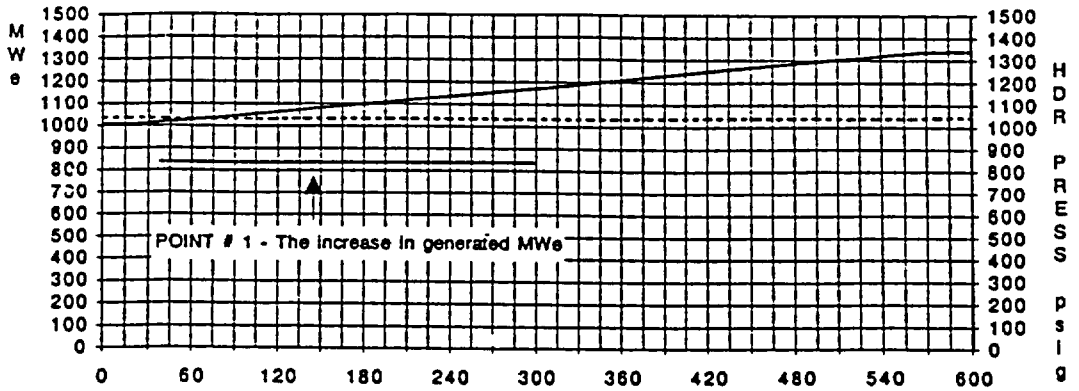
Pressurized Water Reactor
B&W Technology
Crosstraining Course Manual

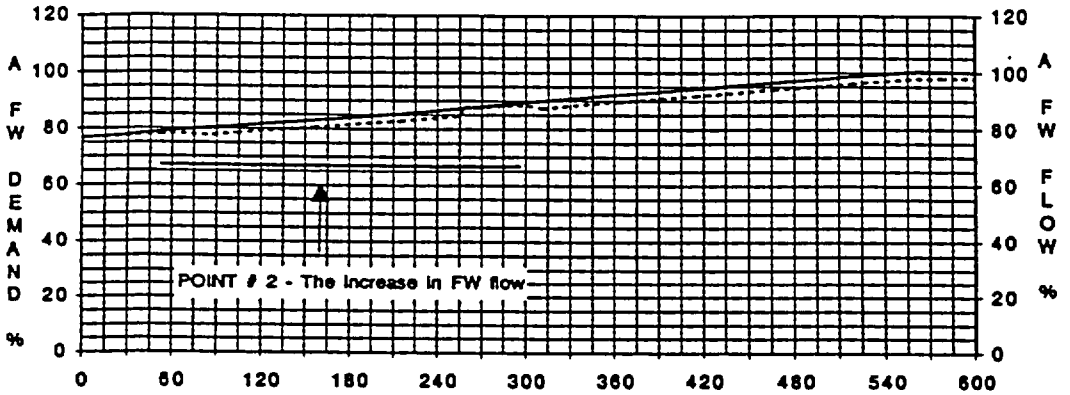
Chapter 19.0

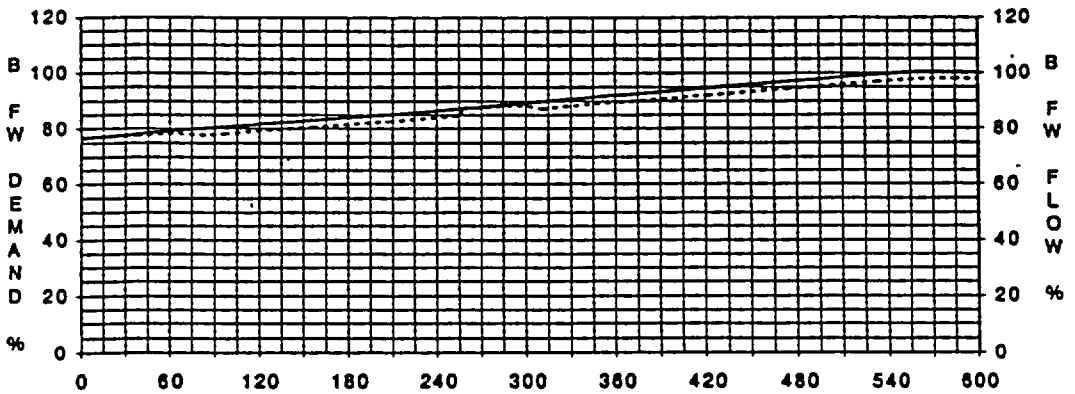
Transients

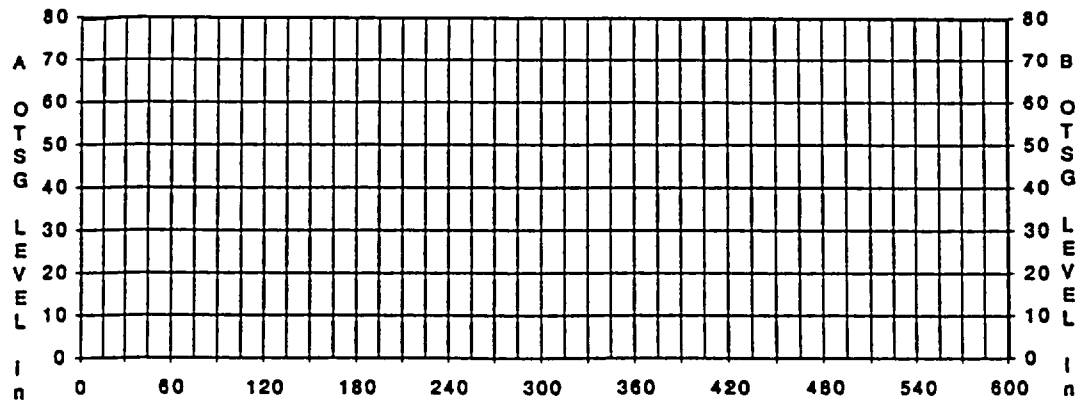
Transient 1

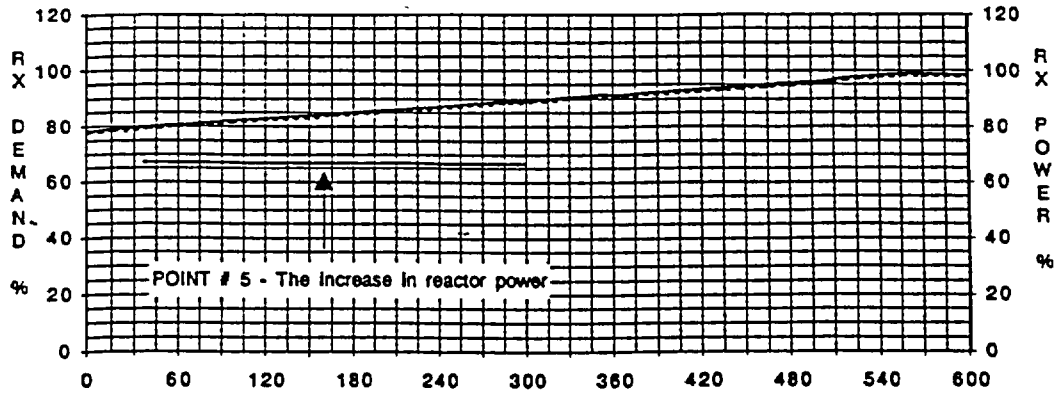
Ramp Increase in Power from 75% to 100% at 2%/Min.

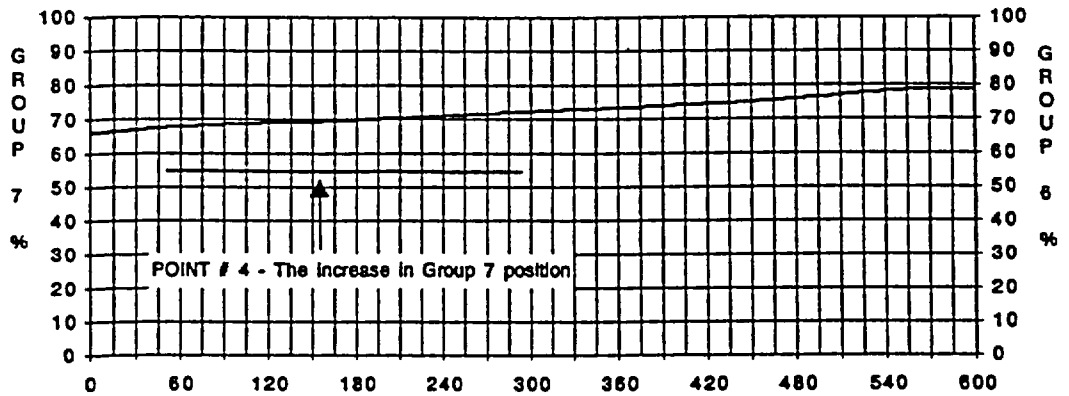

MWe
HDR PRESS psig
POINT # 1 - The increase in generated MWe

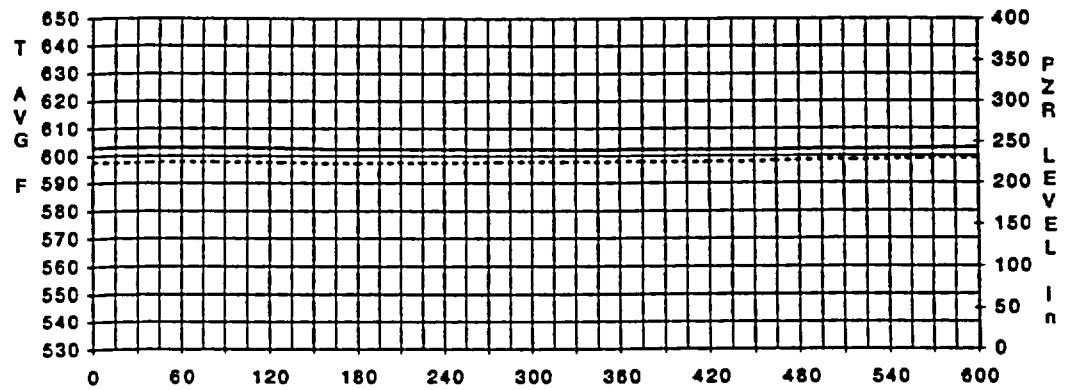

A FW DEMAND %
A FW FLOW %
POINT # 2 - The increase in FW flow

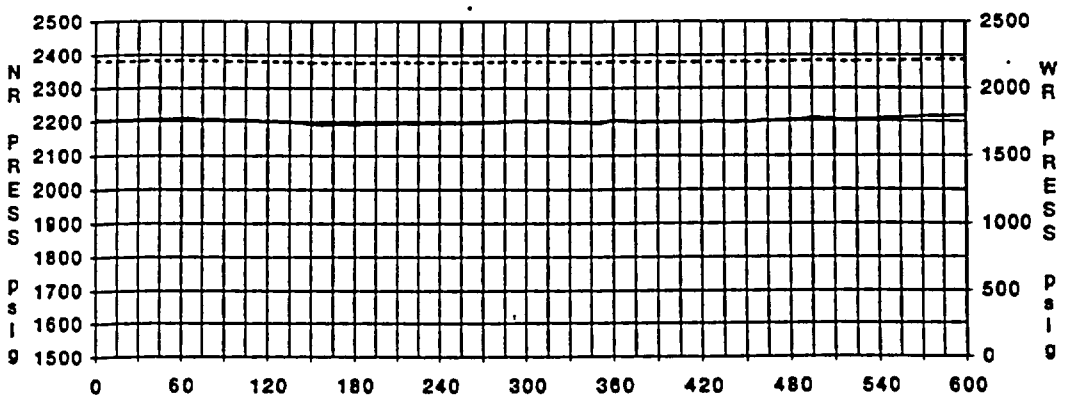

B FW DEMAND %
B FW FLOW %

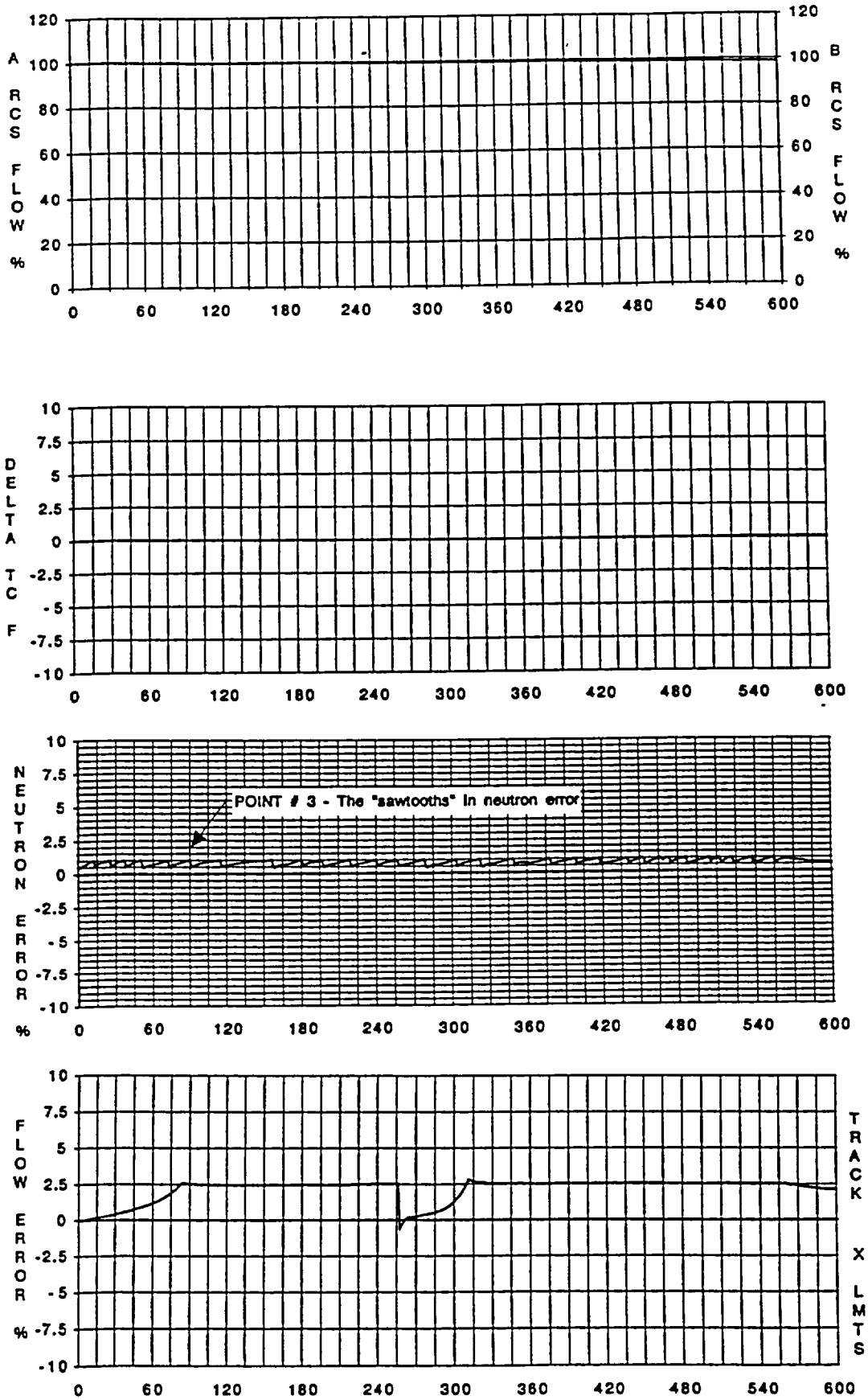

A OTSG LEVEL in
B OTSG LEVEL in


RX DEMAND %
RX POWER %
POINT # 5 - The increase in reactor power
0 60 120 180 240 300 360 420 480 540 600


GROUP 7 %
GROUP 6 %
POINT # 4 - The increase in Group 7 position
0 60 120 180 240 300 360 420 480 540 600

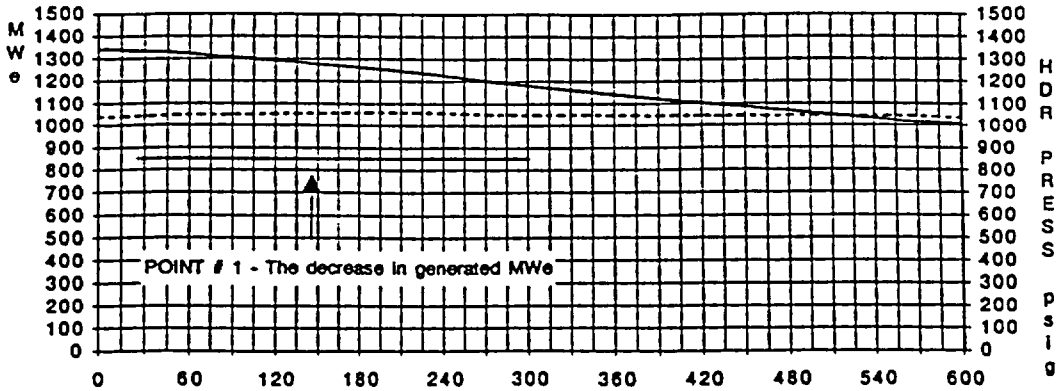

T AVG F
PZR LEVEL in
0 60 120 180 240 300 360 420 480 540 600

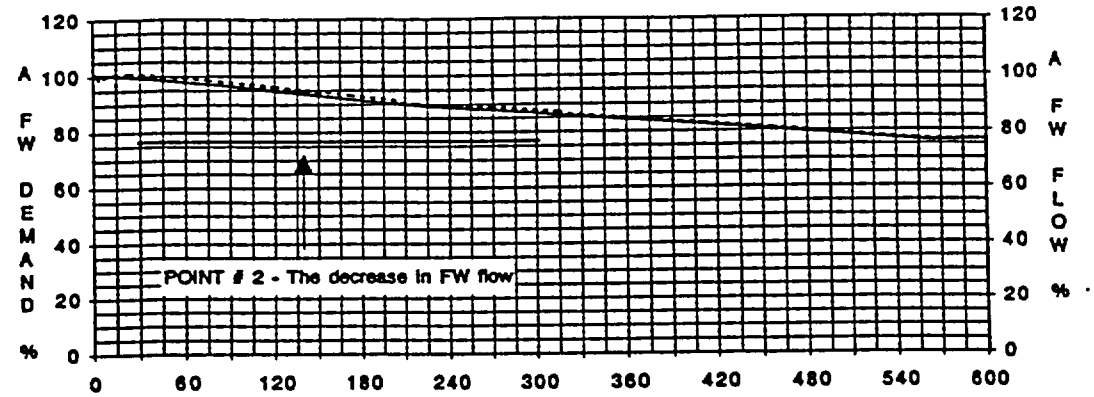

NR PRESS psig
WR PRESS psig
0 60 120 180 240 300 360 420 480 540 600

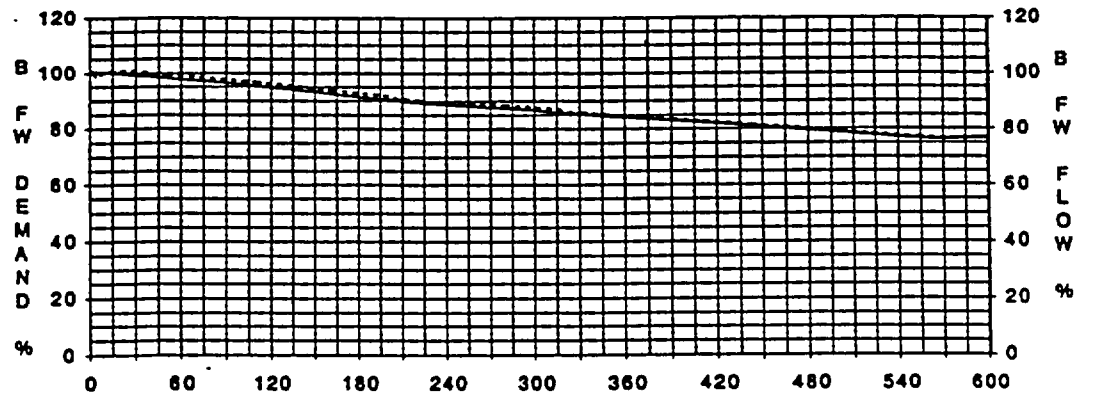

A RCS FLOW %
B RCS FLOW %
DELTA TCF
NEUTRON ERROR %
POINT # 3 - The "sawtooths" in neutron error
FLOW ERROR %
TRACK X LMTS
0 60 120 180 240 300 360 420 480 540 600

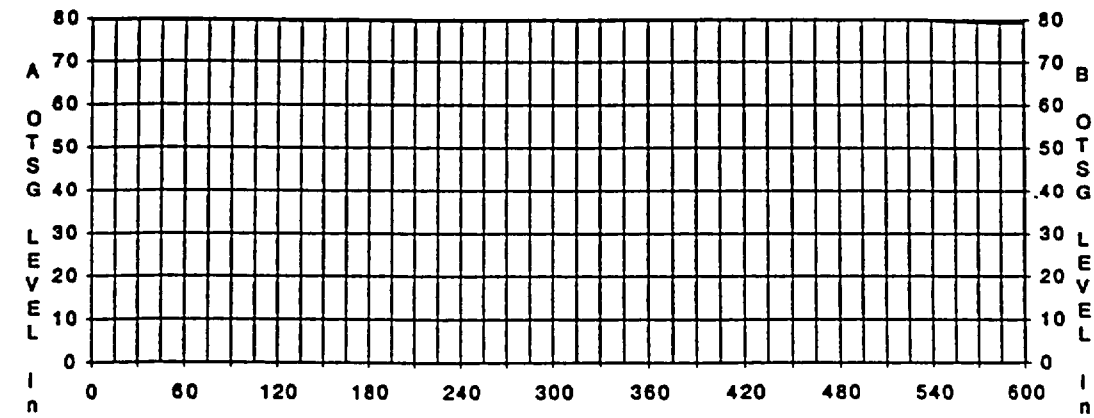
Transient 2

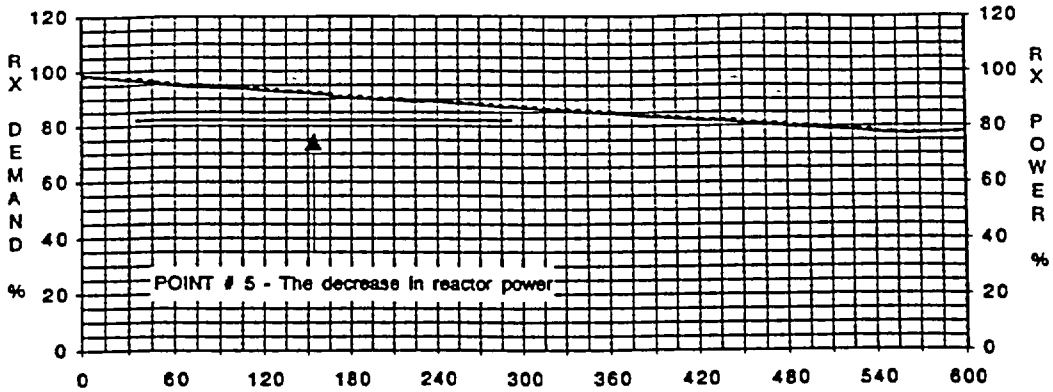
Ramp Decrease in Power from 100% to 75% at 2%/Min.

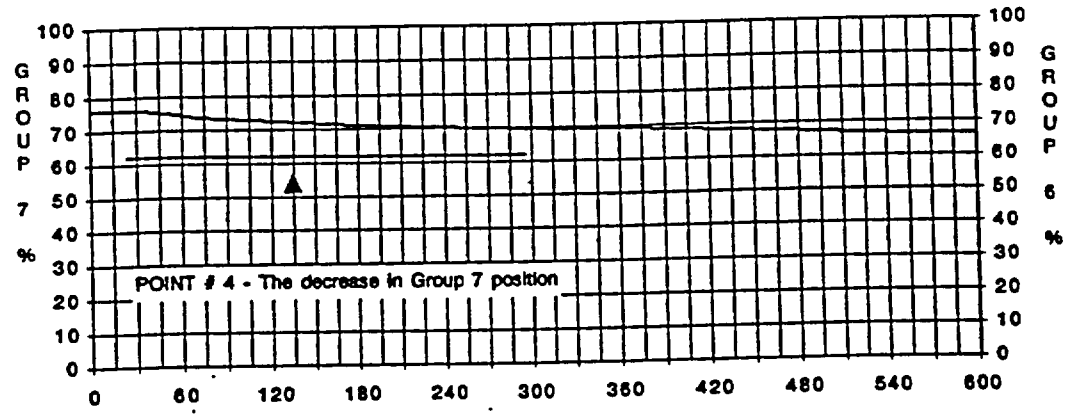

MWe
HDR PRESS psig
POINT # 1 - The decrease in generated MWe

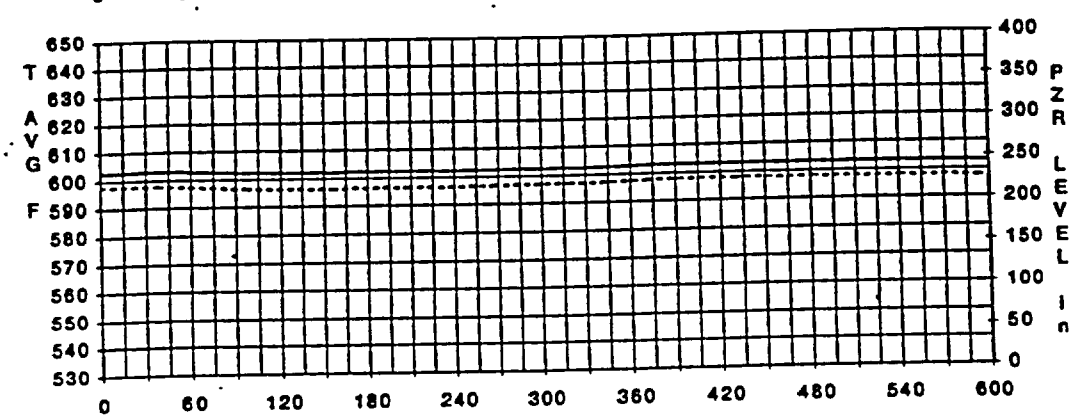

A FW DEMAND %
A FW FLOW %
POINT # 2 - The decrease in FW flow

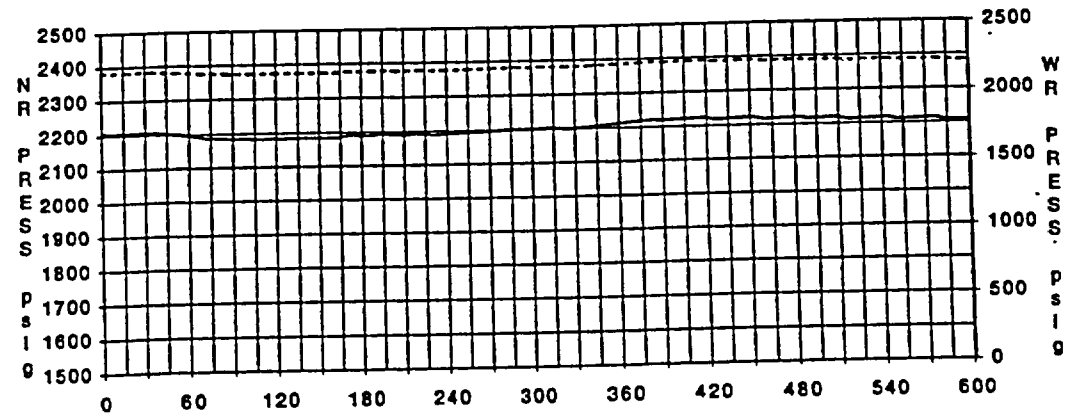

B FW DEMAND %
B FW FLOW %

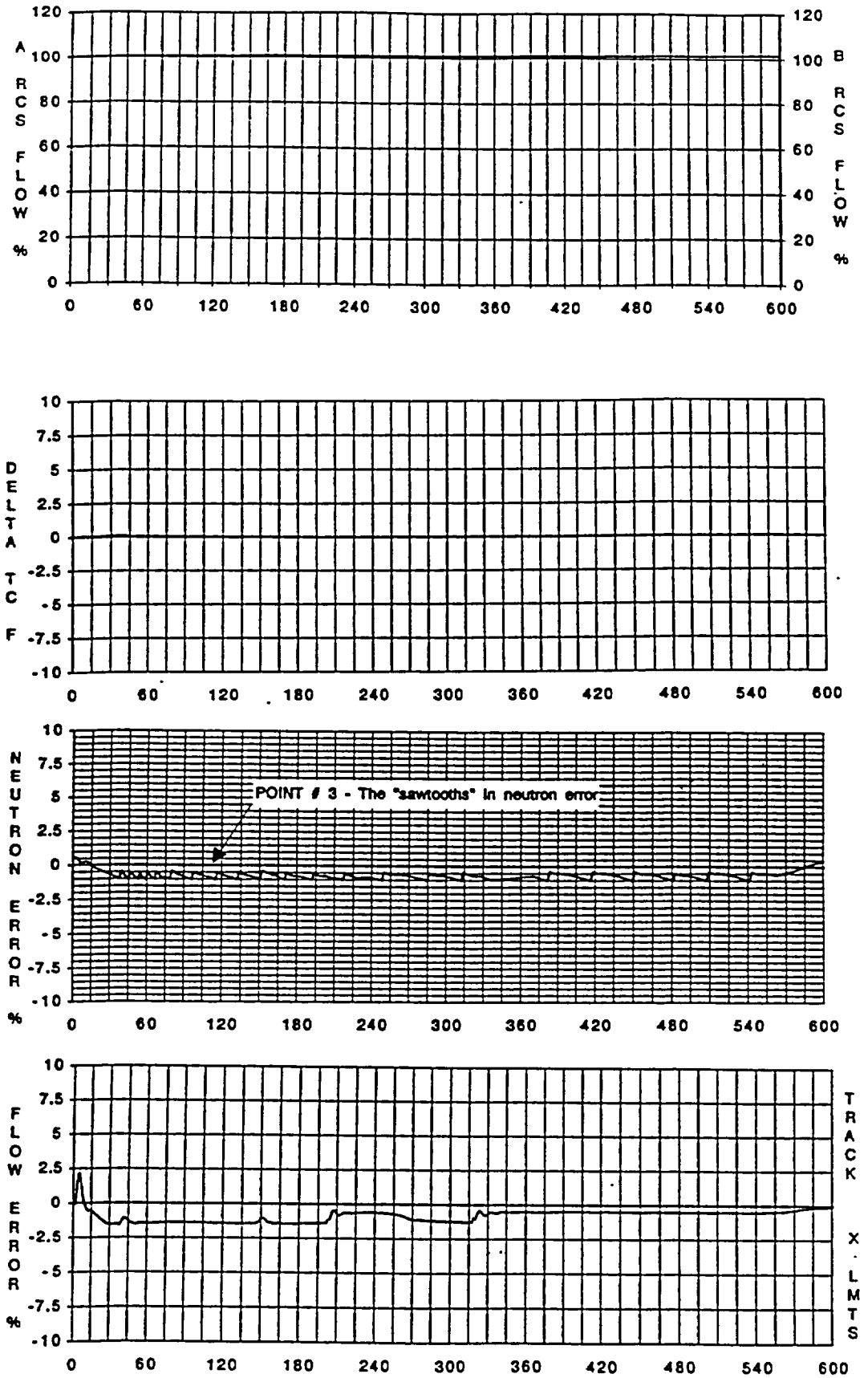

A OTSG LEVEL in
B OTSG LEVEL in


RX DEMAND %
RX POWER %
POINT # 5 - The decrease in reactor power
0 60 120 180 240 300 360 420 480 540 600


