE VALLEY AUTHORITY

JESwindell

for G. T. Jones
Plant Manager
Browns Ferry Nuclear Plant

Enclosures
cc (Enclosures):
Regional Administrator
U. S. Nuclear Regulatory Commission
Office of Inspection and Enforcement
Region II
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Atlanta, Georgia 30303

INPO Records Center
Suite 1500
1100 Circle 75 Parkway
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NRC Resident Inspector, BFN

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