GROUP 7 %
GROUP 6 %
POINT # 4 - The decrease in Group 7 position
0 60 120 180 240 300 360 420 480 540 600

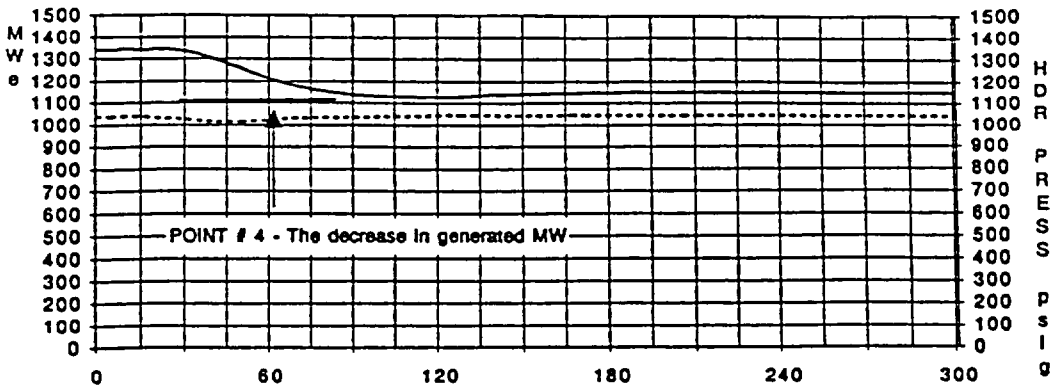

T AVG F
PZR LEVEL in
0 60 120 180 240 300 360 420 480 540 600

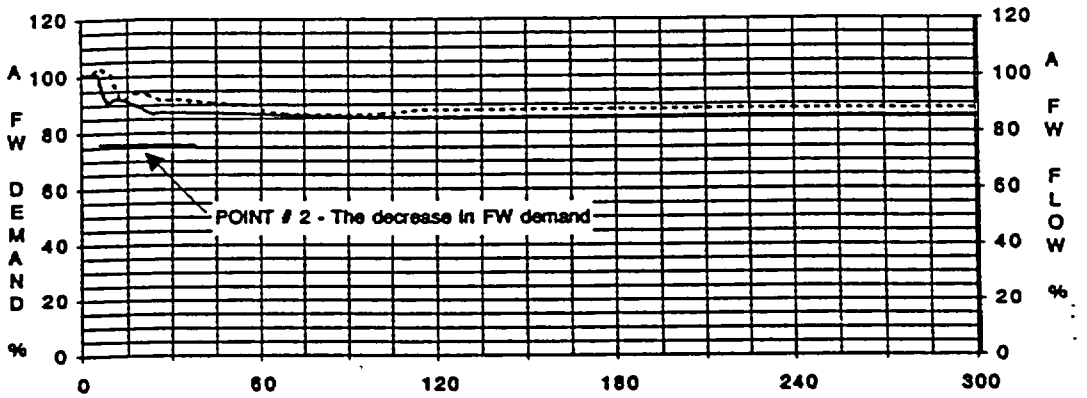

NR PRESS psig
WR PRESS psig
0 60 120 180 240 300 360 420 480 540 600

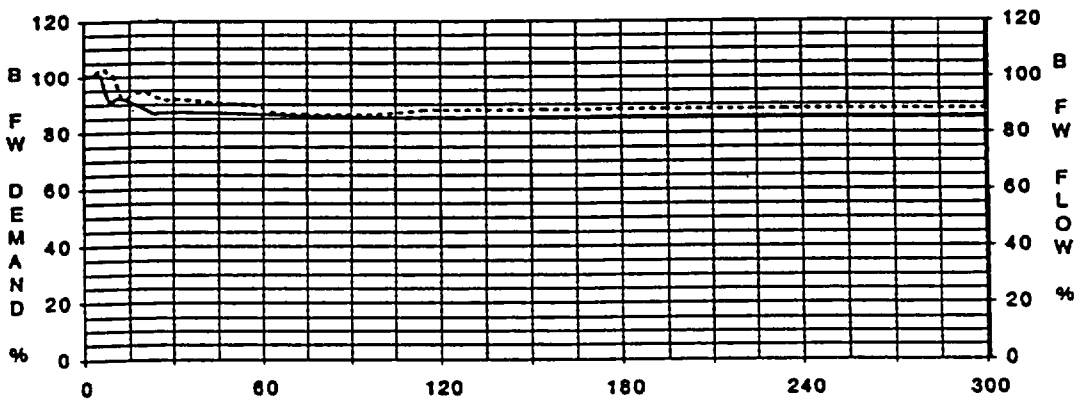

A RCS FLOW %
B RCS FLOW %
DELTA TCF
NEUTRON ERROR %
POINT # 3 - The "sawtooths" in neutron error
FLOW ERROR %
TRACK X-LMTS
0 60 120 180 240 300 360 420 480 540 600

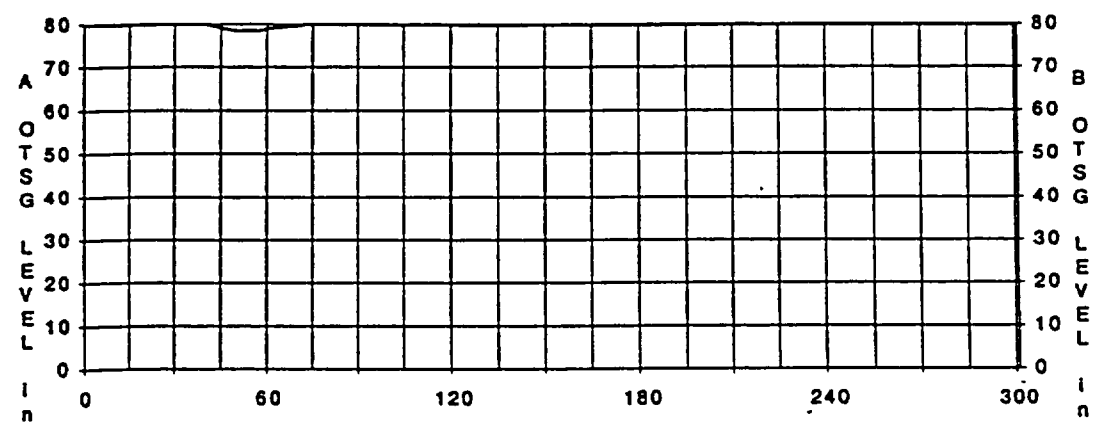
Transient 3

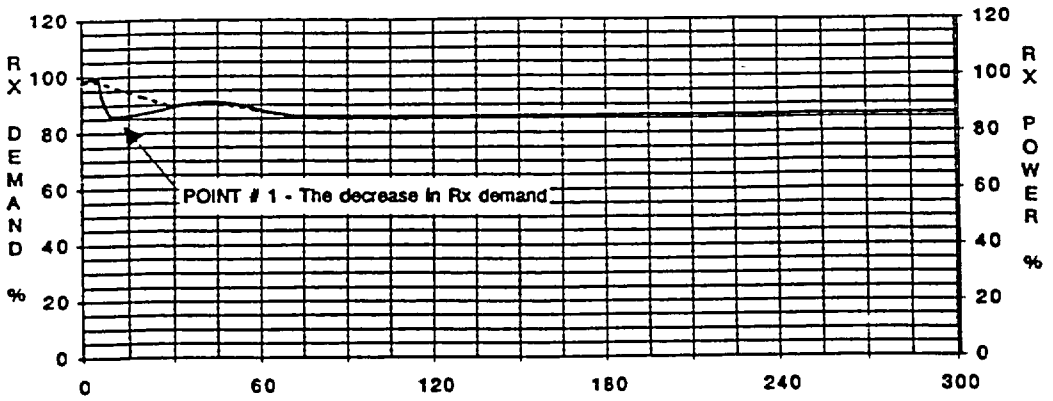
Ramp Decrease 100%-85%, SG/RX in Manual

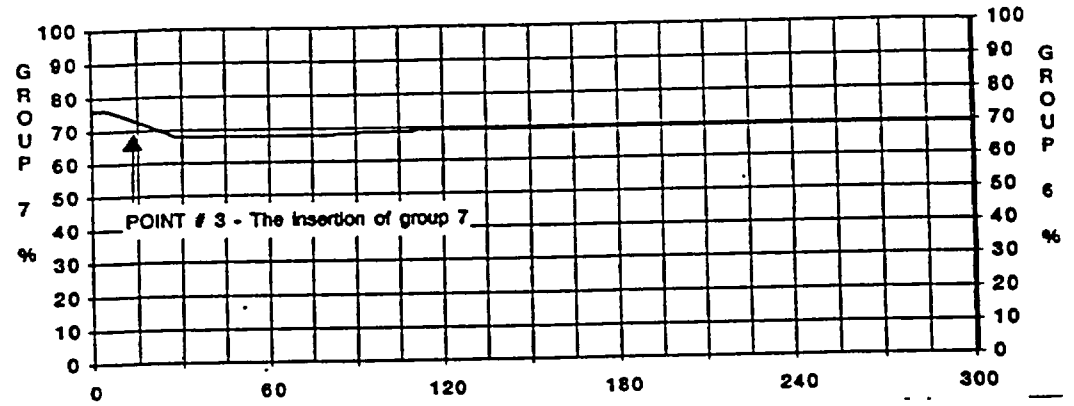

MWe
HDR PRESS psig
POINT # 4 - The decrease in generated MW
0
60
120
180
240
300

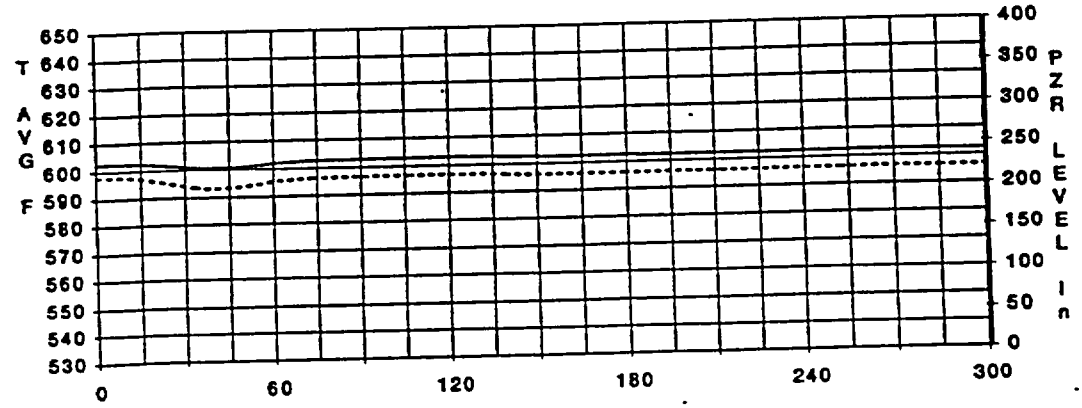

A FW DEMAND %
A FW FLOW %
POINT # 2 - The decrease in FW demand
0
60
120
180
240
300

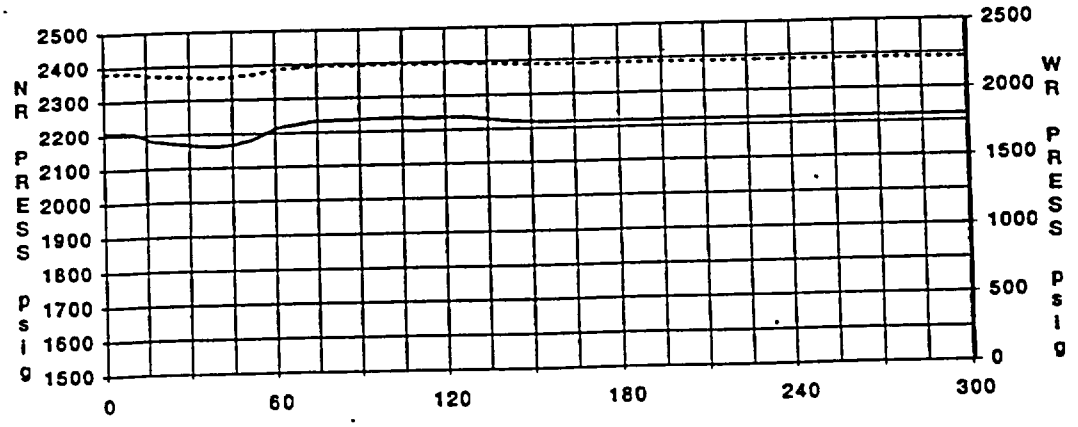

B FW DEMAND %
B FW FLOW %
0
60
120
180
240
300

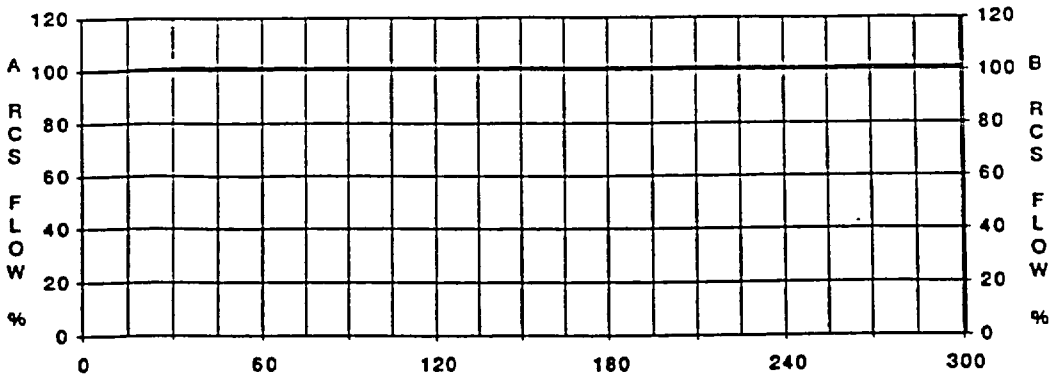

A OTSG LEVEL in
B OTSG LEVEL in
0
60
120
180
240
300

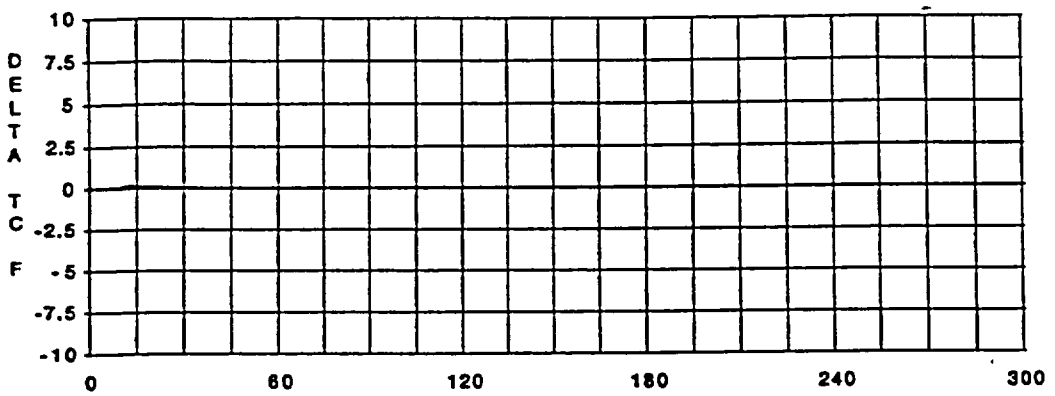

RX DEMAND %
RX POWER %
POINT # 1 - The decrease in Rx demand
0
20
40
60
80
100
120
0
60
120
180
240
300

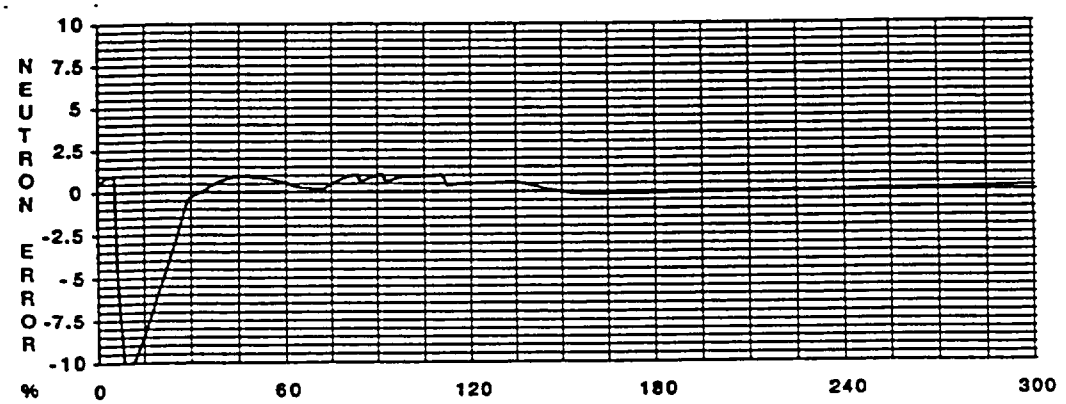

GROUP 7 %
GROUP 6 %
POINT # 3 - The insertion of group 7
0
10
20
30
40
50
60
70
80
90
100
0
60
120
180
240
300

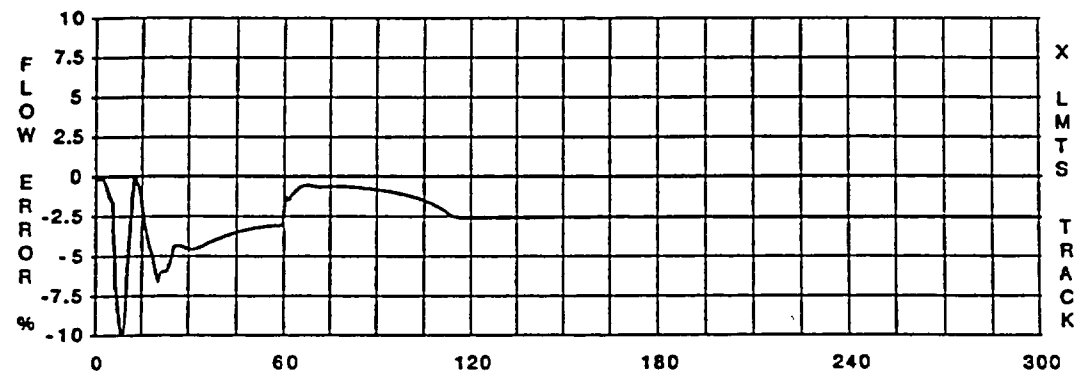

T AVG F
PZR LEVEL in
530
540
550
560
570
580
590
600
610
620
630
640
650
0
50
100
150
200
250
300
350
400
0
60
120
180
240
300


NR PRESS psig
WR PRESS psig
1500
1600
1700
1800
1900
2000
2100
2200
2300
2400
2500
0
500
1000
1500
2000
2500
0
60
120
180
240
300


A RCS FLOW %
B RCS FLOW %
120
100
80
60
40
20
0
0
60
120
180
240
300

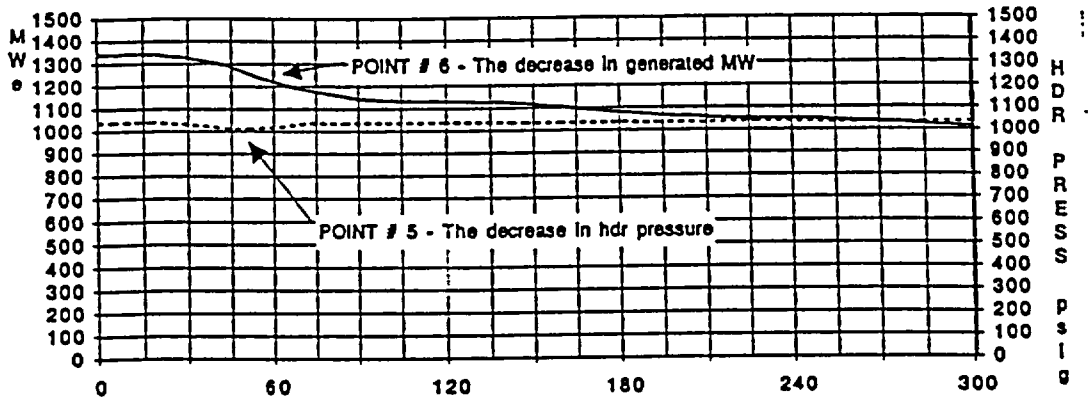

DELTA TCF
10
7.5
5
2.5
0
-2.5
-5
-7.5
-10
0
60
120
180
240
300

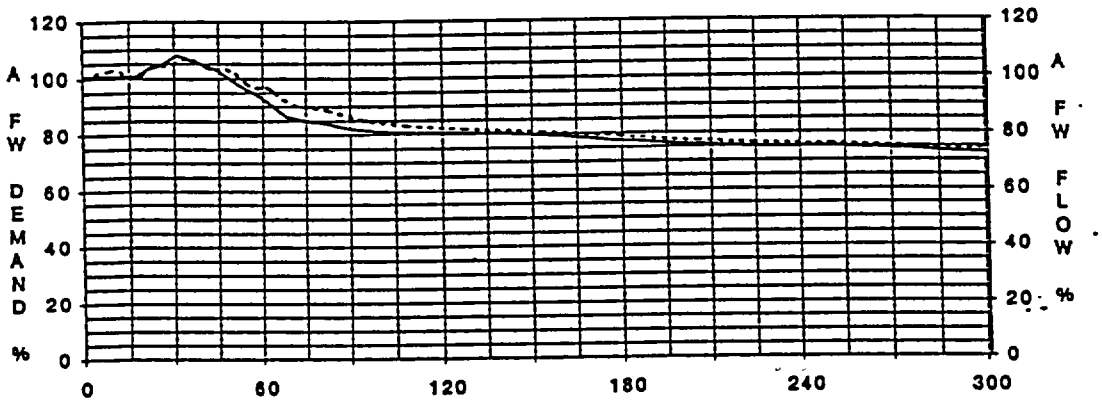

NEUTRON ERROR %
10
7.5
5
2.5
0
-2.5
-5
-7.5
-10
0
60
120
180
240
300

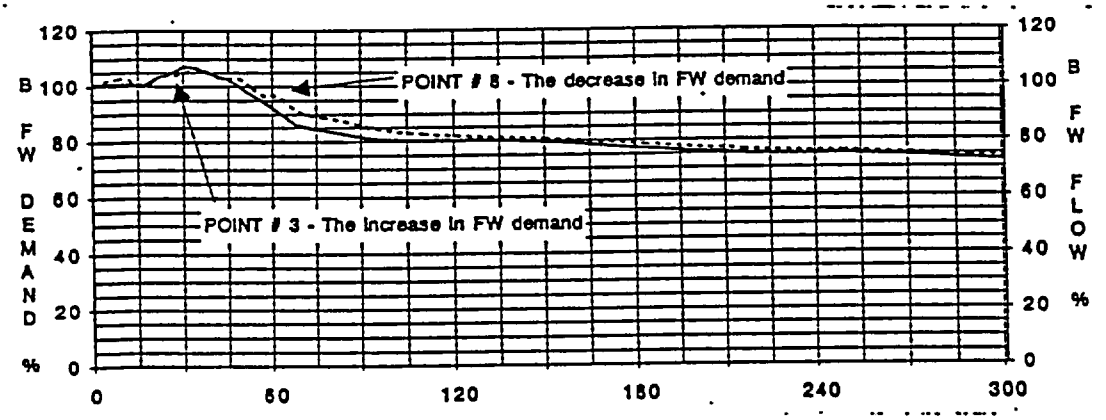

FLOW ERROR %
X LMTS TRACK
10
7.5
5
2.5
0
-2.5
-5
-7.5
-10
0
60
120
180
240
300

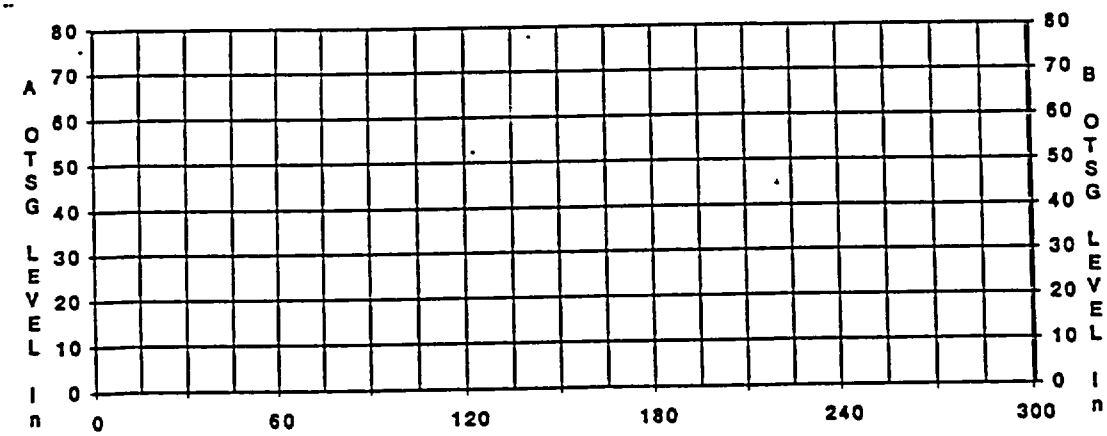
Transient 4

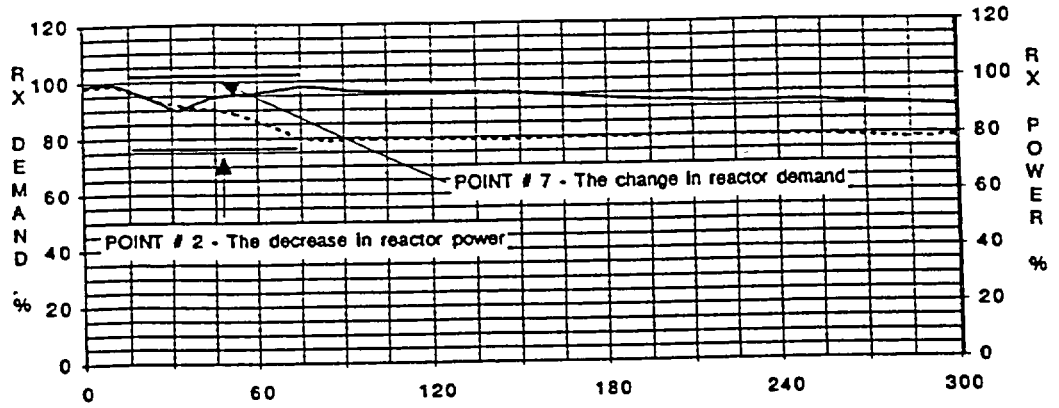
Ramp Decrease 100% to 90% - Reactor Demand in Manual

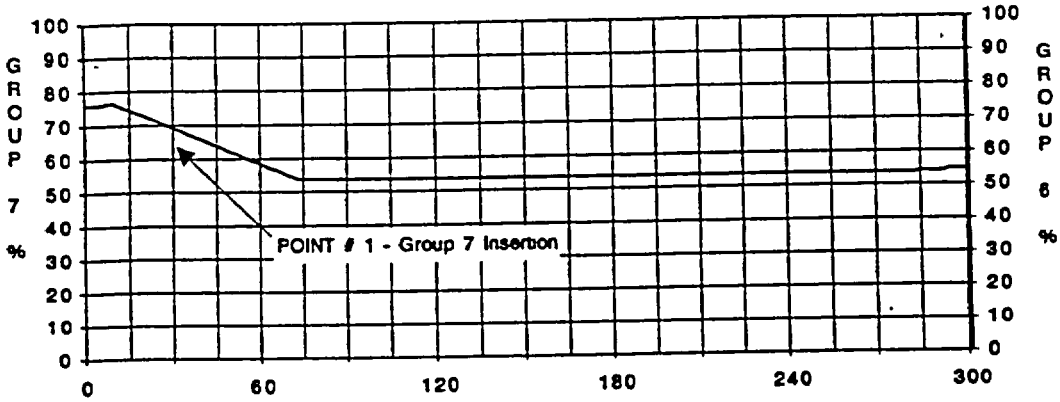

MWe
HDR PRESS psig
POINT # 6 - The decrease in generated MW
POINT # 5 - The decrease in hdr pressure
0 60 120 180 240 300

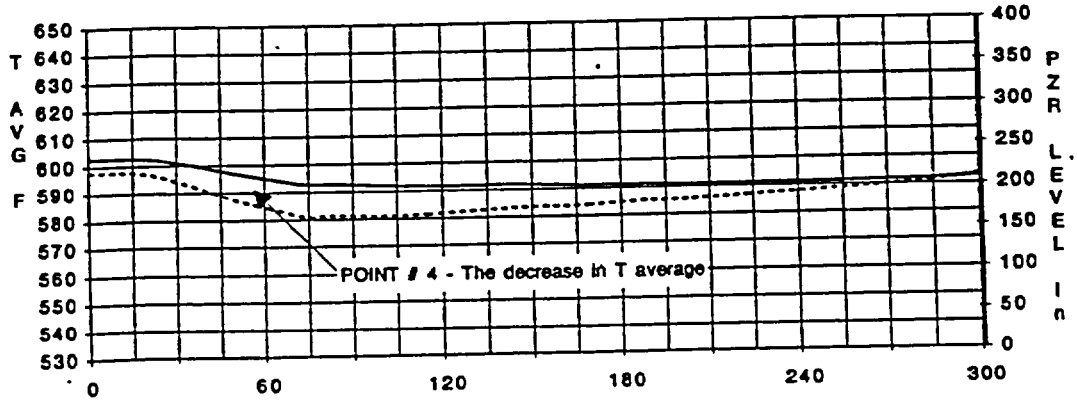

A FW DEMAND %
A FW FLOW %
0 60 120 180 240 300

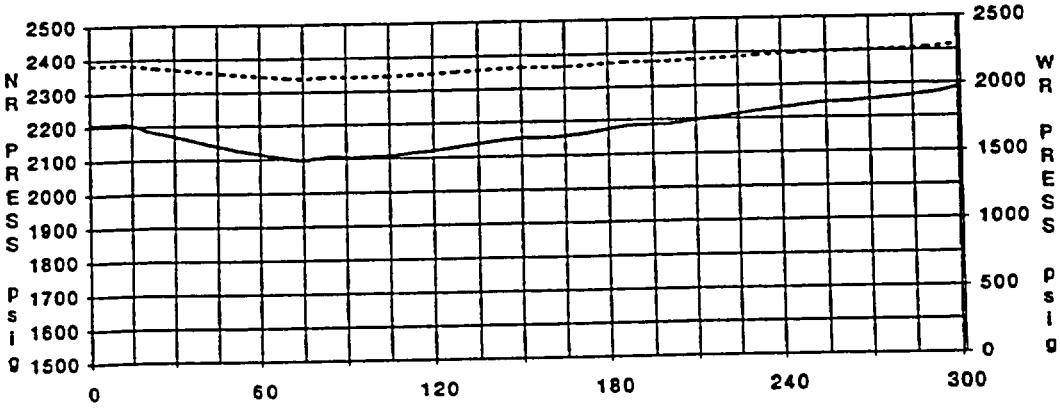

B FW DEMAND %
B FW FLOW %
POINT # 8 - The decrease in FW demand
POINT # 3 - The increase in FW demand
0 60 120 180 240 300

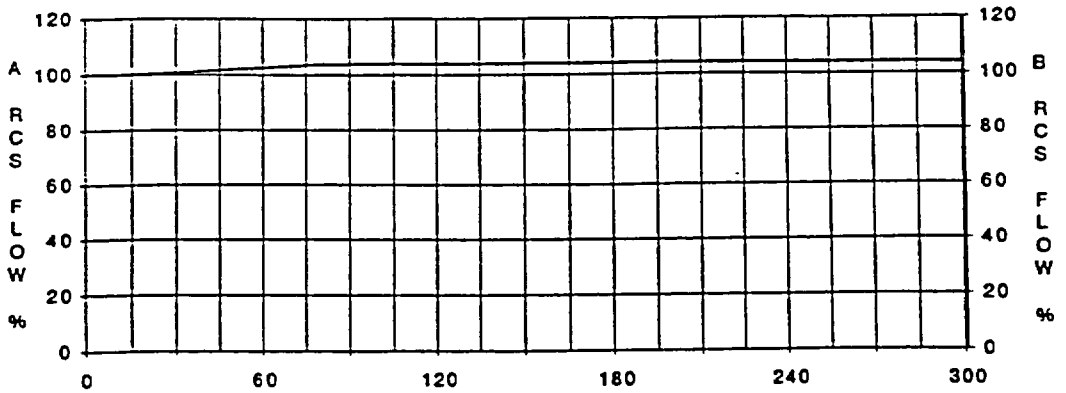

A OTSG LEVEL In
B OTSG LEVEL In
0 60 120 180 240 300

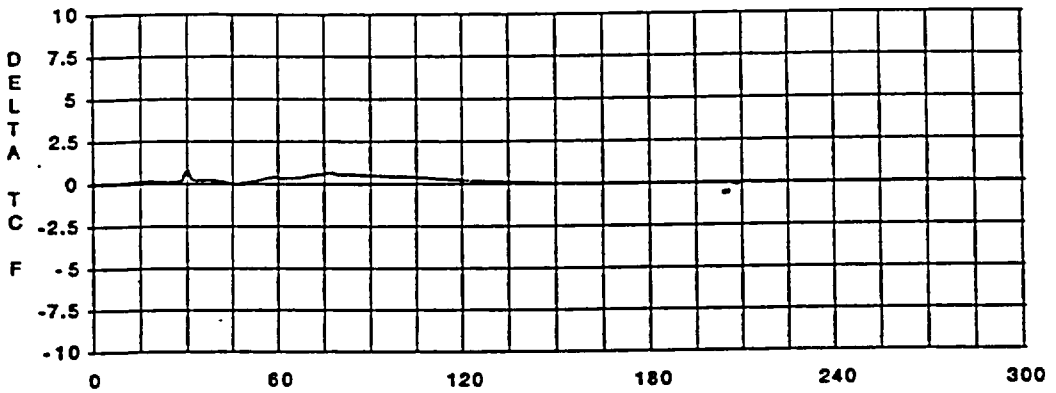

RX DEMAND %
RX POWER %
POINT # 7 - The change in reactor demand
POINT # 2 - The decrease in reactor power
0
60
120
180
240
300

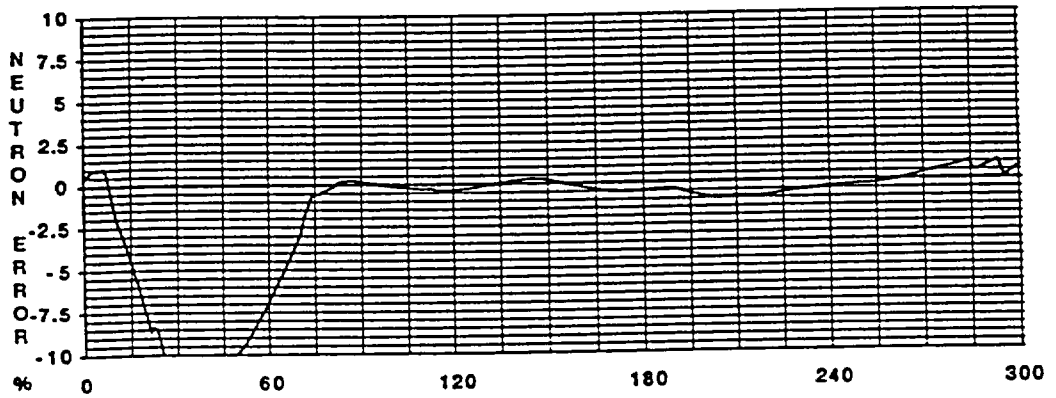

GROUP 7 %
GROUP 6 %
POINT # 1 - Group 7 Insertion
0
60
120
180
240
300

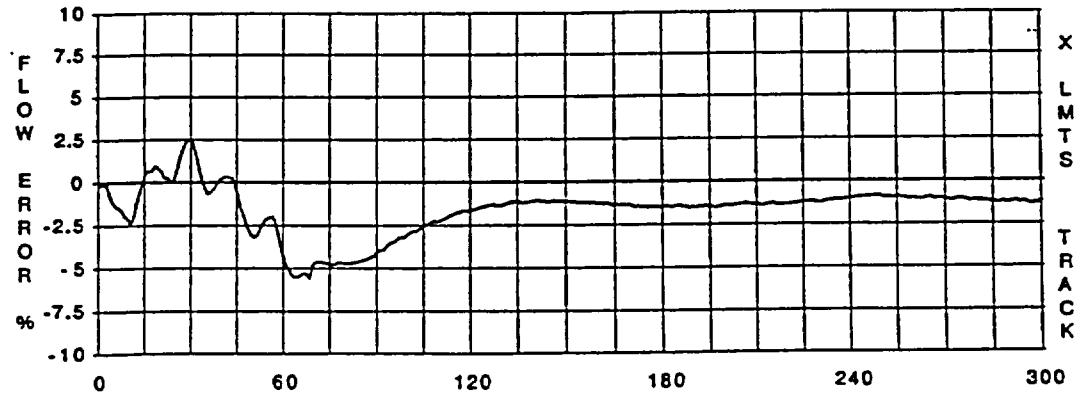

T AVG F
PZR LEVEL in
POINT # 4 - The decrease in T average
0
60
120
180
240
300


NR PRESS psig
WR PRESS psig
0
60
120
180
240
300


A RCS FLOW %
B RCS FLOW %
120
100
80
60
40
20
0
0
60
120
180
240
300

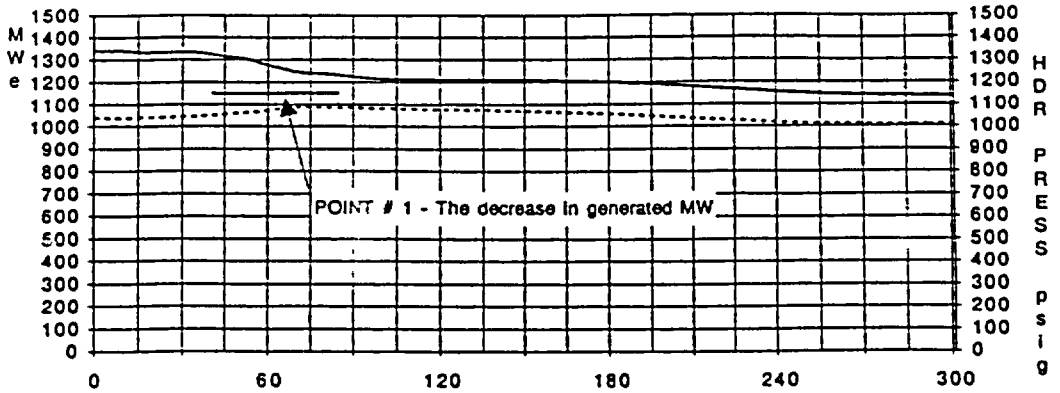

DELTA TCF
10
7.5
5
2.5
0
-2.5
-5
-7.5
-10
0
60
120
180
240
300

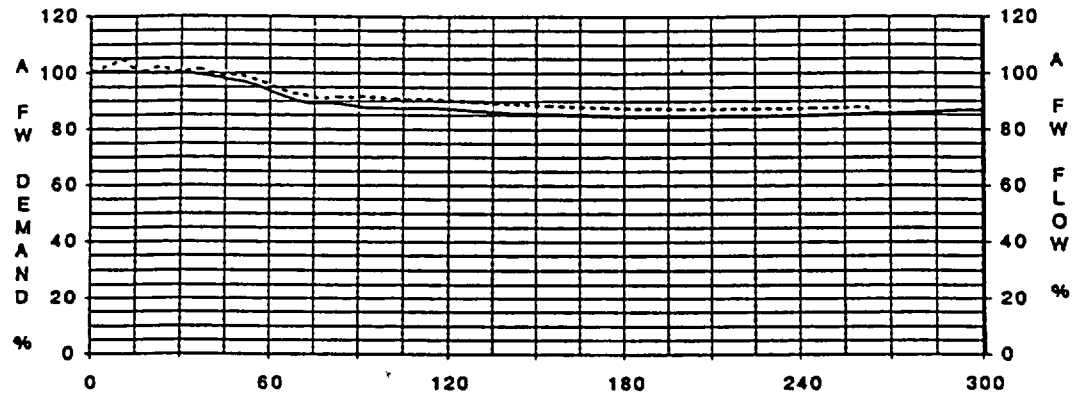

NEUTRON ERROR %
10
7.5
5
2.5
0
-2.5
-5
-7.5
-10
0
60
120
180
240
300

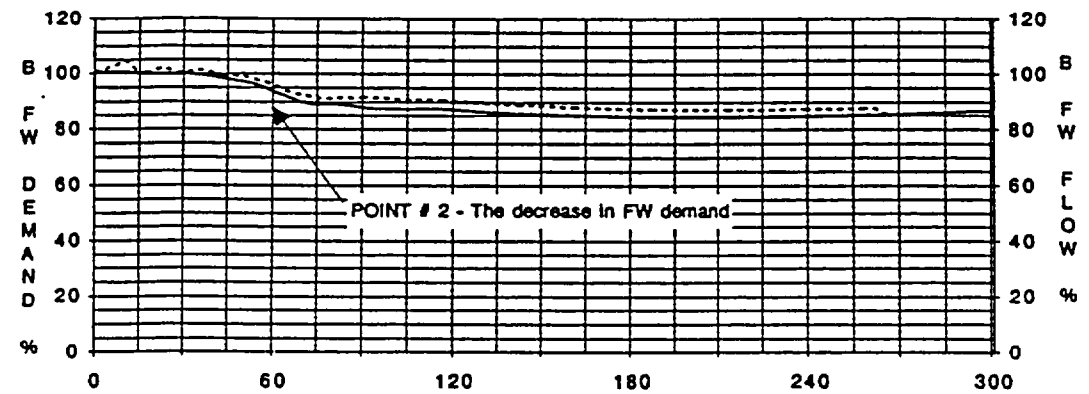

FLOW ERROR %
XLMTS TRACK
10
7.5
5
2.5
0
-2.5
-5
-7.5
-10
0
60
120
180
240
300

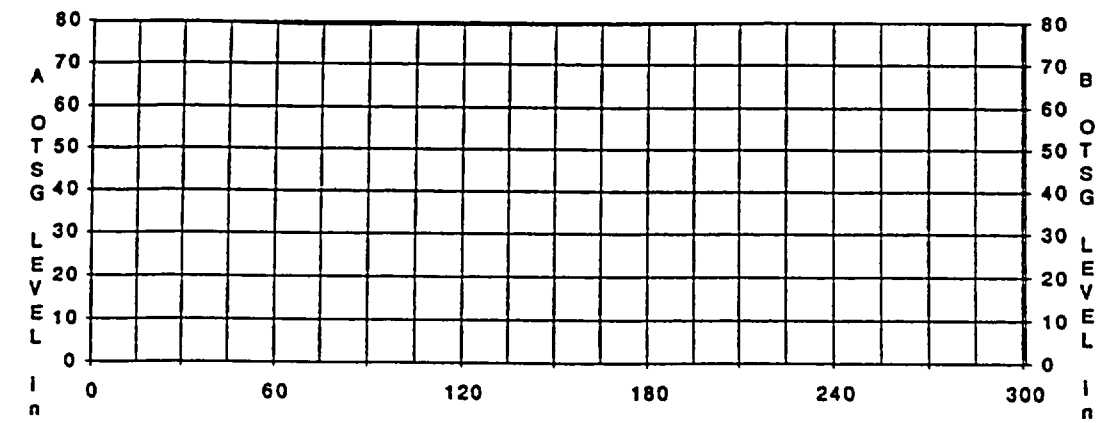
Transient 5

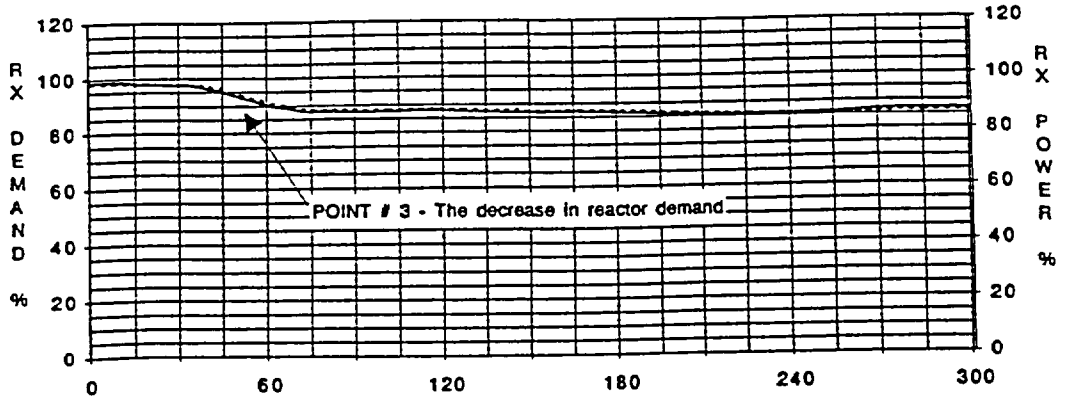
Power Decrease - Turbine in Manual

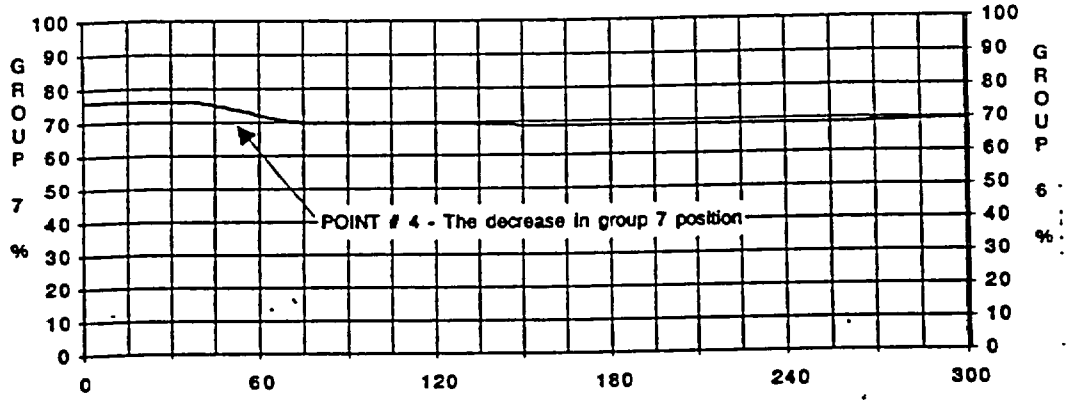

MWe
HDR PRESS psig
POINT # 1 - The decrease in generated MW
0
60
120
180
240
300

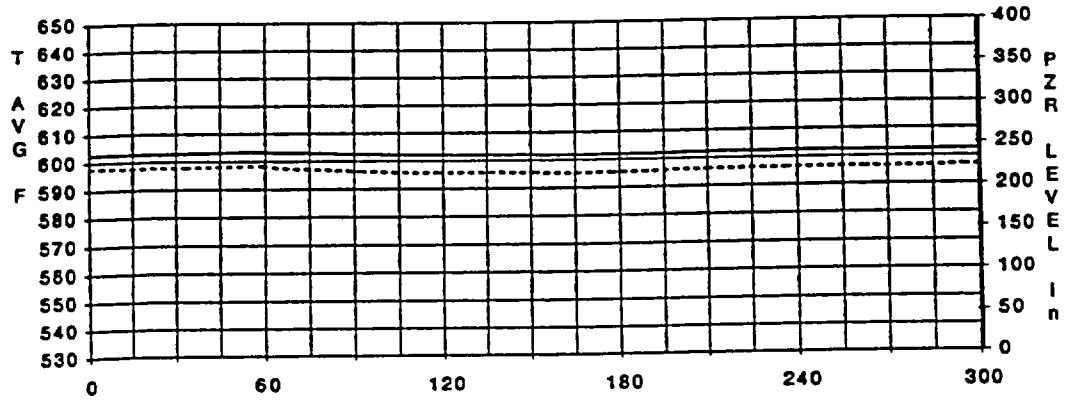

A FW DEMAND %
A FW FLOW %
0
60
120
180
240
300

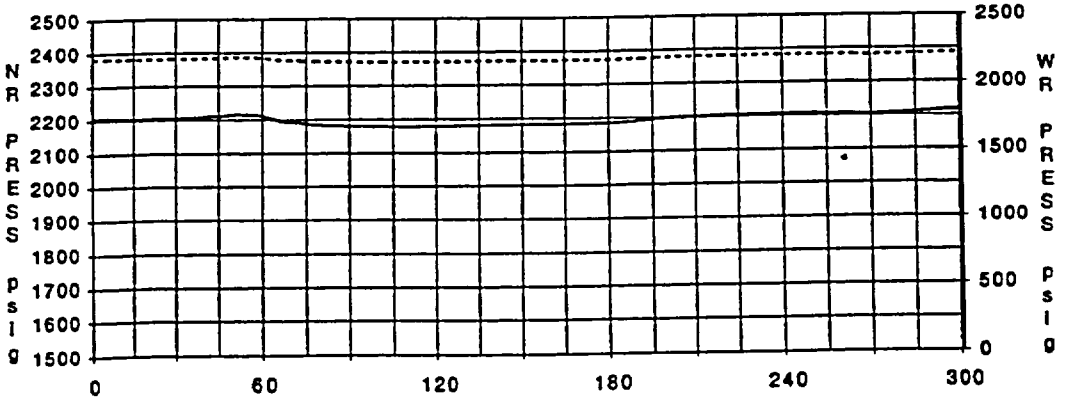

B FW DEMAND %
B FW FLOW %
POINT # 2 - The decrease in FW demand
0
60
120
180
240
300

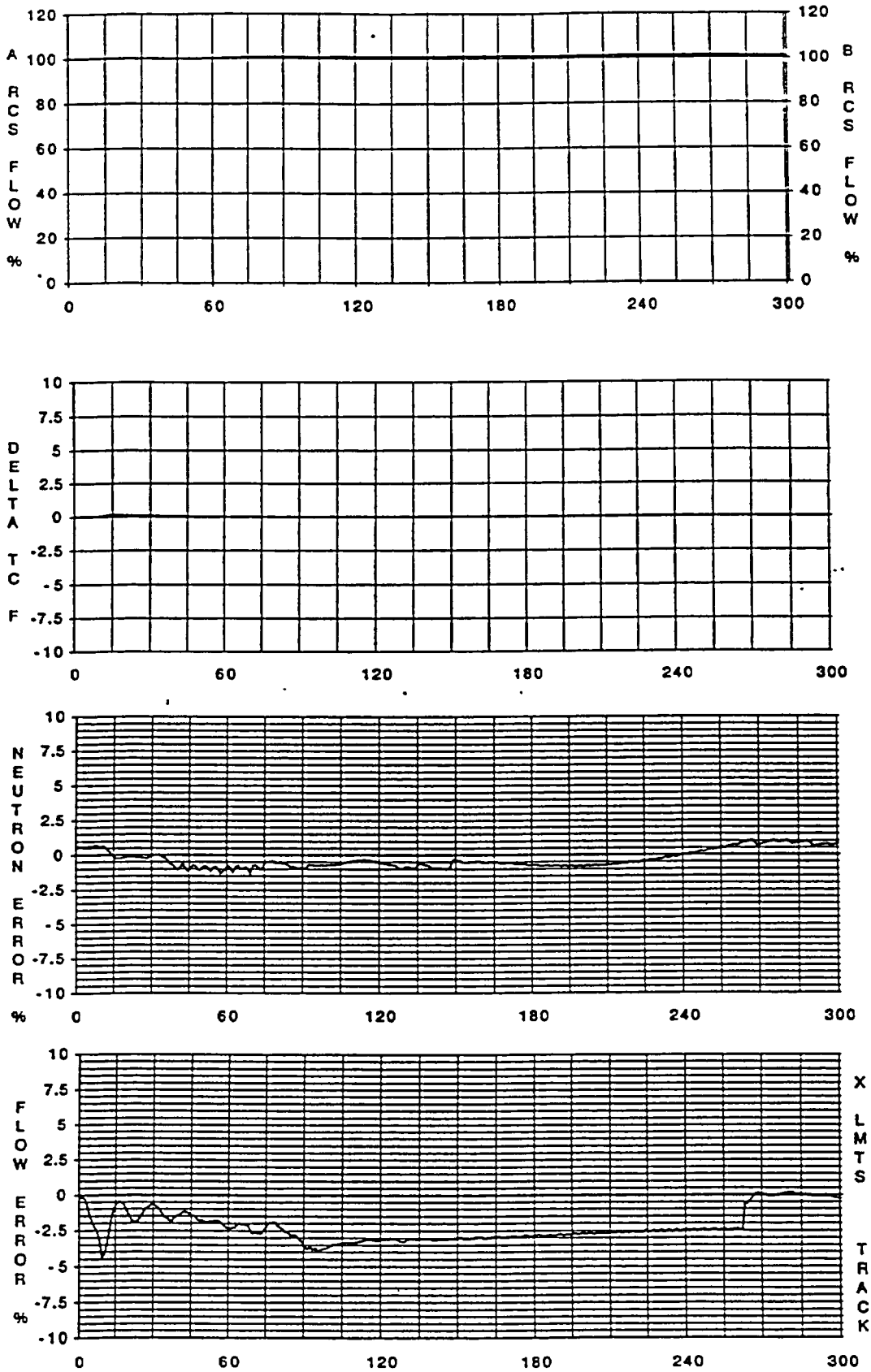

A OTSG LEVEL in
B OTSG LEVEL in
0
60
120
180
240
300


RX DEMAND %
RX POWER %
POINT # 3 - The decrease in reactor demand
0
60
120
180
240
300


GROUP 7 %
GROUP 6 %
POINT # 4 - The decrease in group 7 position
0
60
120
180
240
300

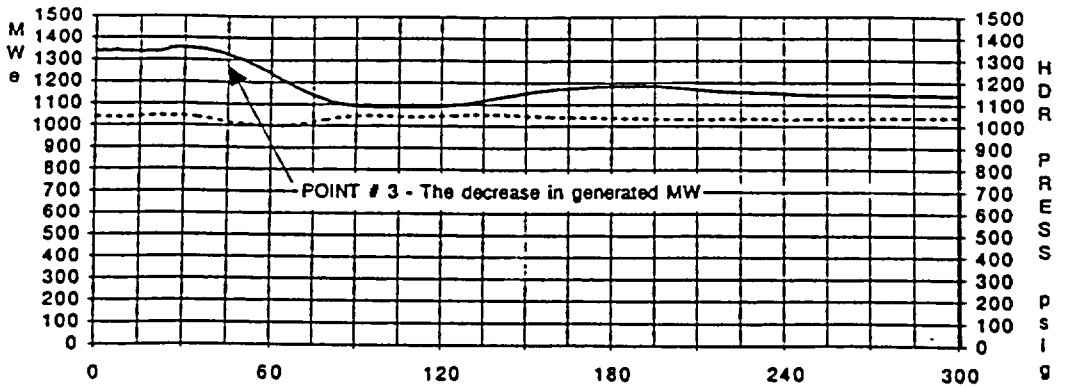

T AVG F
PZR LEVEL In
0
60
120
180
240
300

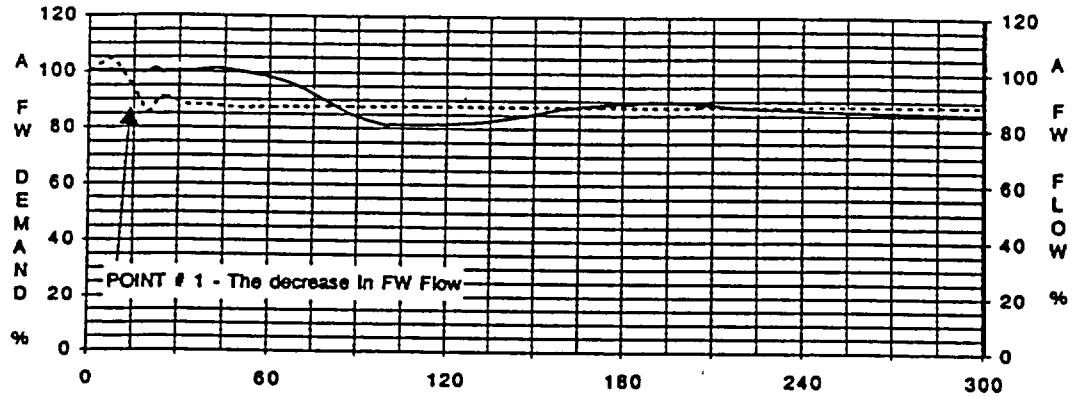

NR PRESS psig
WR PRESS psig
0
60
120
180
240
300

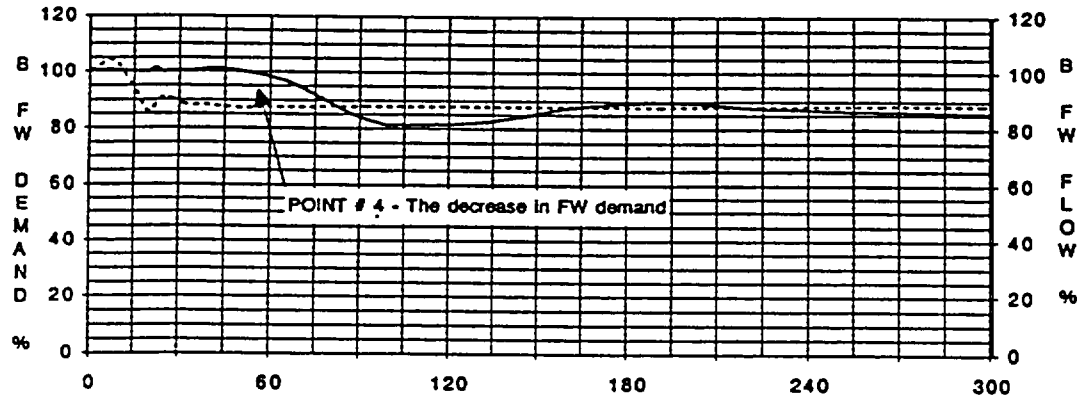

A RCS FLOW %
B RCS FLOW %
DELTA TCF
NEUTRON ERROR %
FLOW ERROR %
X LMTS
TRACK
120
100
80
60
40
20
0
10
7.5
5
2.5
-2.5
-5
-7.5
-10
0
60
120
180
240
300

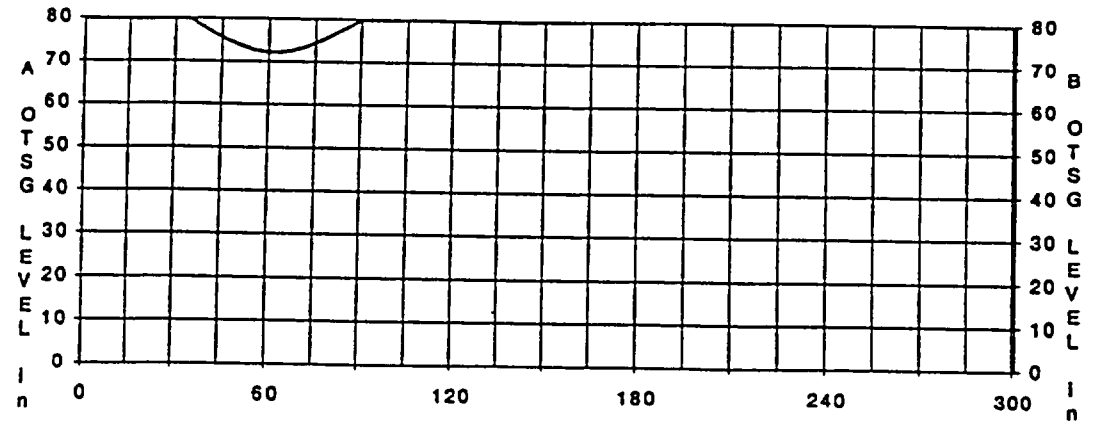
Transient 6

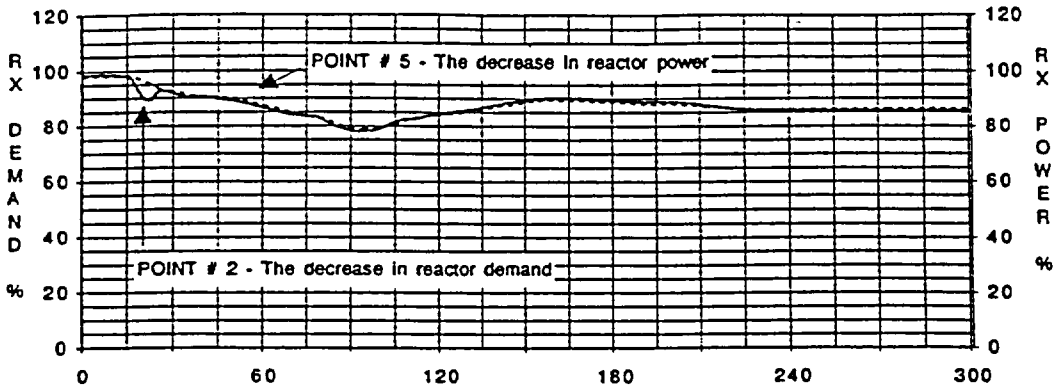
Power Decrease - FW in Manual

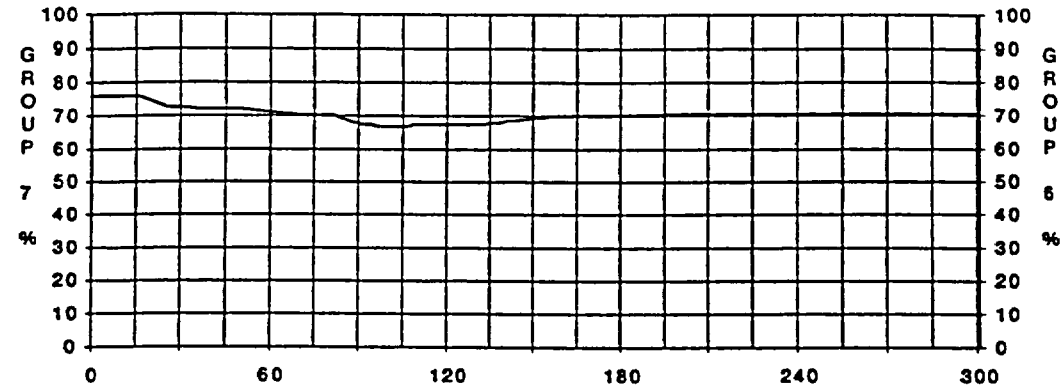

MWe
HDR PRESS psig
POINT # 3 - The decrease in generated MW
0 60 120 180 240 300

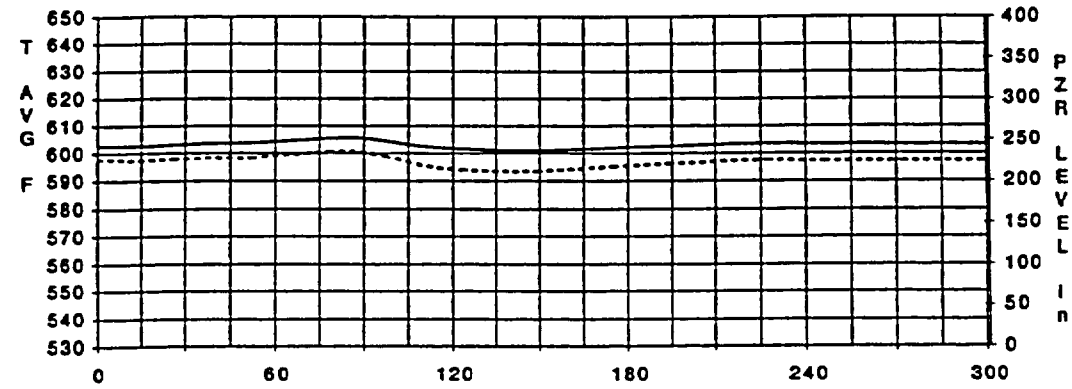

A FW DEMAND %
A FW FLOW %
POINT # 1 - The decrease in FW Flow
0 60 120 180 240 300

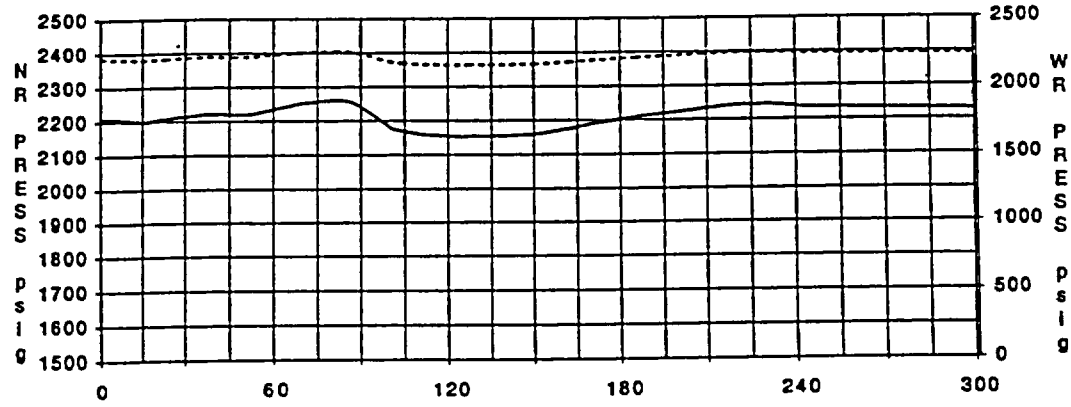

B FW DEMAND %
B FW FLOW %
POINT # 4 - The decrease in FW demand
0 60 120 180 240 300

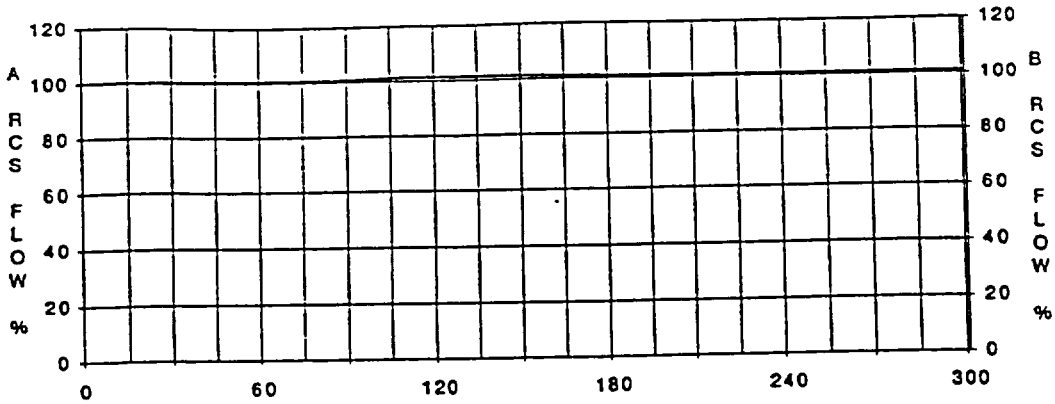

A OTSG LEVEL in
B OTSG LEVEL in
0 60 120 180 240 300

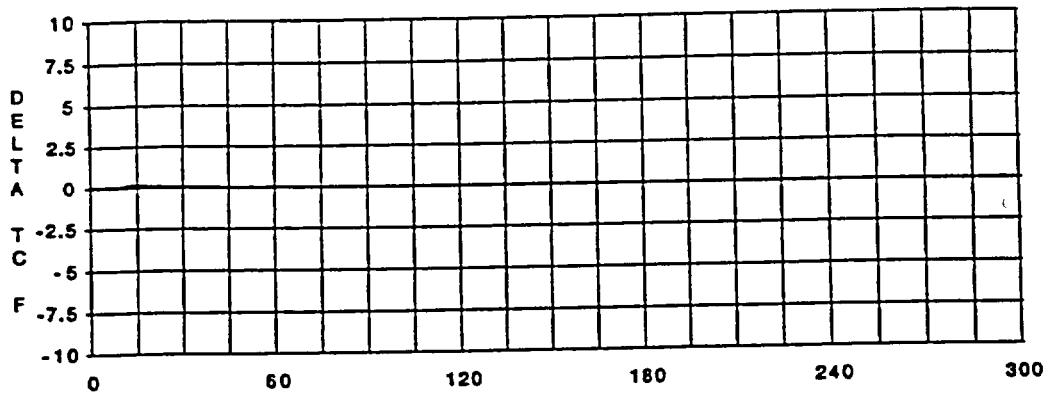

POINT # 5 - The decrease in reactor power
POINT # 2 - The decrease in reactor demand
RX DEMAND %
RX POWER %
0
20
40
60
80
100
120
0
60
120
180
240
300

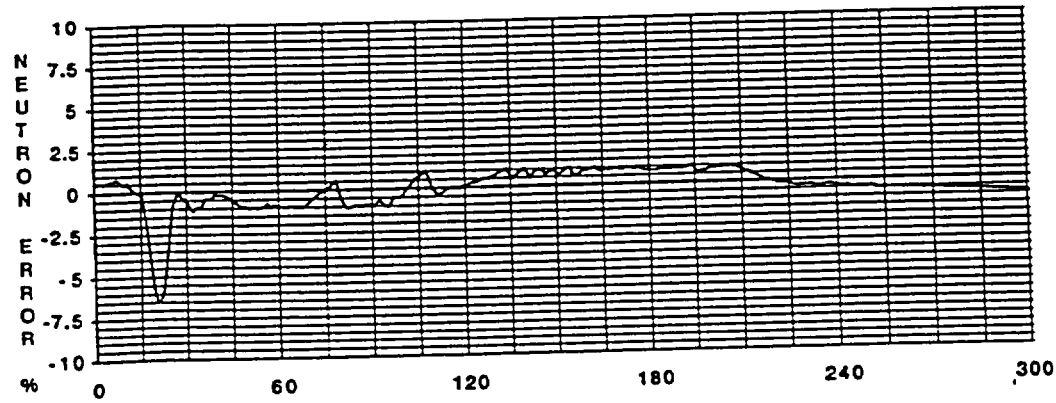

GROUP 7 %
GROUP 6 %
0
10
20
30
40
50
60
70
80
90
100
0
60
120
180
240
300

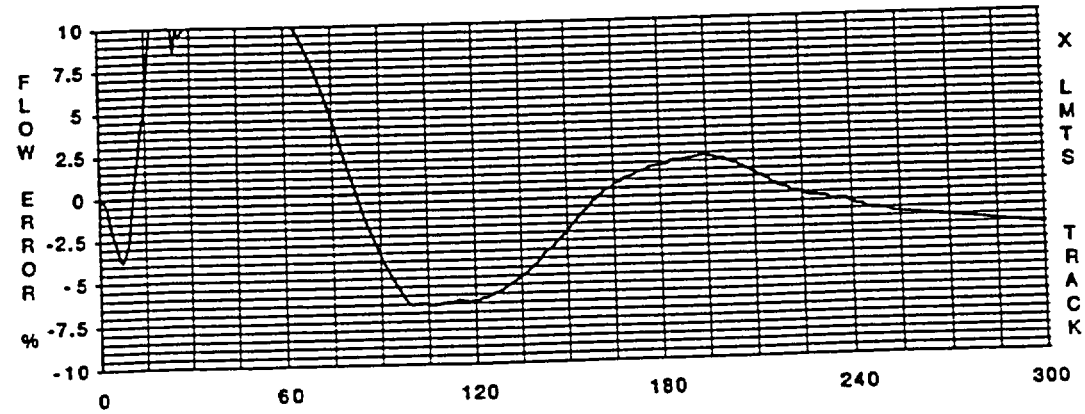

T AVG F
PZR LEVEL in
530
540
550
560
570
580
590
600
610
620
630
640
650
0
50
100
150
200
250
300
350
400
0
60
120
180
240
300


NR PRESS psig
WR PRESS psig
1500
1600
1700
1800
1900
2000
2100
2200
2300
2400
2500
0
500
1000
1500
2000
2500
0
60
120
180
240
300


A RCS FLOW %
B RCS FLOW %
120
100
80
60
40
20
0
0
60
120
180
240
300

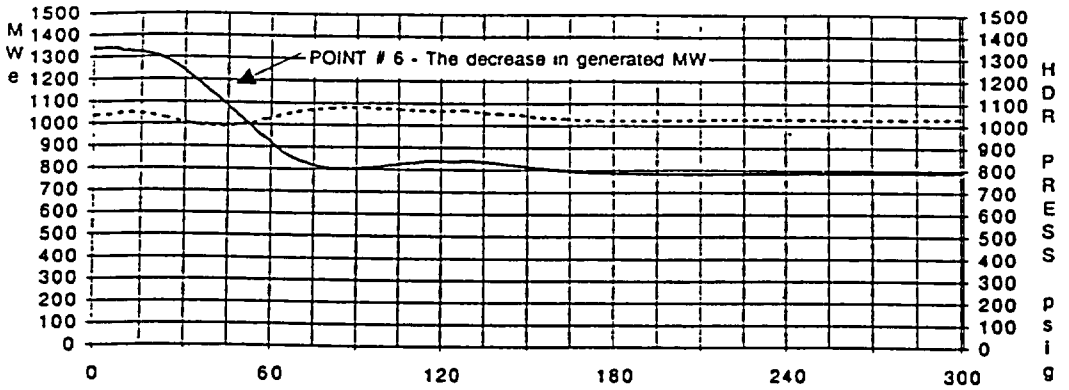

DELTA TCF
10
7.5
5
2.5
0
-2.5
-5
-7.5
-10
0
60
120
180
240
300

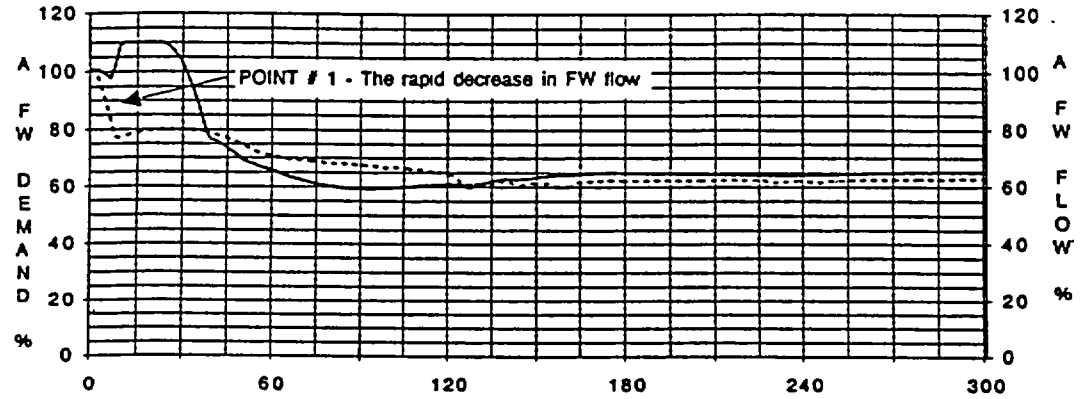

NEUTRON ERROR %
10
7.5
5
2.5
0
-2.5
-5
-7.5
-10
0
60
120
180
240
300

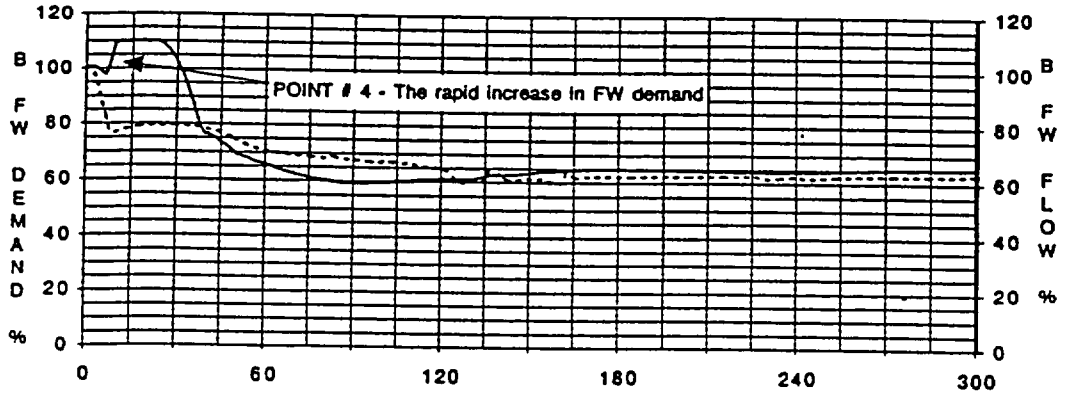

FLOW ERROR %
X LMTS
TRACK
10
7.5
5
2.5
0
-2.5
-5
-7.5
-10
0
60
120
180
240
300

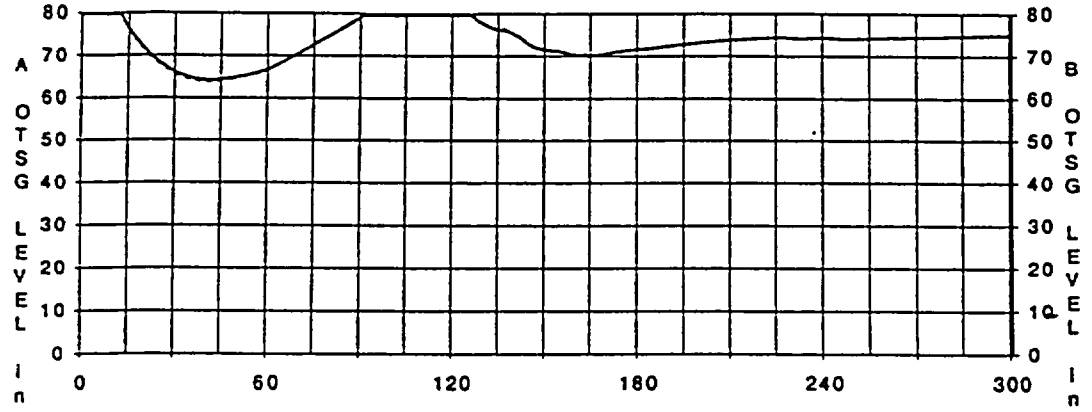
Transient 7

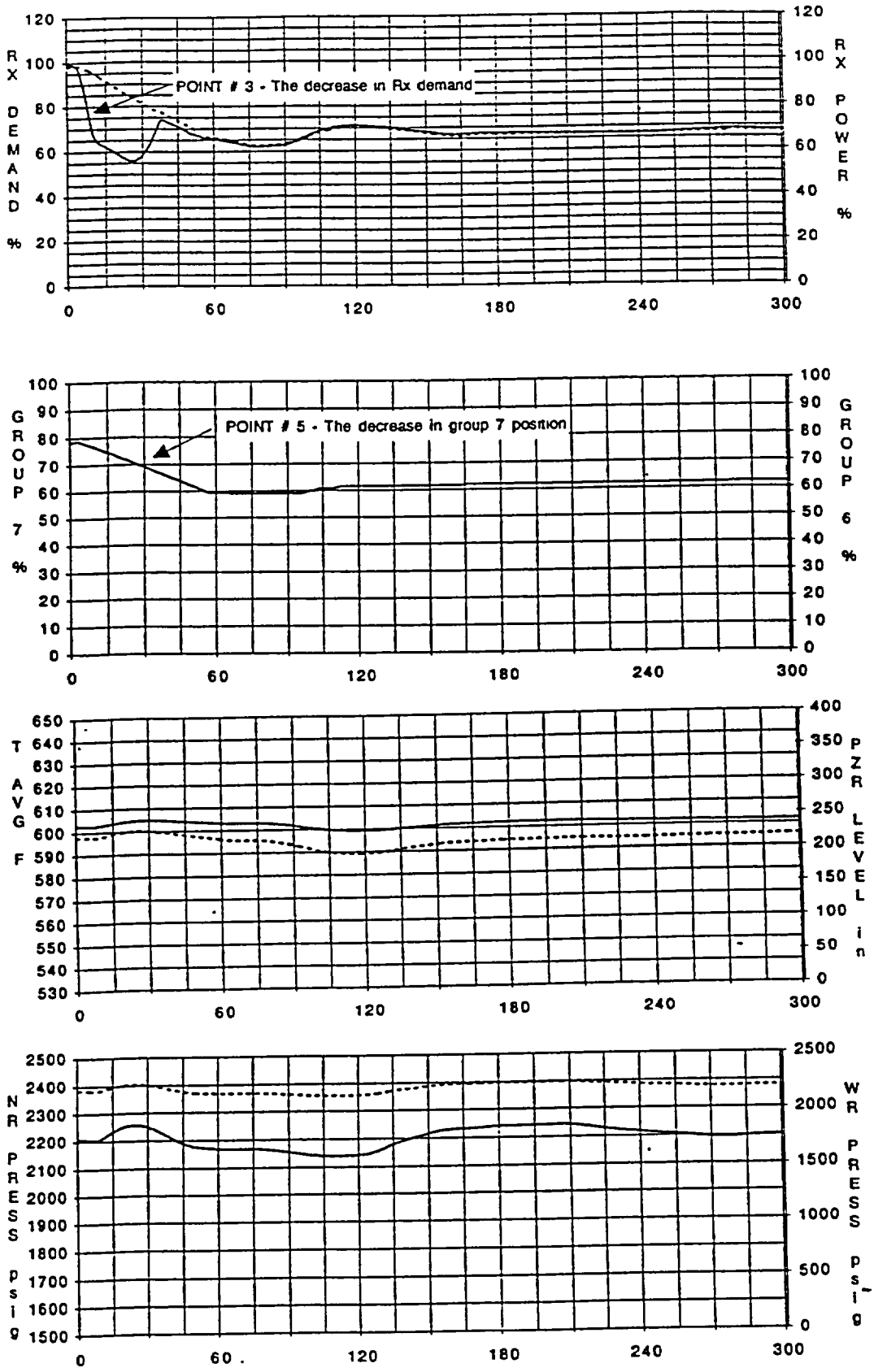
Main Feed Pump Trip

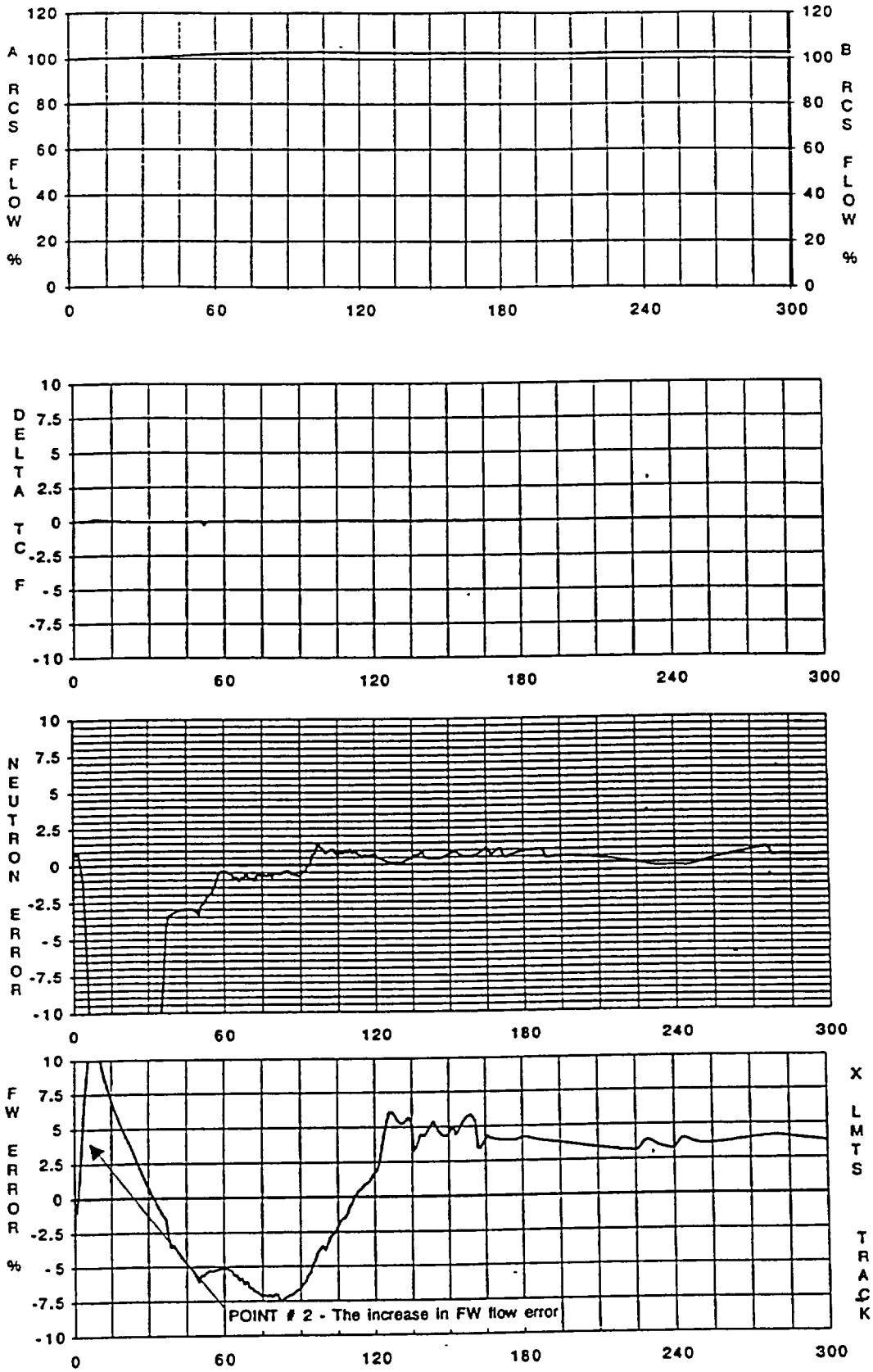

MWe
HDR PRESS psig
POINT # 6 - The decrease in generated MW
1500
1400
1300
1200
1100
1000
900
800
700
600
500
400
300
200
100
0
0
60
120
180
240
300


A FW DEMAND %
A FW FLOW %
POINT # 1 - The rapid decrease in FW flow
120
100
80
60
40
20
0
0
60
120
180
240
300


B FW DEMAND %
B FW FLOW %
POINT # 4 - The rapid increase in FW demand
120
100
80
60
40
20
0
0
60
120
180
240
300

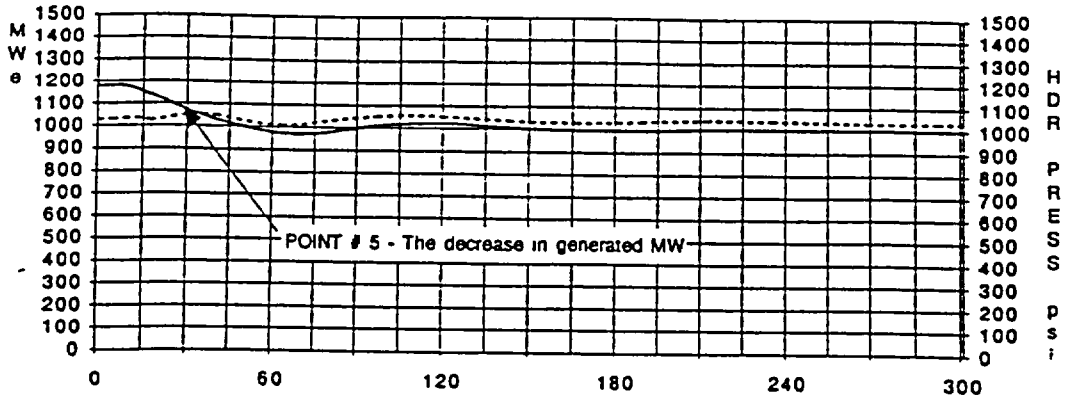

A OTSG LEVEL in
B OTSG LEVEL in
80
70
60
50
40
30
20
10
0
0
60
120
180
240
300

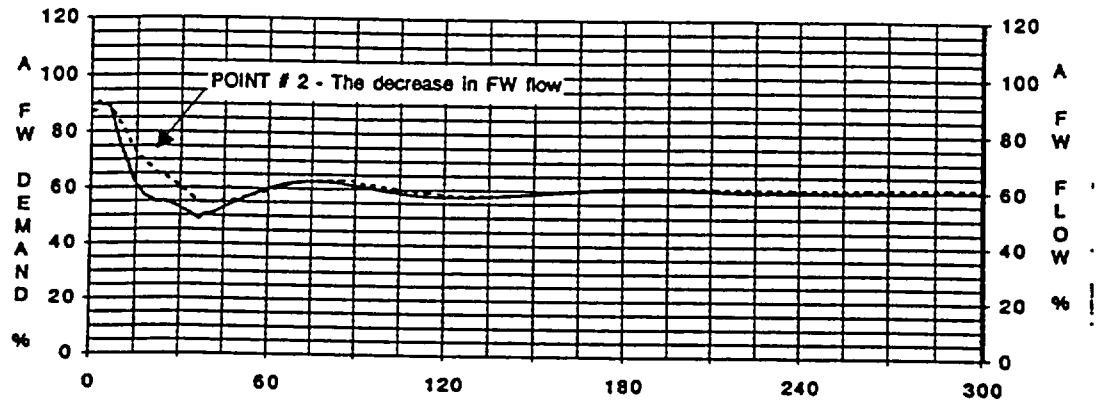

RX DEMAND %
RX POWER %
POINT # 3 - The decrease in Rx demand
GROUP 7 %
GROUP 6 %
POINT # 5 - The decrease in group 7 position
T AVG F
PZR LEVEL in
NR PRESS psig
WR PRESS psig
0
60
120
180
240
300

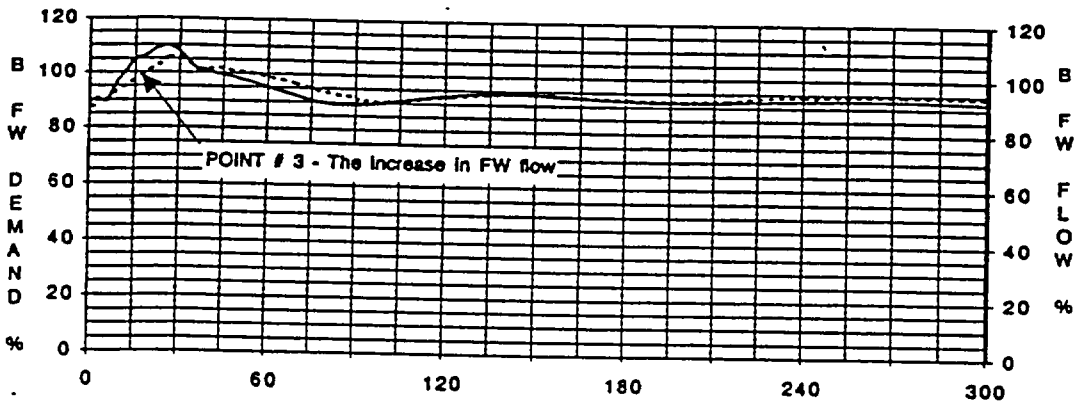

A RCS FLOW %
B RCS FLOW %
DELTA TCF
NEUTRON ERROR
FW ERROR %
X LMTS TRACK
POINT # 2 - The increase in FW flow error
0
60
120
180
240
300

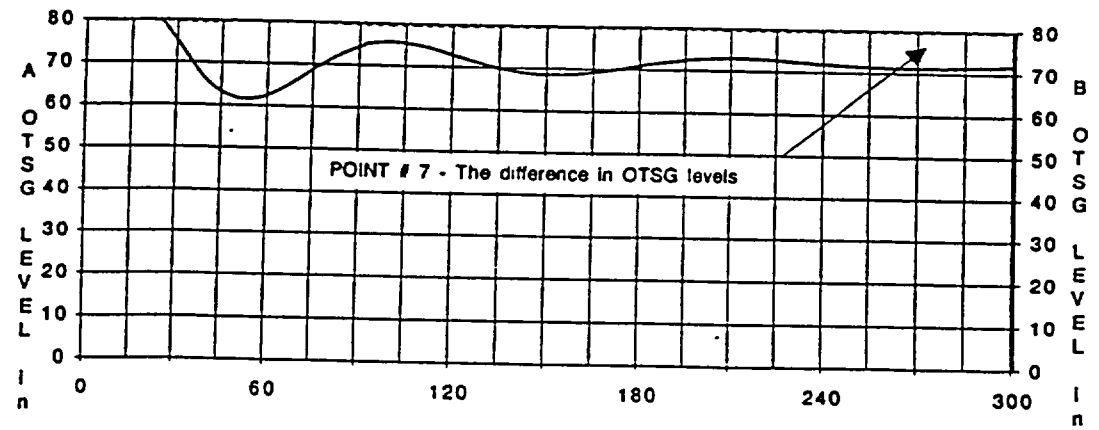
Transient 8

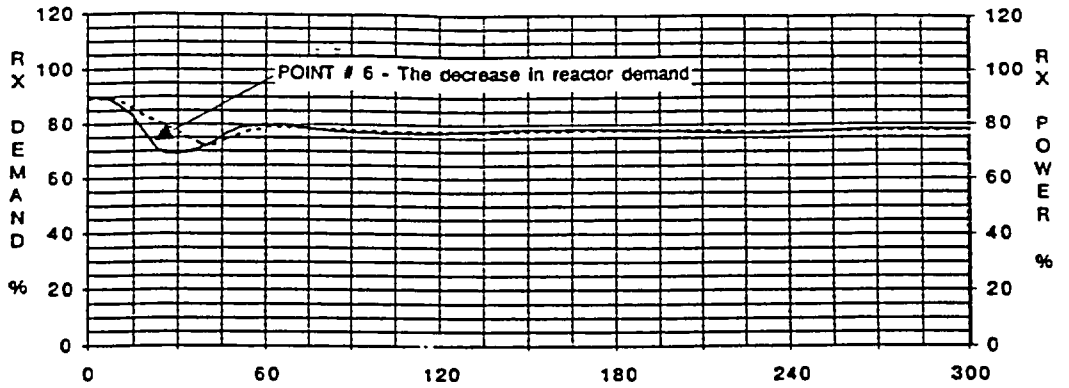
RCP Trip

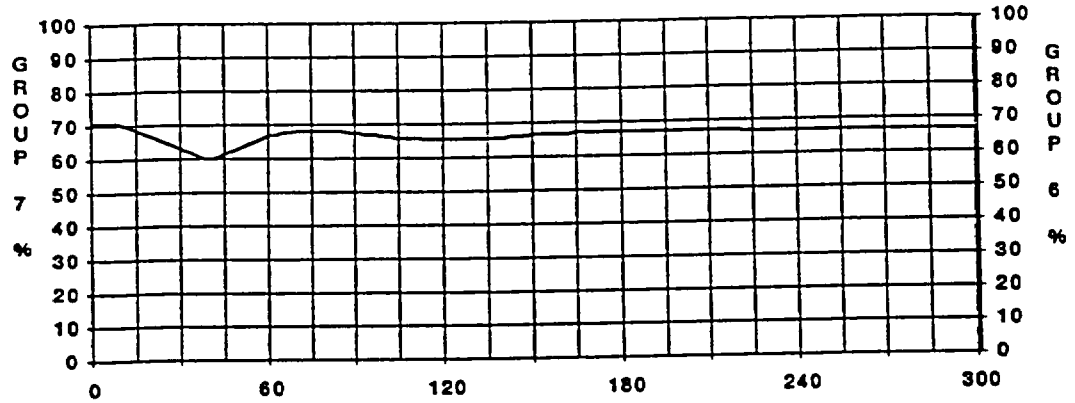

MWe
HDR PRESS psi
POINT # 5 - The decrease in generated MW
0
60
120
180
240
300

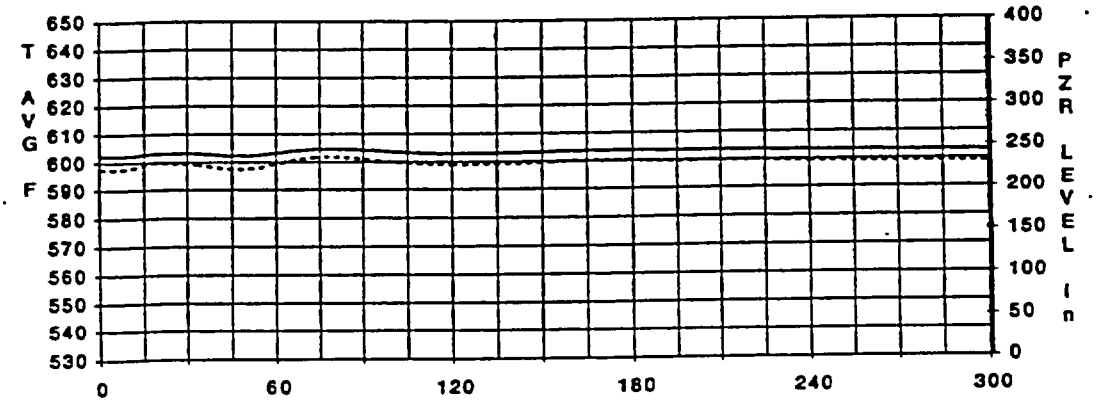

A FW DEMAND %
A FW FLOW %
POINT # 2 - The decrease in FW flow
0
60
120
180
240
300

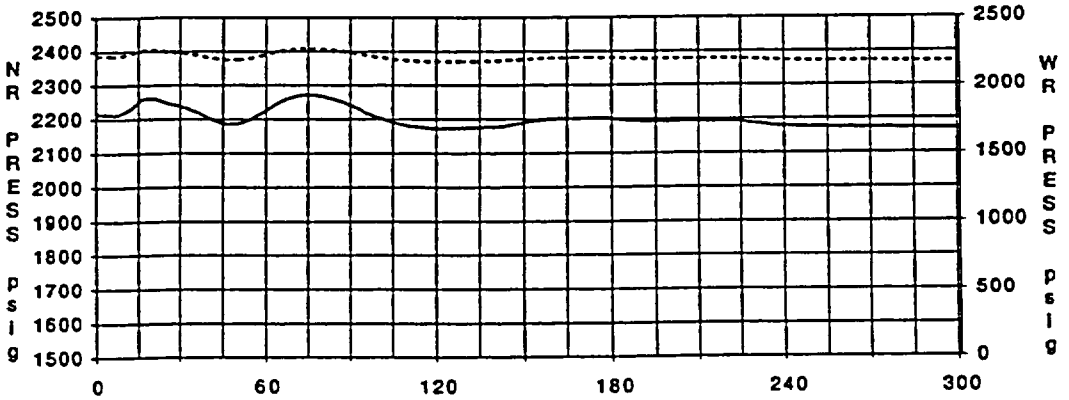

B FW DEMAND %
B FW FLOW %
POINT # 3 - The increase in FW flow
0
60
120
180
240
300

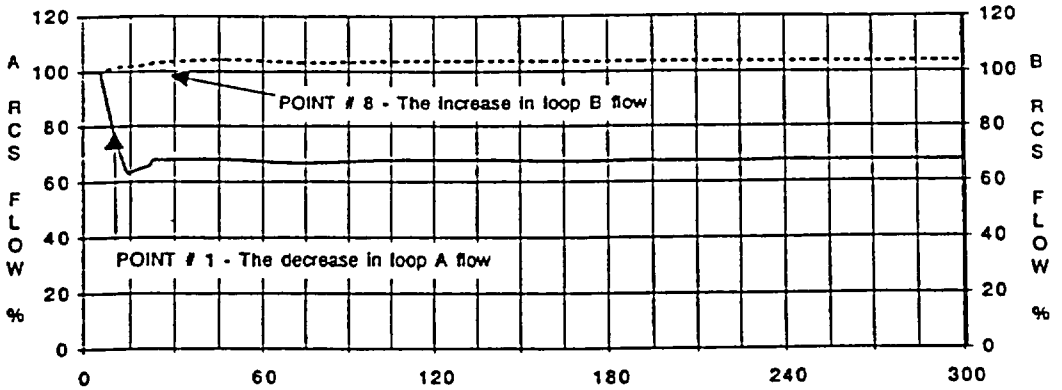

A OTSG LEVEL in
B OTSG LEVEL in
POINT # 7 - The difference in OTSG levels
0
60
120
180
240
300

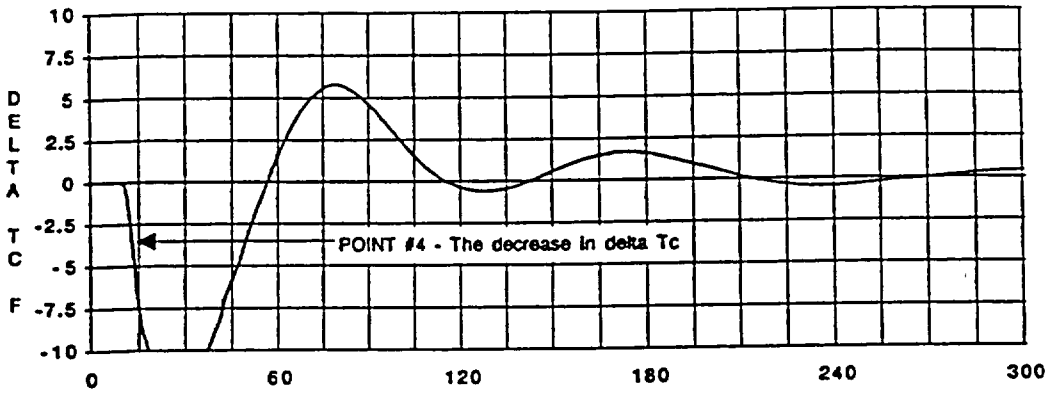

POINT # 6 - The decrease in reactor demand
RX DEMAND %
RX POWER %
0
20
40
60
80
100
120
0
60
120
180
240
300

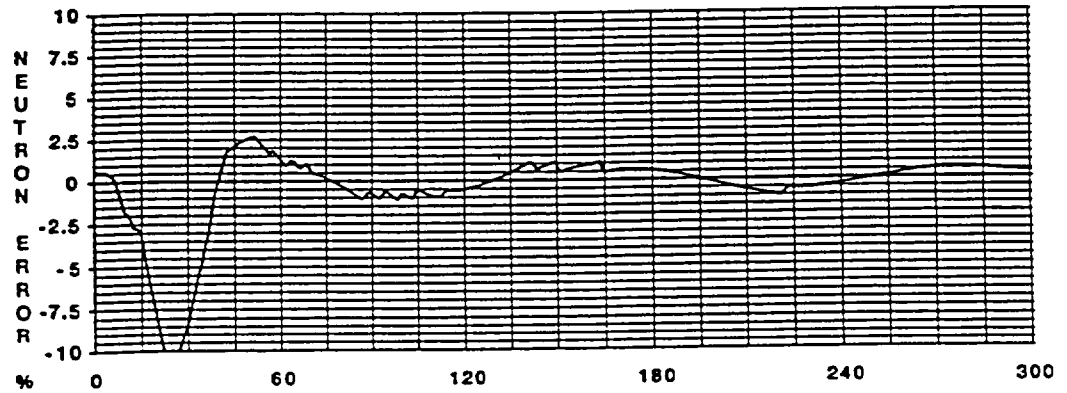

GROUP 7 %
GROUP 6 %
0
10
20
30
40
50
60
70
80
90
100
0
60
120
180
240
300

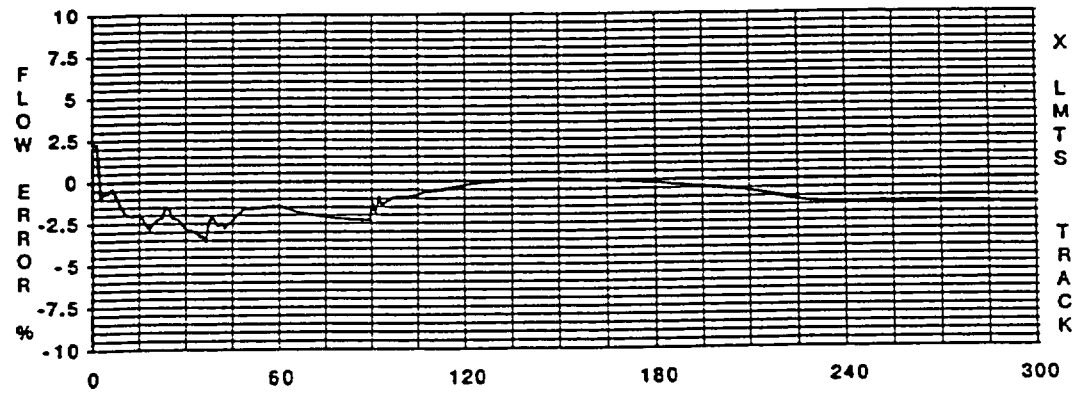

T AVG F
PZR LEVEL in
530
540
550
560
570
580
590
600
610
620
630
640
650
0
50
100
150
200
250
300
350
400
0
60
120
180
240
300


NR PRESS psig
WR PRESS psig
1500
1600
1700
1800
1900
2000
2100
2200
2300
2400
2500
0
500
1000
1500
2000
2500
0
60
120
180
240
300


A RCS FLOW %
B RCS FLOW %
POINT # 8 - The increase in loop B flow
POINT # 1 - The decrease in loop A flow
120
100
80
60
40
20
0
0
60
120
180
240
300

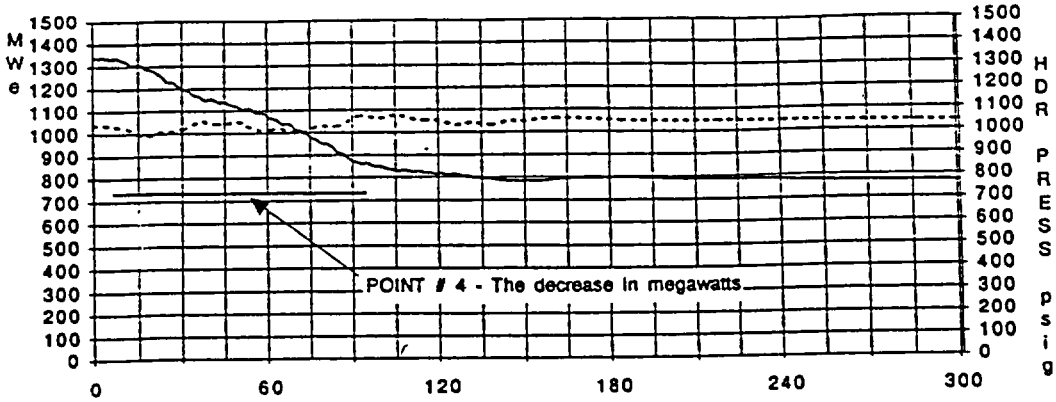

DELTA TCF
POINT #4 - The decrease in delta Tc
10
7.5
5
2.5
0
-2.5
-5
-7.5
-10
0
60
120
180
240
300

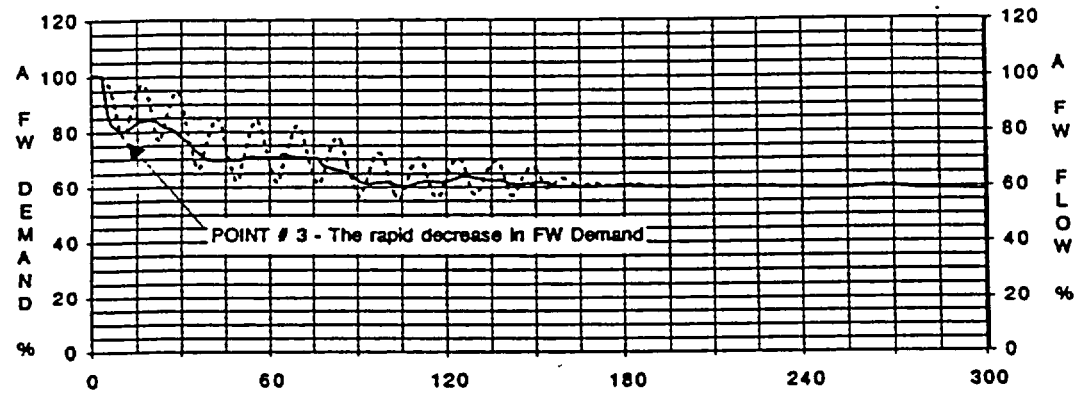

NEUTRON ERROR %
10
7.5
5
2.5
0
-2.5
-5
-7.5
-10
0
60
120
180
240
300

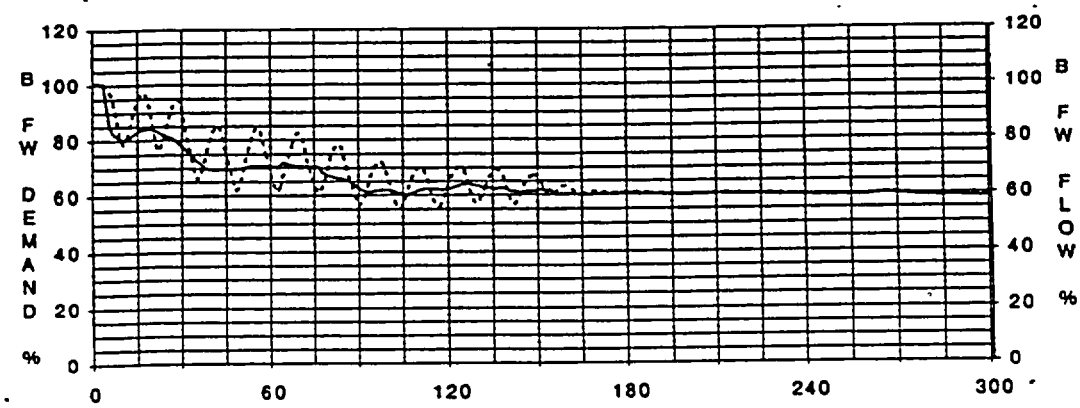

FLOW ERROR %
X LMTS TRACK
10
7.5
5
2.5
0
-2.5
-5
-7.5
-10
0
60
120
180
240
300

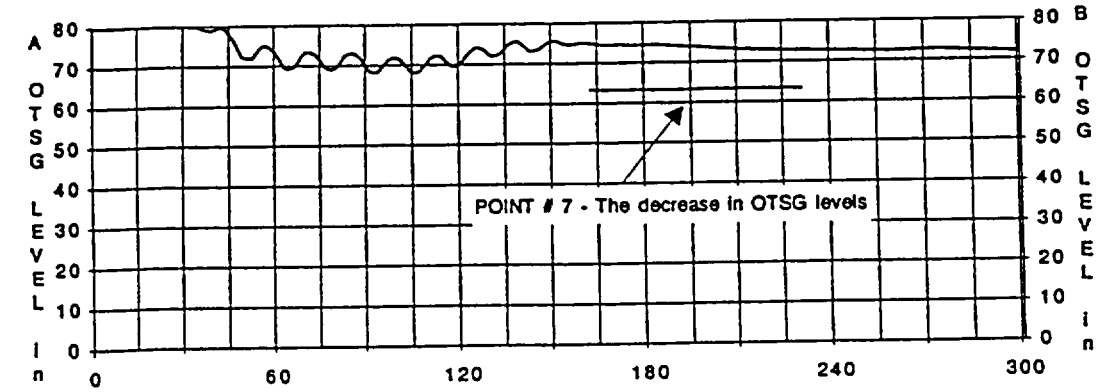
Transient 9

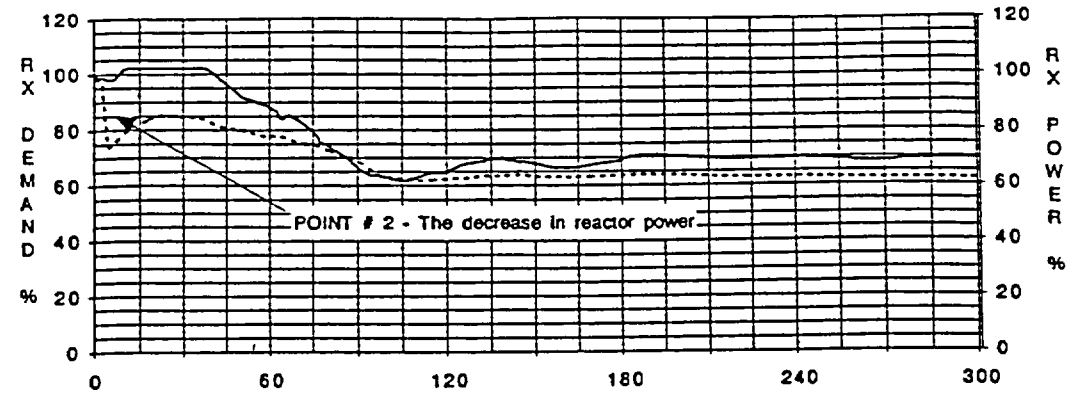
Dropped Rod

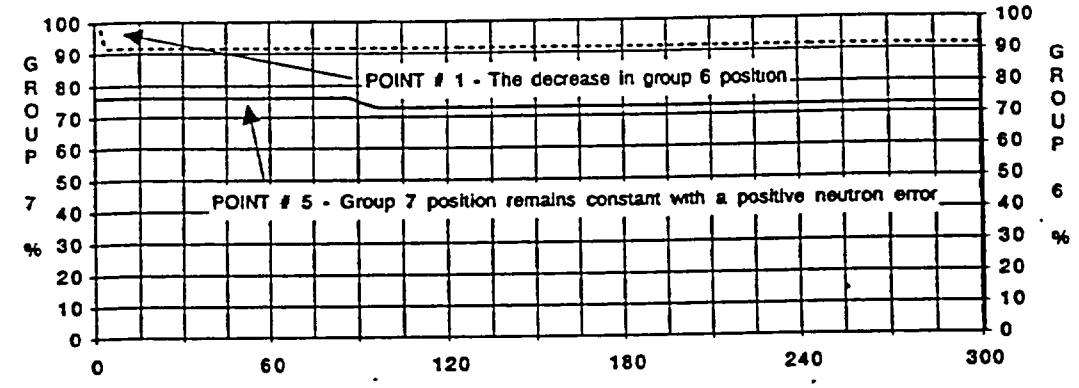

MWe
HDR PRESS psig
POINT # 4 - The decrease in megawatts

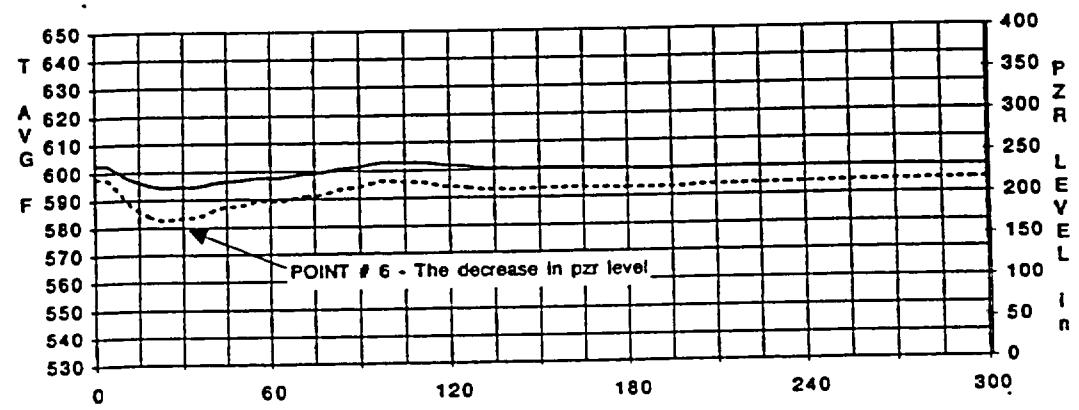

A FW DEMAND %
A FW FLOW %
POINT # 3 - The rapid decrease in FW Demand

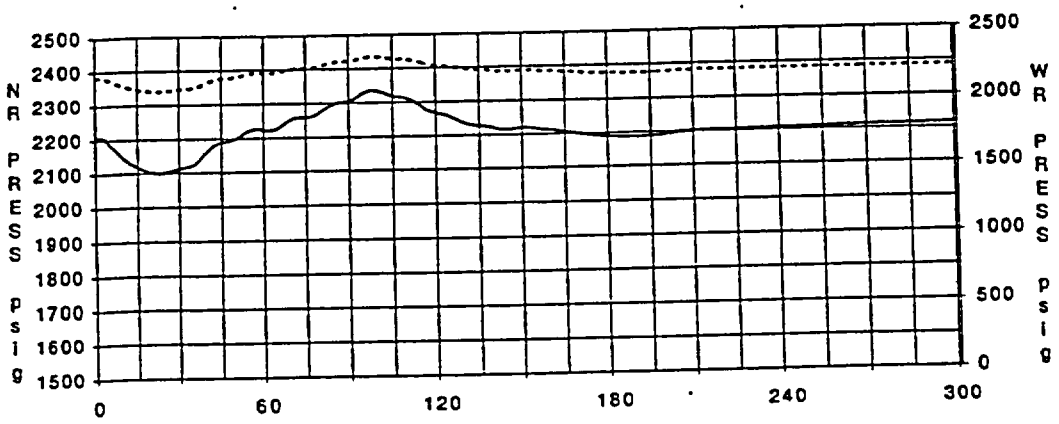

B FW DEMAND %
B FW FLOW %

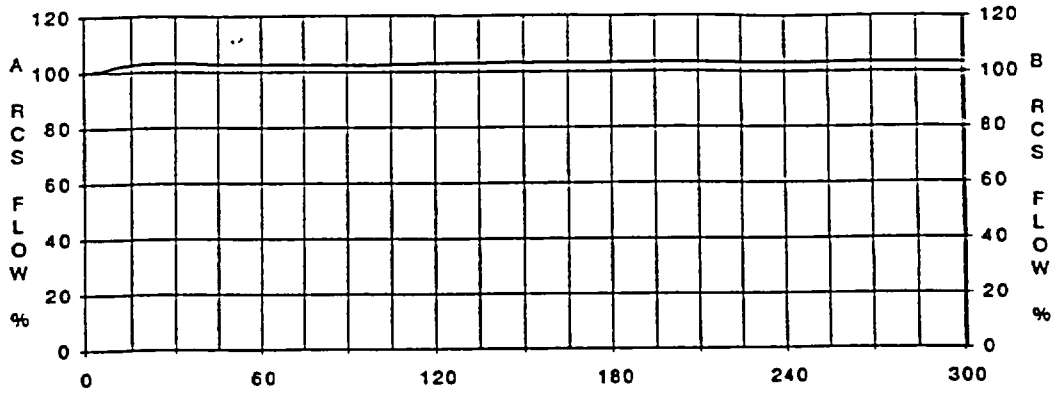

A OTSG LEVEL in
B OTSG LEVEL in
POINT # 7 - The decrease in OTSG levels

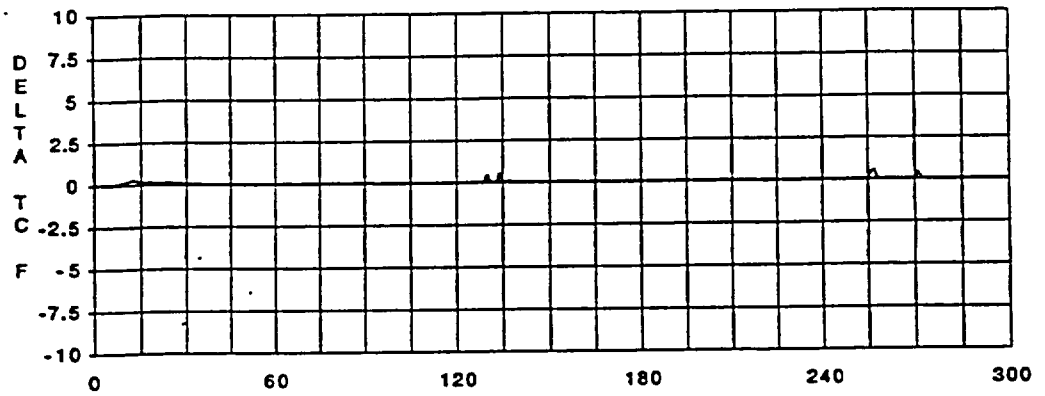

RX DEMAND %
RX POWER %
POINT # 2 - The decrease in reactor power
0
60
120
180
240
300

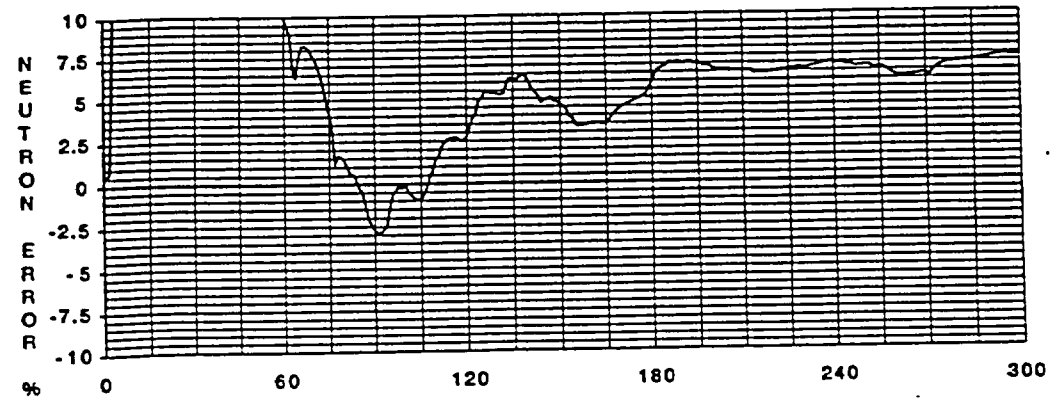

GROUP 7 %
GROUP 6 %
POINT # 1 - The decrease in group 6 position
POINT # 5 - Group 7 position remains constant with a positive neutron error
0
60
120
180
240
300

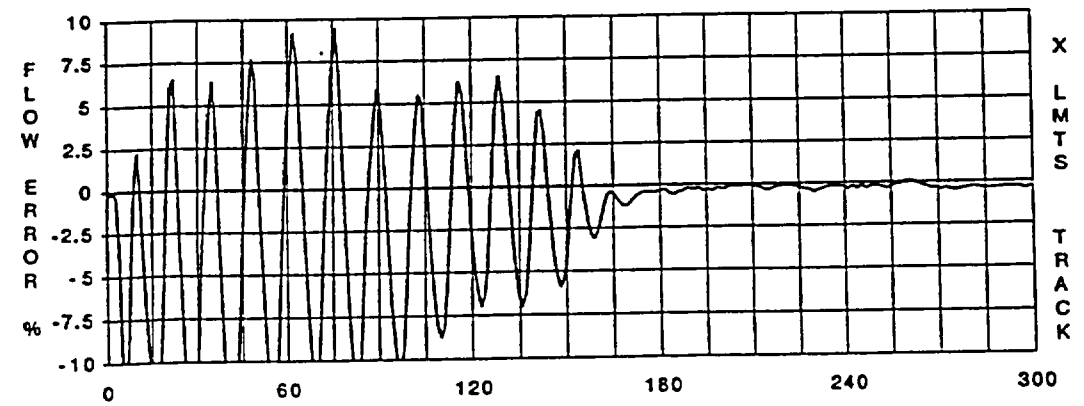

T AVG F
PZR LEVEL in
POINT # 6 - The decrease in pzr level
0
60
120
180
240
300


NR PRESS psig
WR PRESS psig
0
60
120
180
240
300


A RCS FLOW %
B RCS FLOW %
120
100
80
60
40
20
0
0
60
120
180
240
300

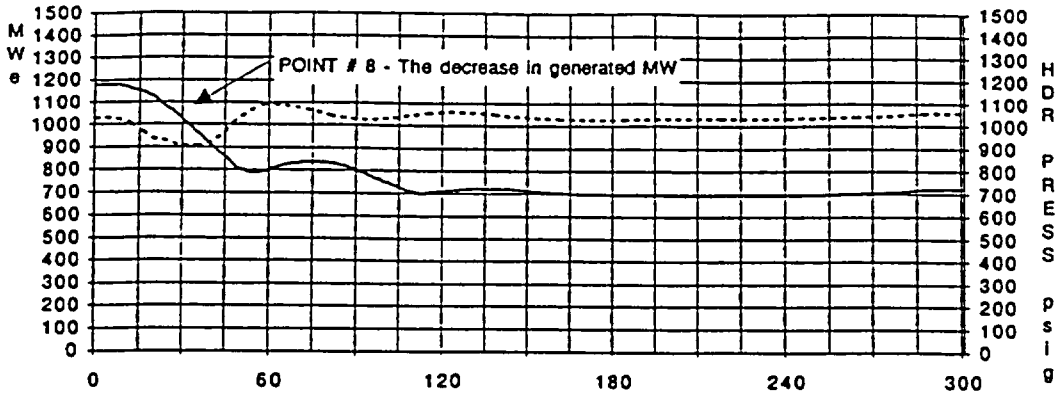

DELTA TCF
10
7.5
5
2.5
0
-2.5
-5
-7.5
-10
0
60
120
180
240
300

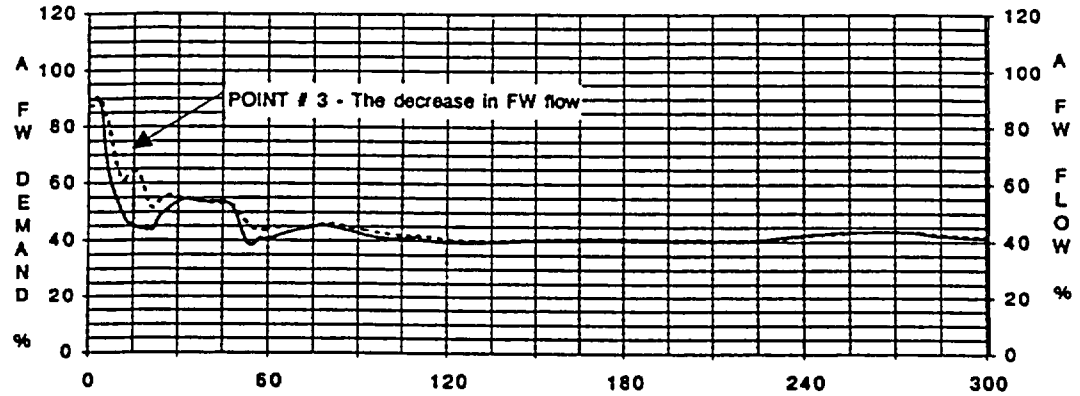

NEUTRON ERROR %
10
7.5
5
2.5
0
-2.5
-5
-7.5
-10
0
60
120
180
240
300

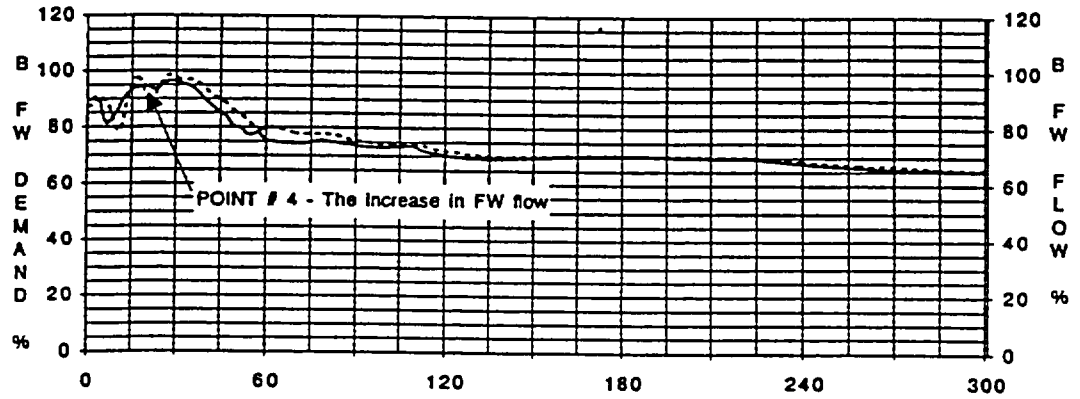

FLOW ERROR %
XLMTS TRACK
10
7.5
5
2.5
0
-2.5
-5
-7.5
-10
0
60
120
180
240
300

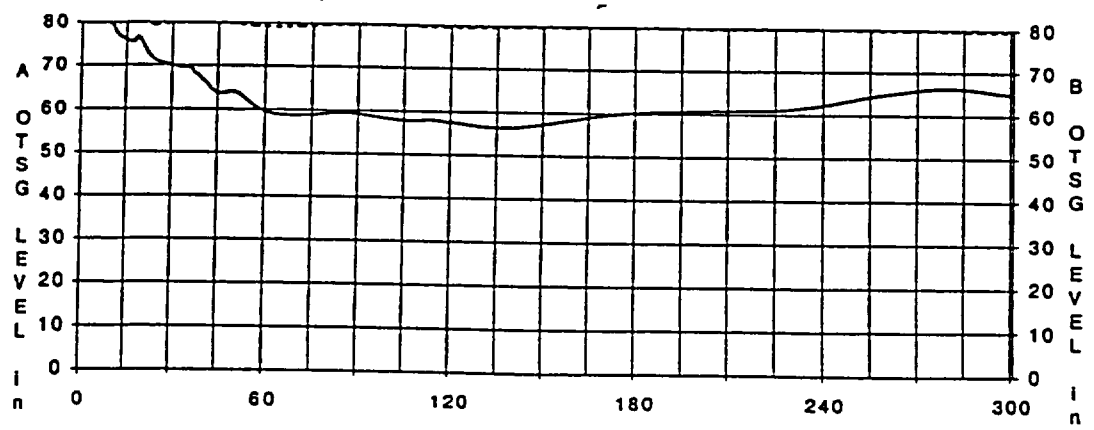
Transient 10

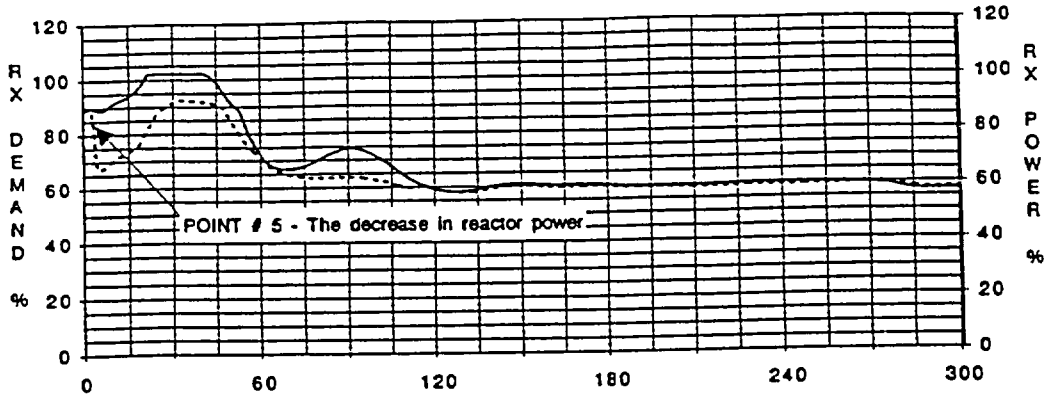
RCP Trip and Dropped Rod

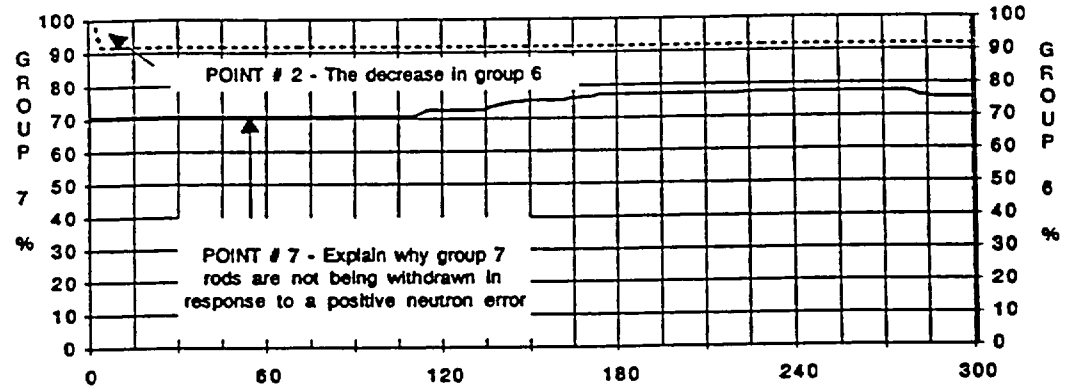

MWe
HDR PRESS psig
POINT # 8 - The decrease in generated MW
0
60
120
180
240
300

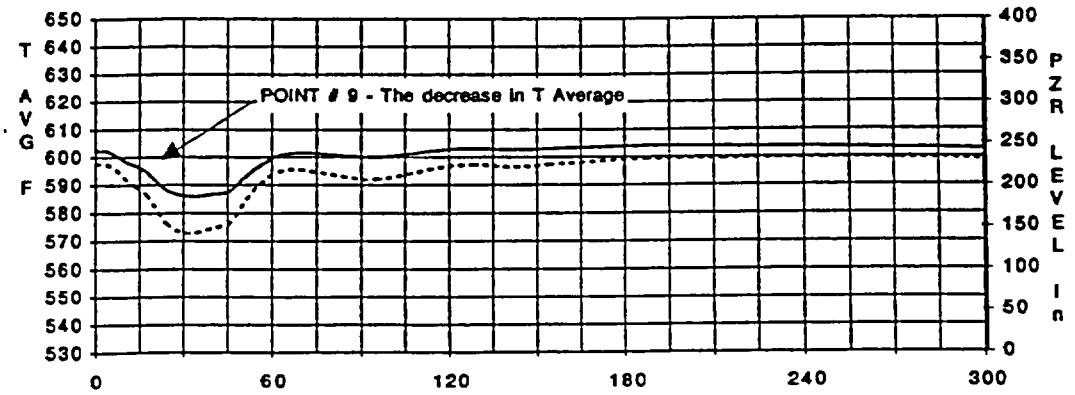

AFW DEMAND %
AFW FLOW %
POINT # 3 - The decrease in FW flow
0
60
120
180
240
300

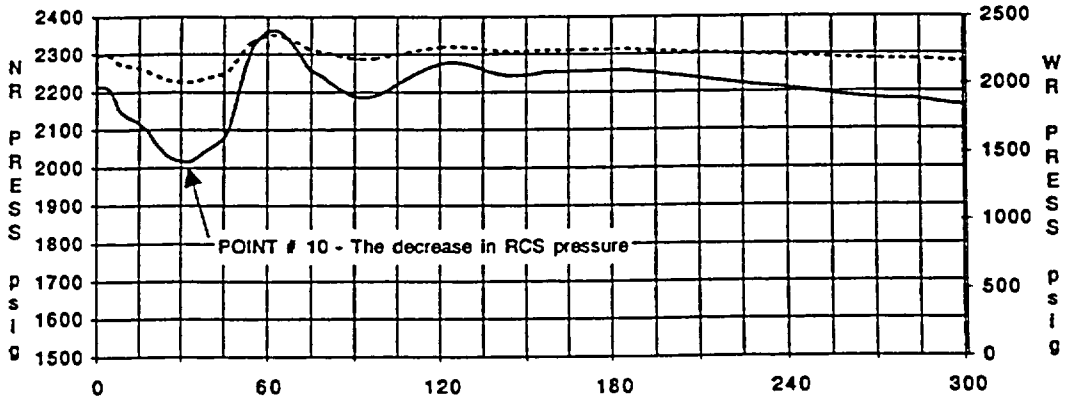

BFW DEMAND %
BFW FLOW %
POINT # 4 - The increase in FW flow
0
60
120
180
240
300

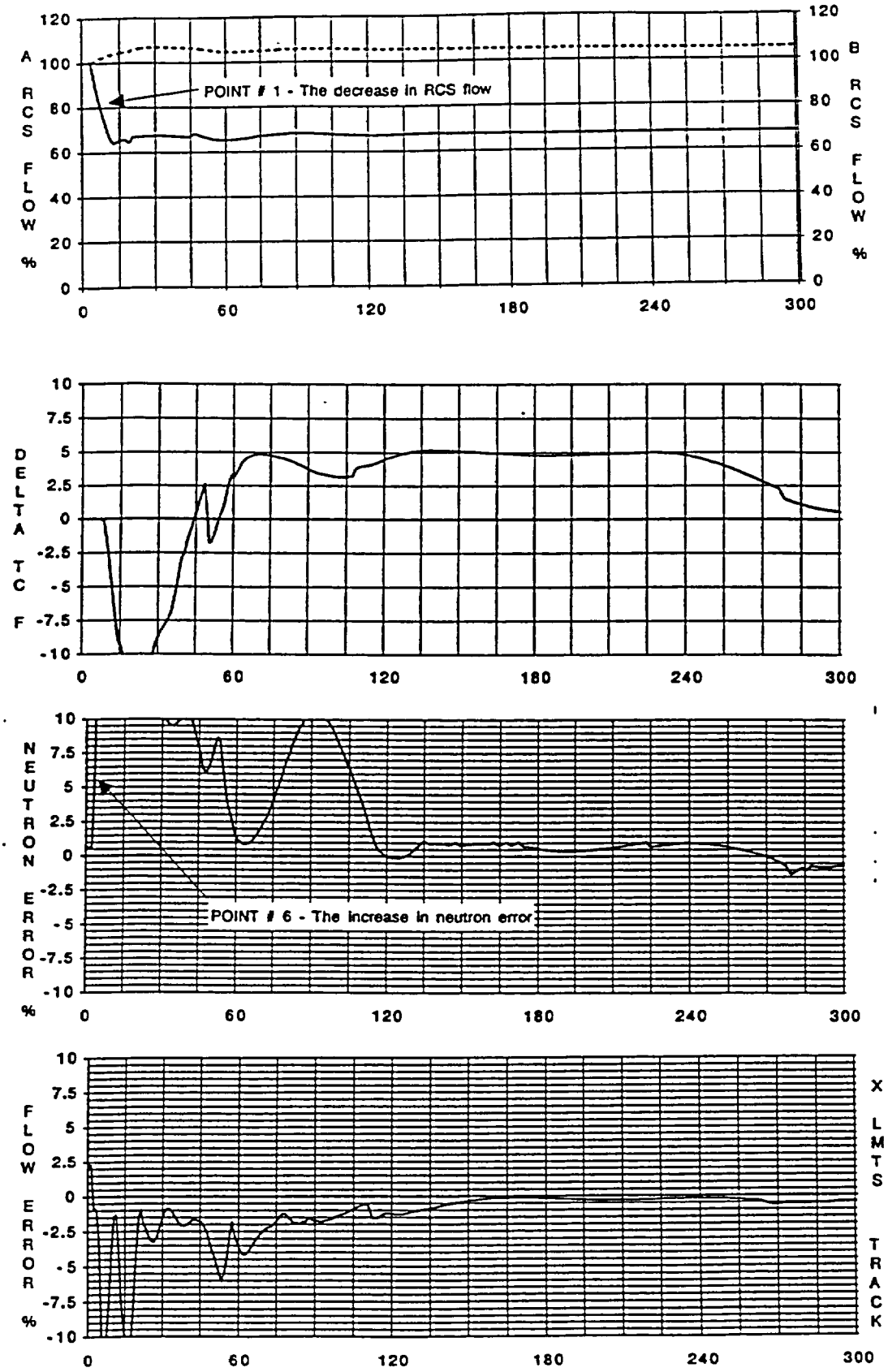

AOTSG LEVEL in
BOTSG LEVEL in
0
60
120
180
240
300


RX DEMAND %
RX POWER %
POINT # 5 - The decrease in reactor power
0
60
120
180
240
300


GROUP 7 %
GROUP 6 %
POINT # 2 - The decrease in group 6
POINT # 7 - Explain why group 7 rods are not being withdrawn in response to a positive neutron error
0
60
120
180
240
300

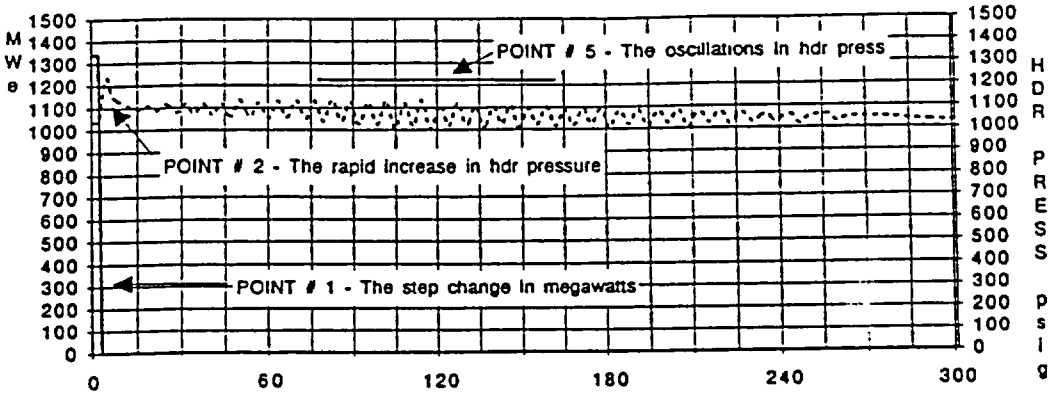

T AVG F
PZR LEVEL in
POINT # 9 - The decrease in T Average
0
60
120
180
240
300

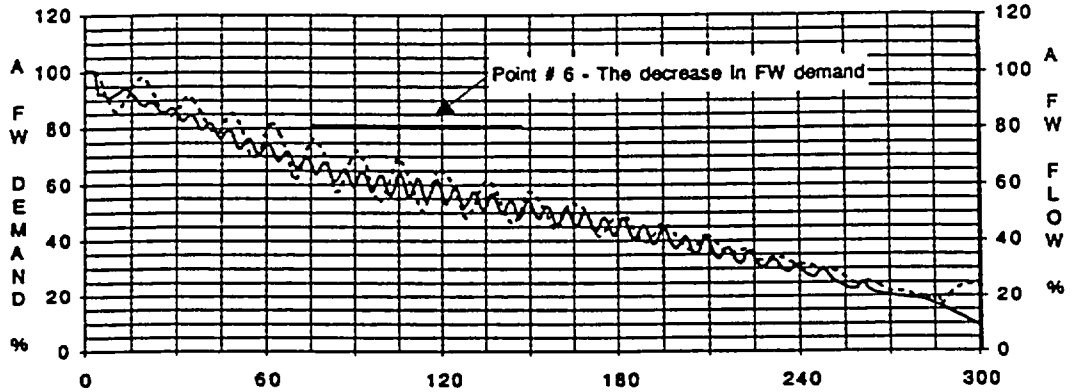

NR PRESS psig
WR PRESS psig
POINT # 10 - The decrease in RCS pressure
0
60
120
180
240
300

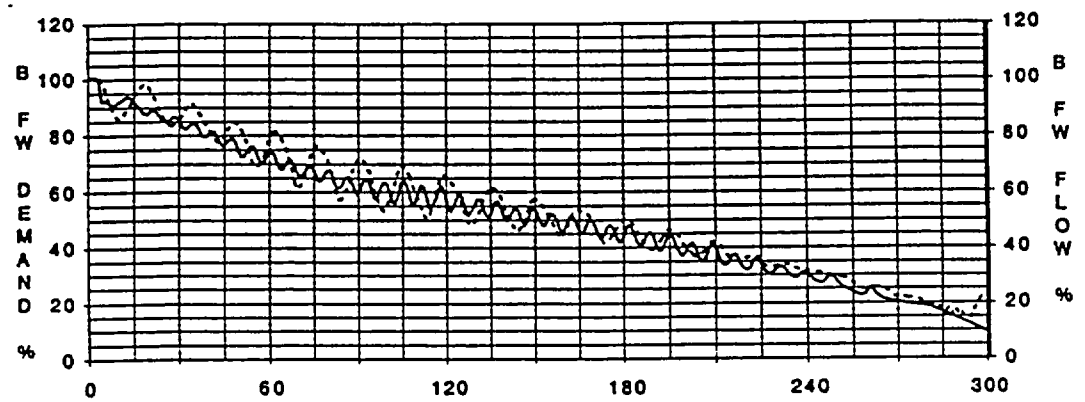

A RCS FLOW %
B RCS FLOW %
POINT # 1 - The decrease in RCS flow
DELTA TCF
NEUTRON ERROR %
POINT # 6 - The increase in neutron error
FLOW ERROR %
X LMTS TRACK
0
60
120
180
240
300

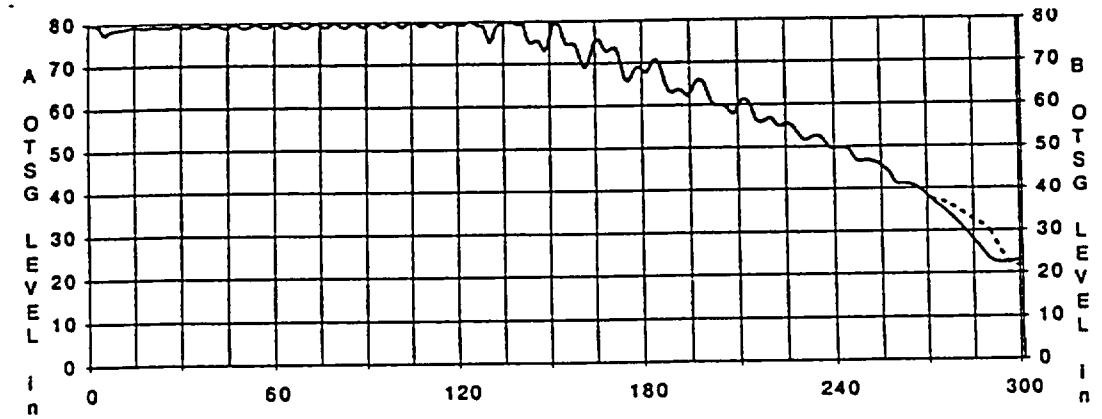
Transient 11

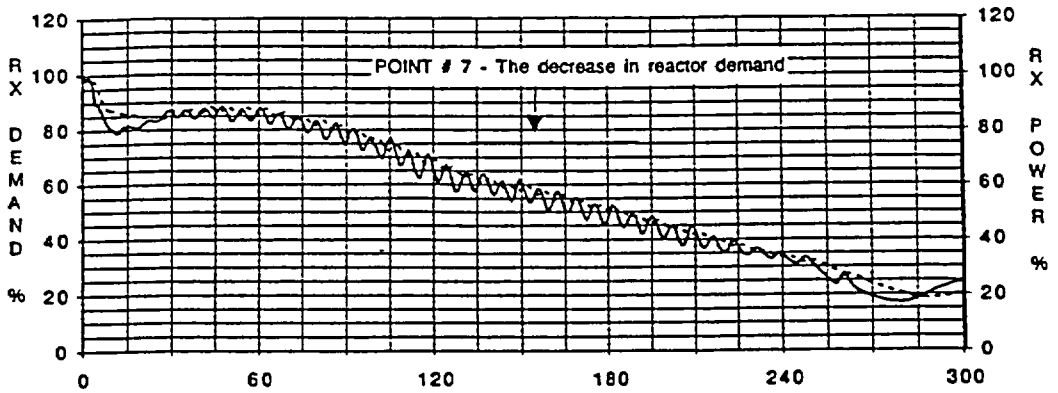
Turbine Trip

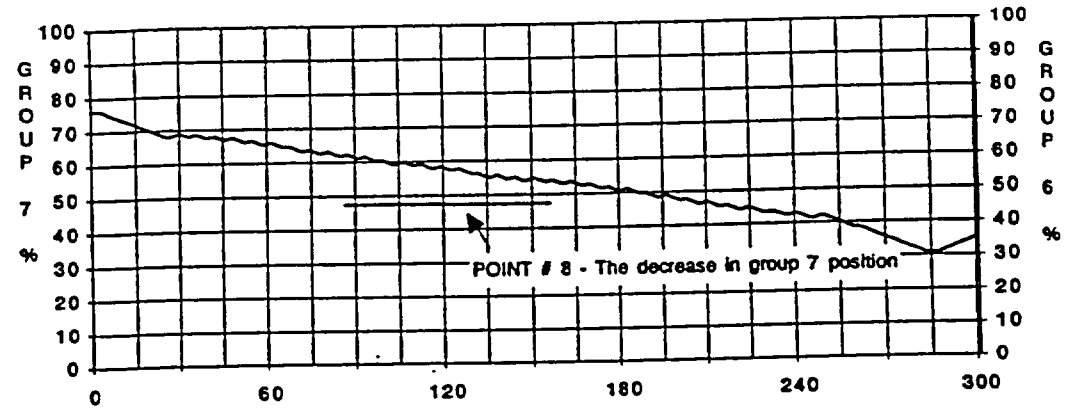

MWe
HDR PRESS psig
POINT # 5 - The oscillations in hdr press
POINT # 2 - The rapid increase in hdr pressure
POINT # 1 - The step change in megawatts

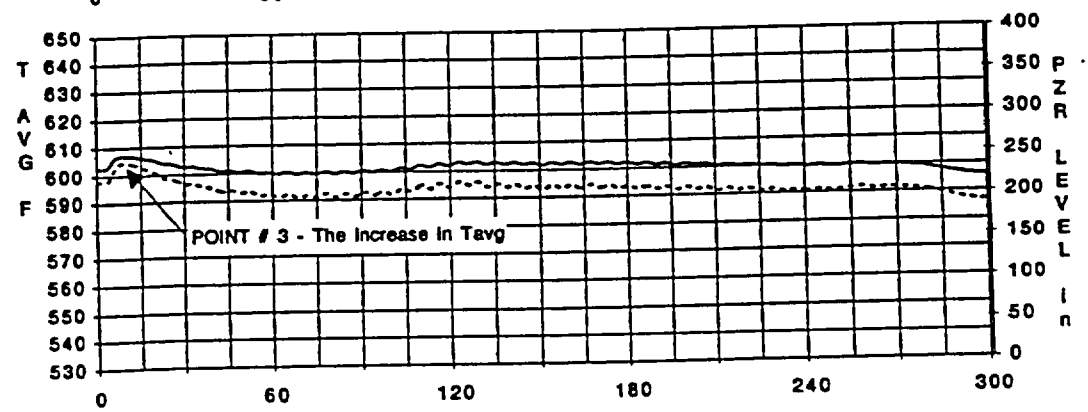

A FW DEMAND %
A FW FLOW %
Point # 6 - The decrease in FW demand

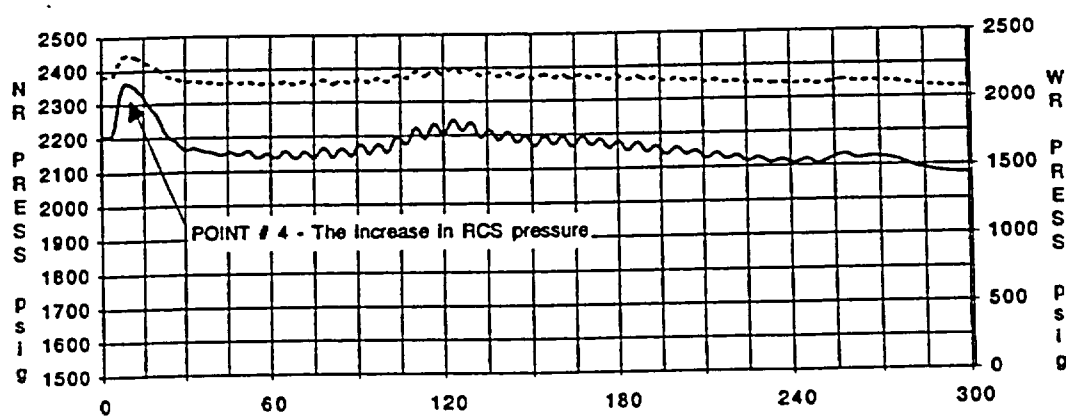

B FW DEMAND %
B FW FLOW %

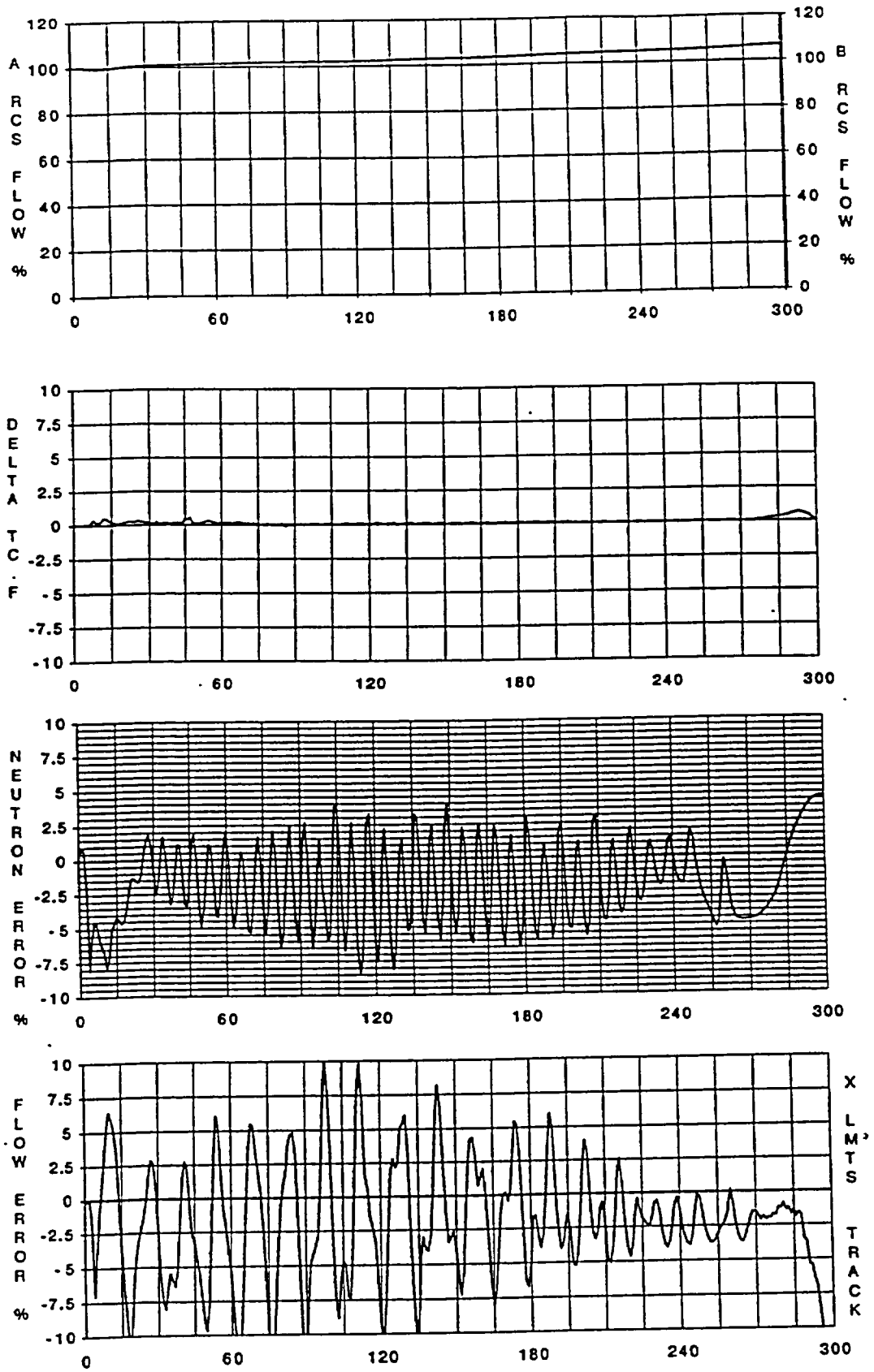

A OTSG LEVEL in
B OTSG LEVEL in


RX DEMAND %
RX POWER %
POINT # 7 - The decrease in reactor demand
0
60
120
180
240
300


GROUP 7 %
GROUP 6 %
POINT # 8 - The decrease in group 7 position
0
60
120
180
240
300

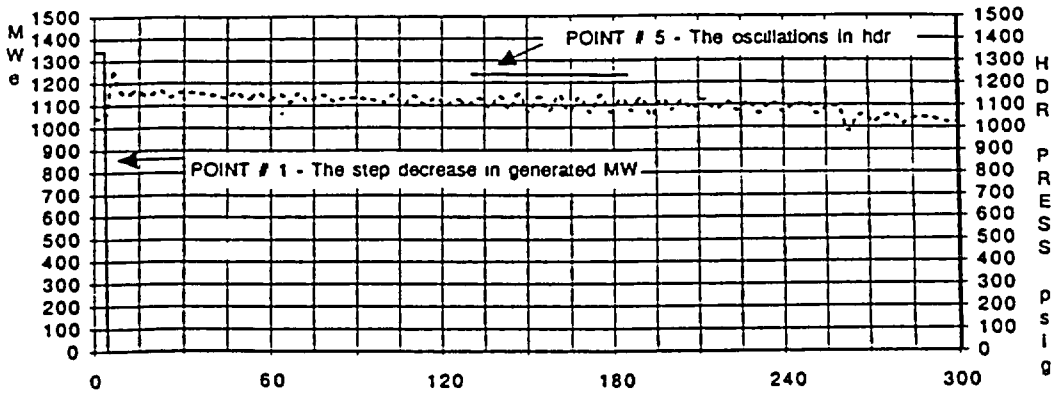

TAVG F
PZR LEVEL in
POINT # 3 - The increase in Tavg
0
60
120
180
240
300

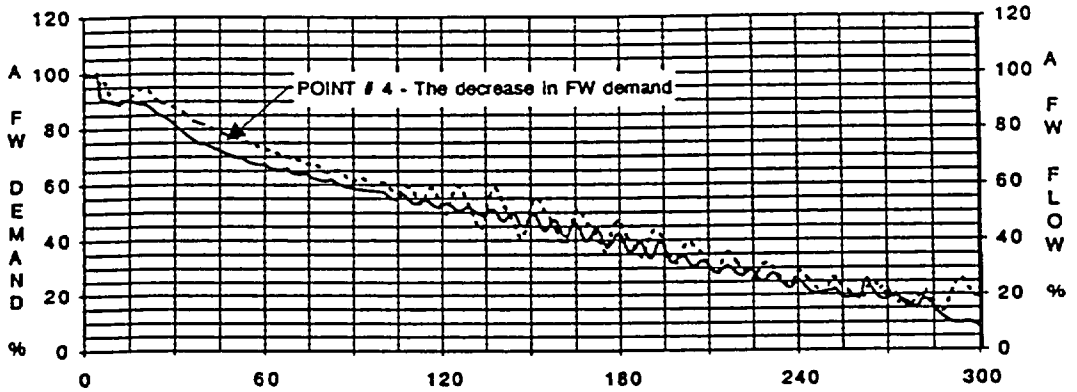

NR PRESS psig
WR PRESS psig
POINT # 4 - The increase in RCS pressure
0
60
120
180
240
300

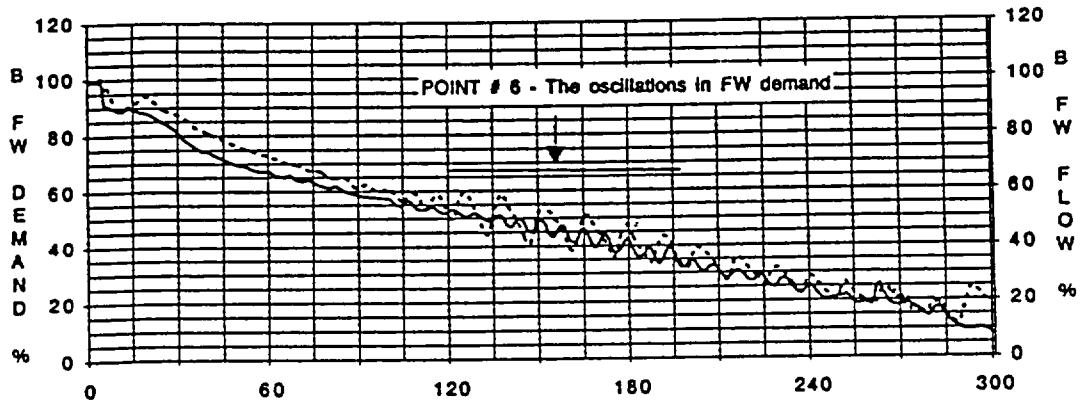

A RCS FLOW %
B RCS FLOW %
DELTA T C.F
NEUTRON ERROR %
FLOW ERROR %
X LMTS TRACK
0
60
120
180
240
300
120
100
80
60
40
20
0
10
7.5
5
2.5
-2.5
-5
-7.5
-10

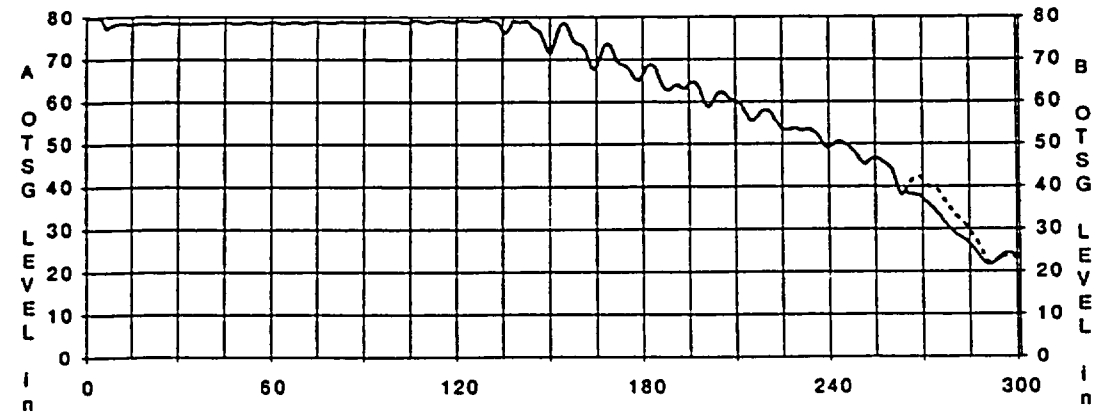
Transient 12

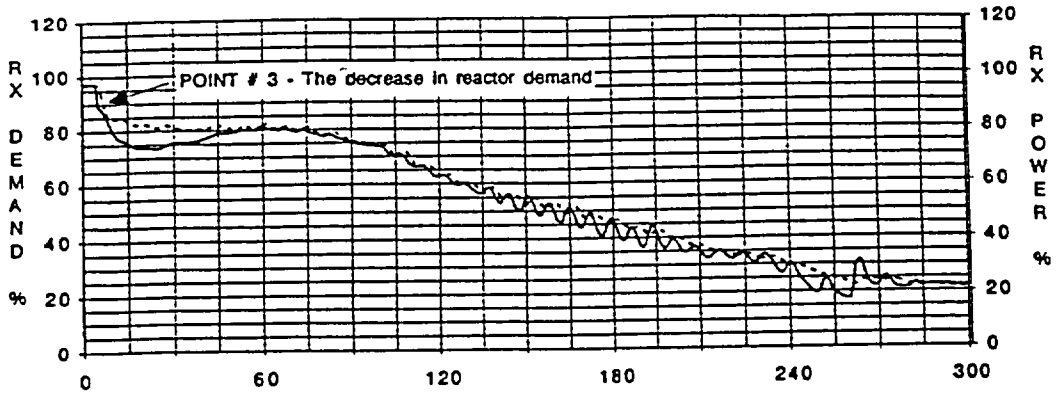
Load Rejection

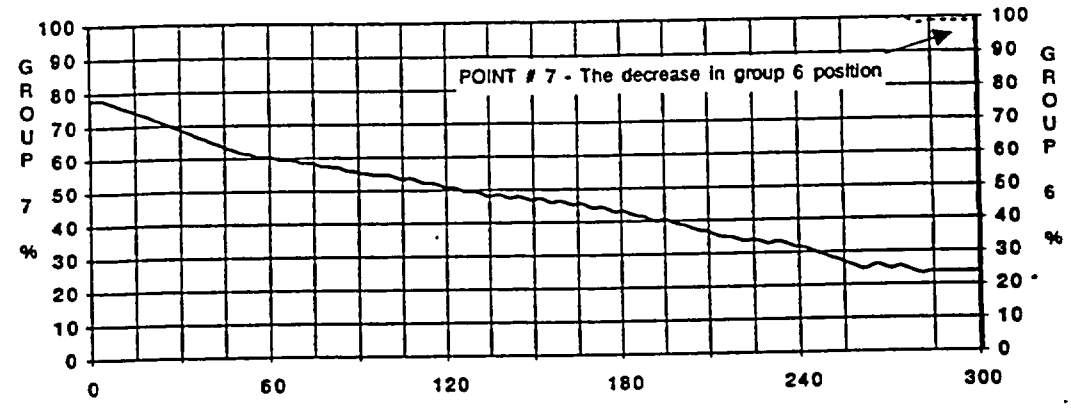

POINT # 5 - The oscillations in hdr
POINT # 1 - The step decrease in generated MW
MWe
HDR PRESS psig
0
60
120
180
240
300

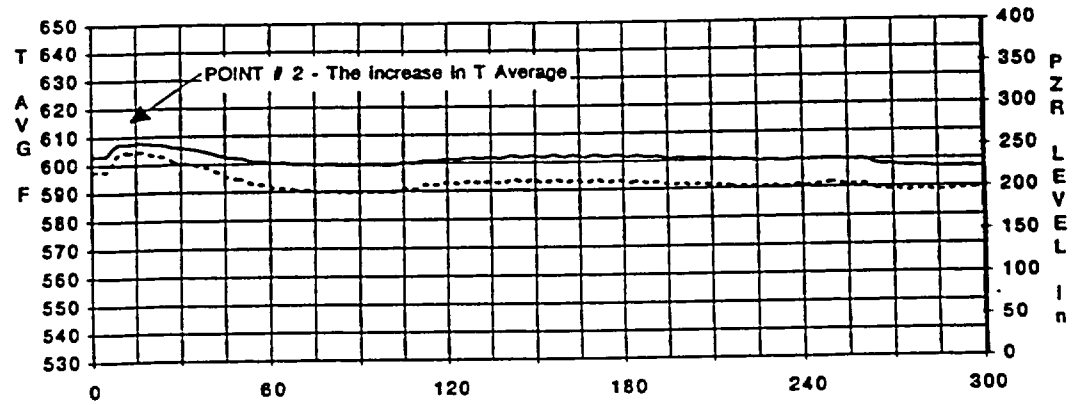

POINT # 4 - The decrease in FW demand
A FW DEMAND %
A FW FLOW %
0
60
120
180
240
300

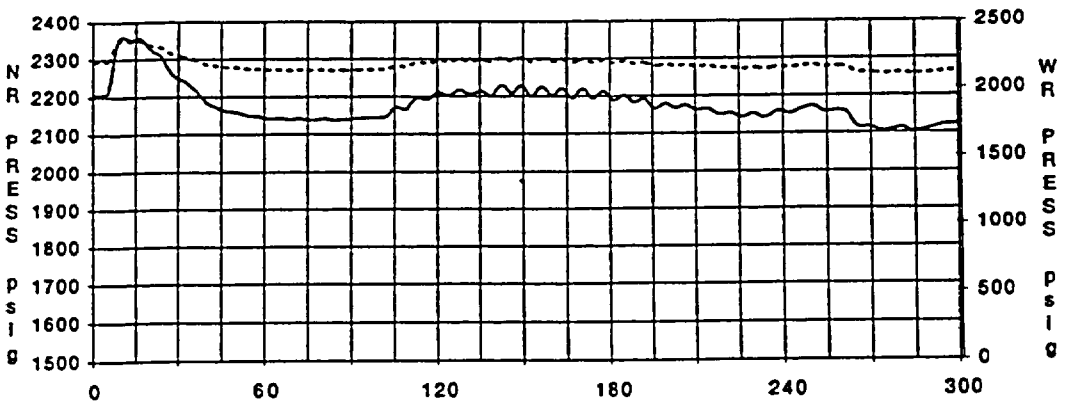

POINT # 6 - The oscillations in FW demand
B FW DEMAND %
B FW FLOW %
0
60
120
180
240
300

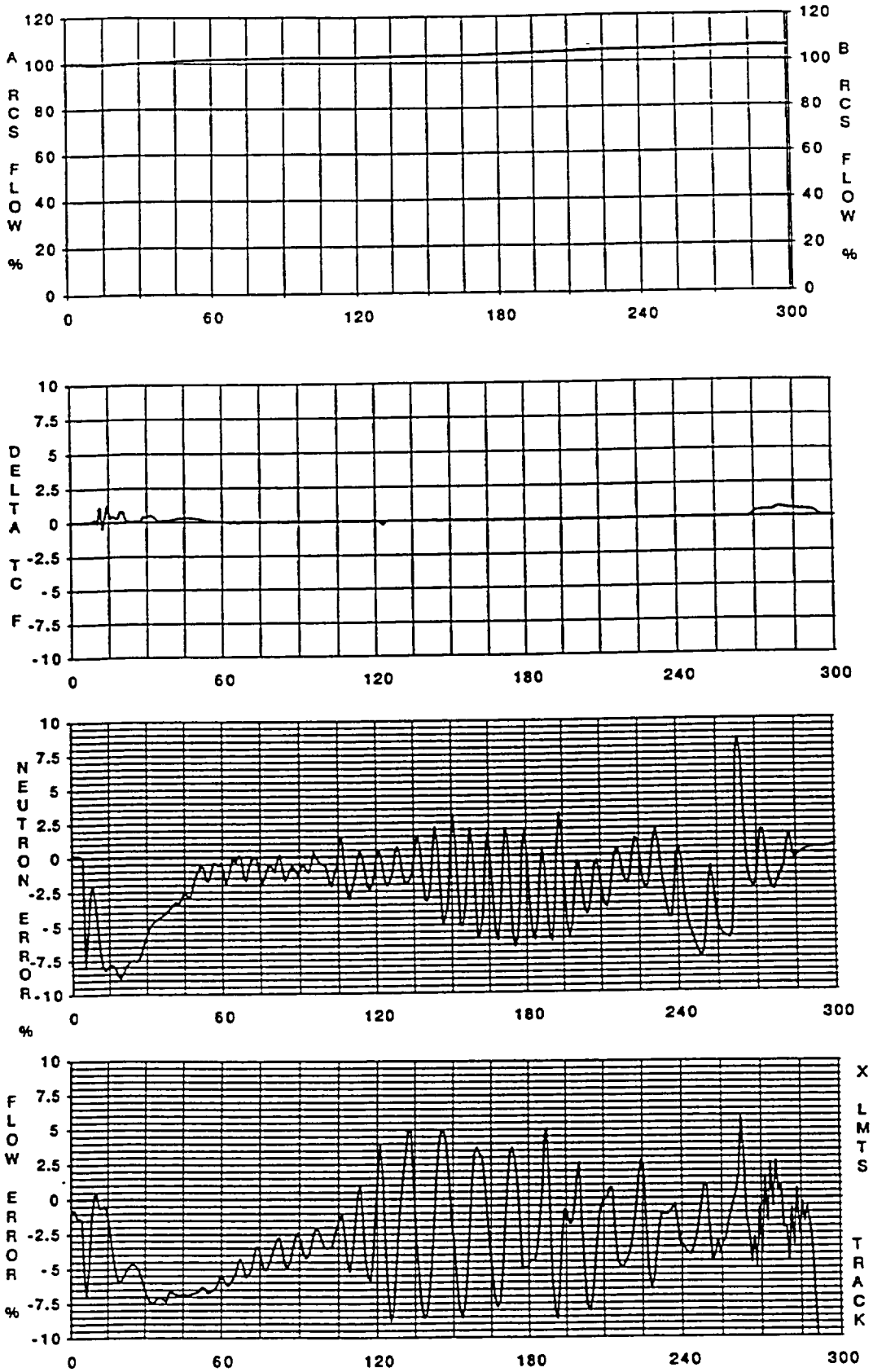

A OTSG LEVEL in
B OTSG LEVEL in
0
60
120
180
240
300


POINT # 3 - The decrease in reactor demand
RX DEMAND %
RX POWER %
0
20
40
60
80
100
120
0
60
120
180
240
300


POINT # 7 - The decrease in group 6 position
GROUP 7 %
GROUP 6 %
0
10
20
30
40
50
60
70
80
90
100
0
60
120
180
240
300

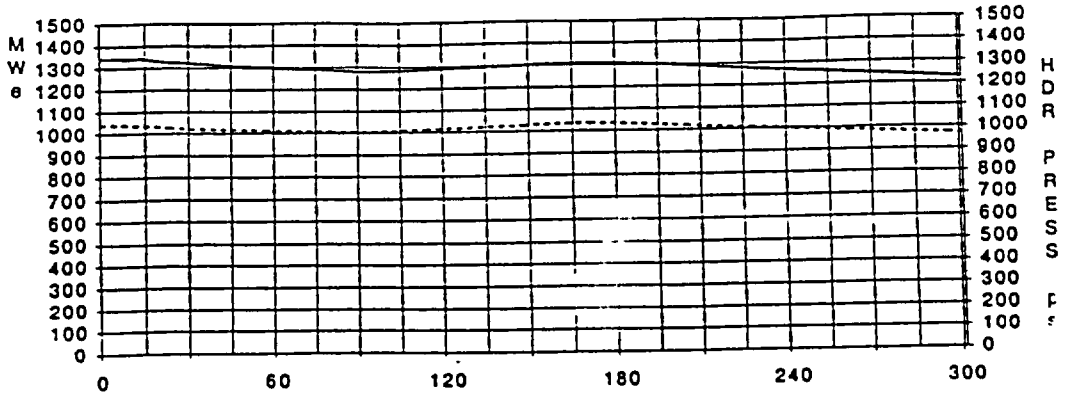

POINT # 2 - The increase in T Average
T AVG F
PZR LEVEL in
530
540
550
560
570
580
590
600
610
620
630
640
650
0
50
100
150
200
250
300
350
400
0
60
120
180
240
300

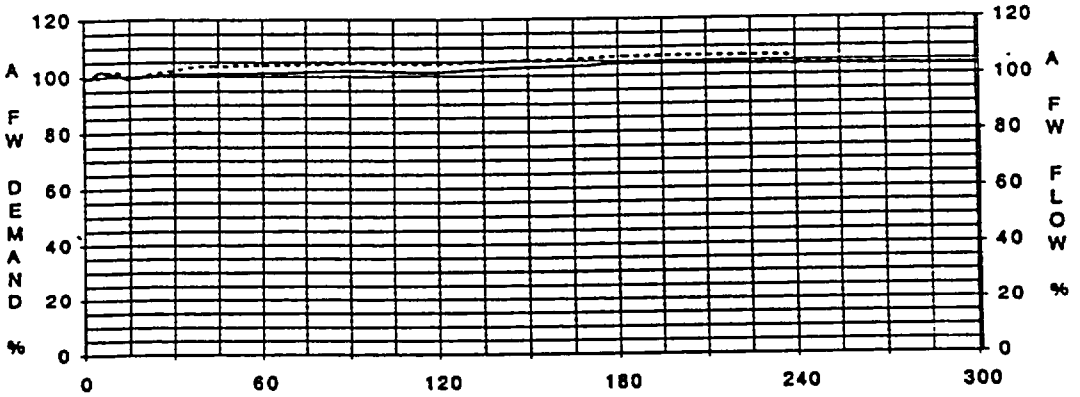

NR PRESS psig
WR PRESS psig
1500
1600
1700
1800
1900
2000
2100
2200
2300
2400
0
500
1000
1500
2000
2500
0
60
120
180
240
300

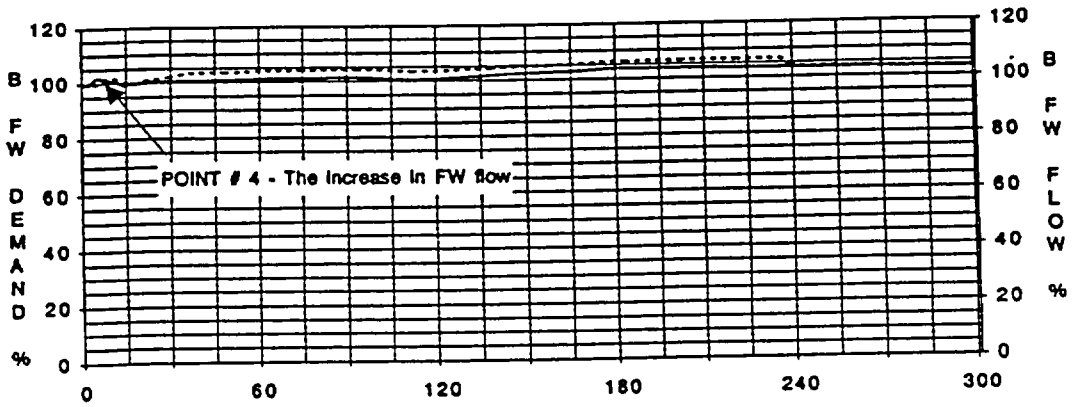

A RCS FLOW %
B RCS FLOW %
DELTA TCF
NEUTRON ERROR %
FLOW ERROR %
X LMTS TRACK
120
100
80
60
40
20
0
10
7.5
5
2.5
-2.5
-5
-7.5
-10
0
60
120
180
240
300

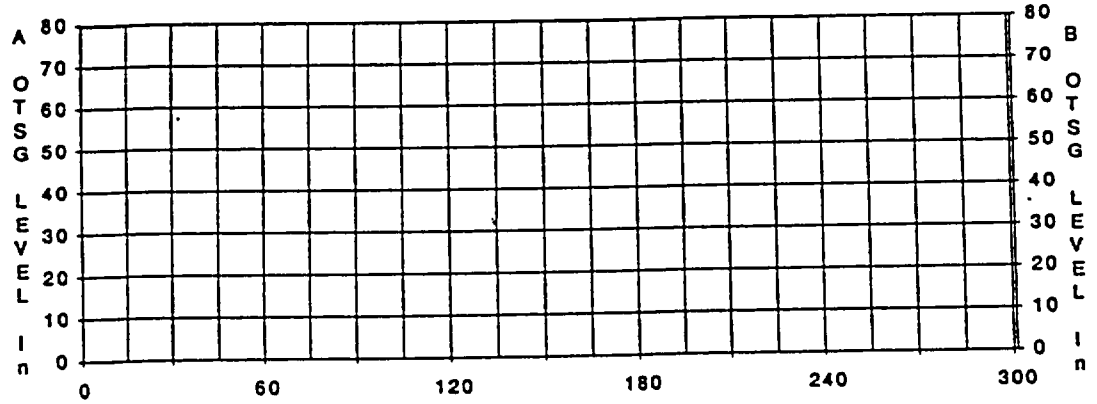
Transient 13

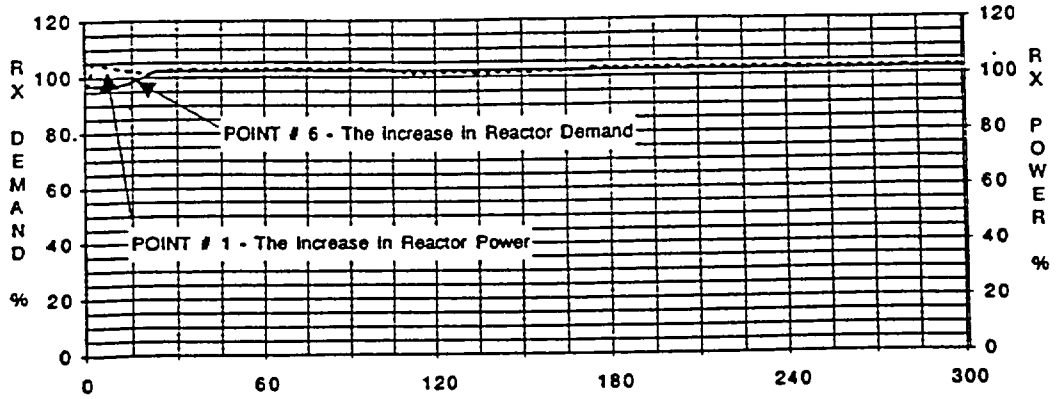
Power Range Fails High

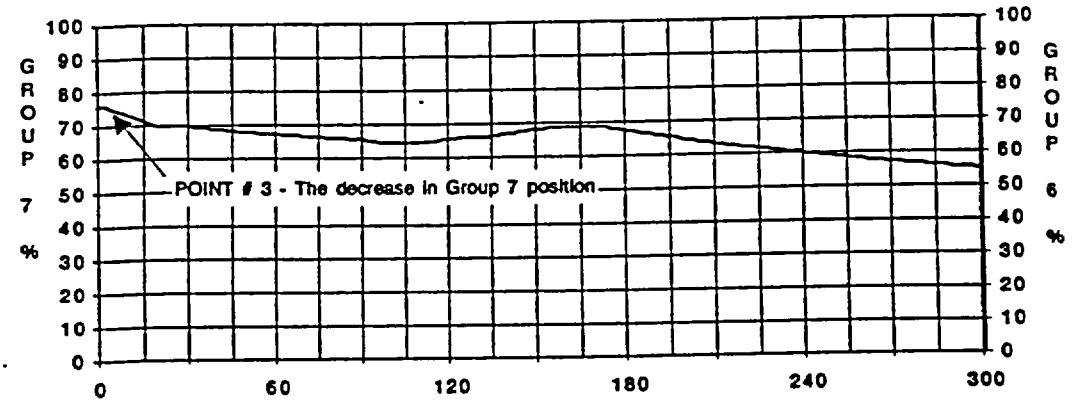

MWe
HDR PRESS psi
0
60
120
180
240
300

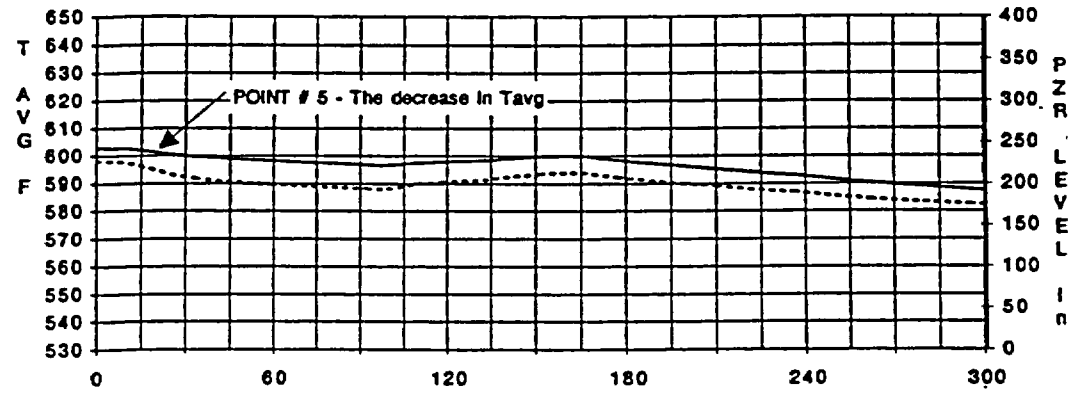

A FW DEMAND %
A FW FLOW %
0
60
120
180
240
300

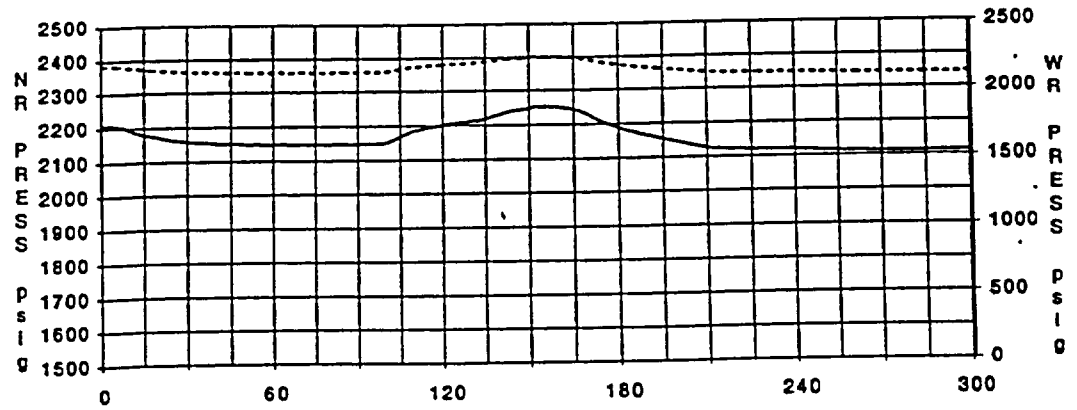

B FW DEMAND %
B FW FLOW %
POINT # 4 - The Increase In FW flow
0
60
120
180
240
300

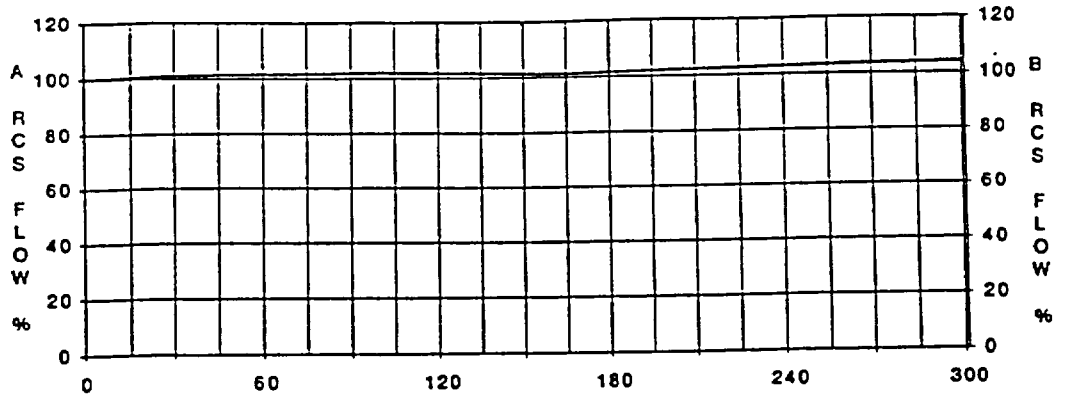

A OTSG LEVEL In
B OTSG LEVEL In
0
60
120
180
240
300

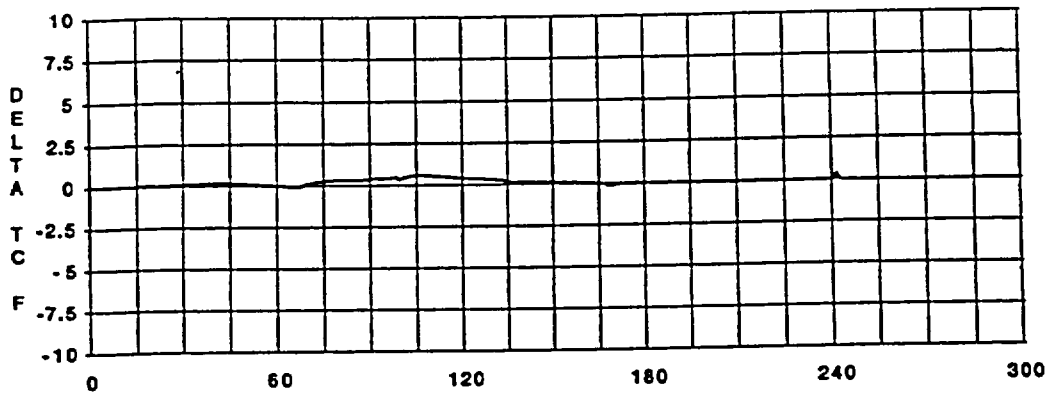

RX DEMAND %
RX POWER %
POINT # 6 - The increase in Reactor Demand
POINT # 1 - The Increase in Reactor Power
0
60
120
180
240
300

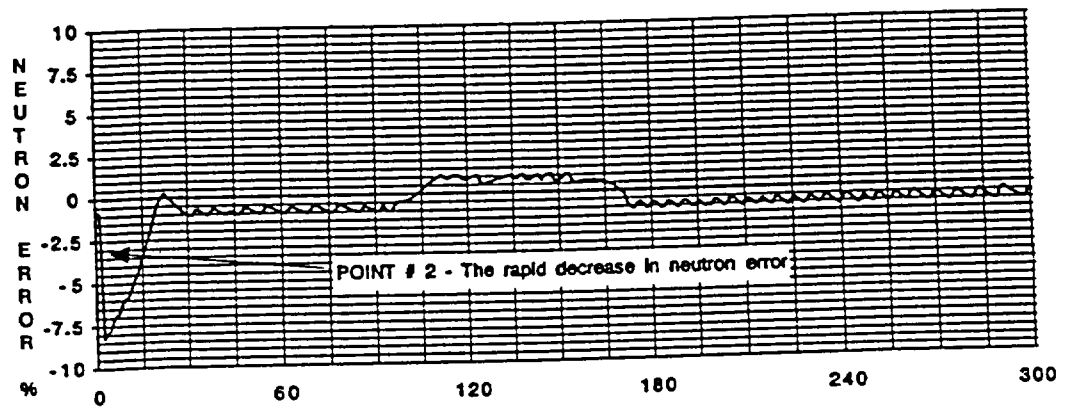

GROUP 7 %
GROUP 6 %
POINT # 3 - The decrease in Group 7 position
0
60
120
180
240
300

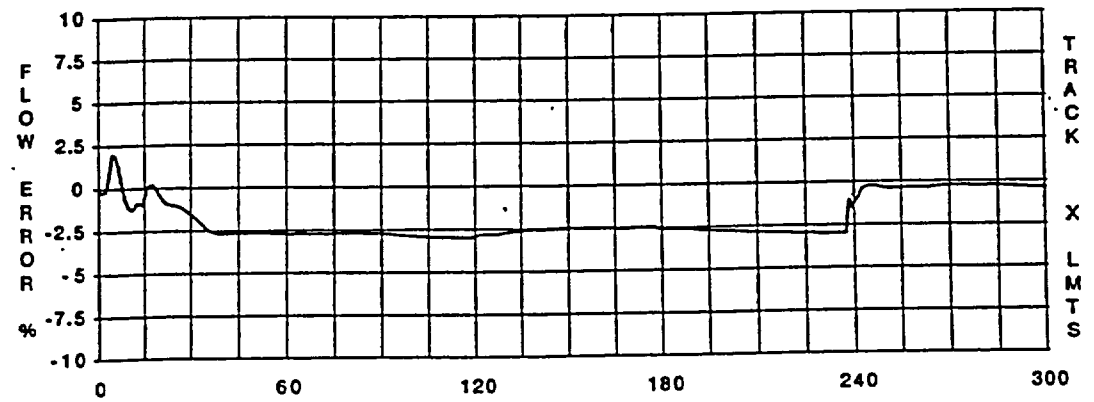

TAVG F
PZR LEVEL in
POINT # 5 - The decrease in Tavg
0
60
120
180
240
300


NR PRESS psig
WR PRESS psig
0
60
120
180
240
300


A RCS FLOW %
B RCS FLOW %
120
100
80
60
40
20
0
0
60
120
180
240
300

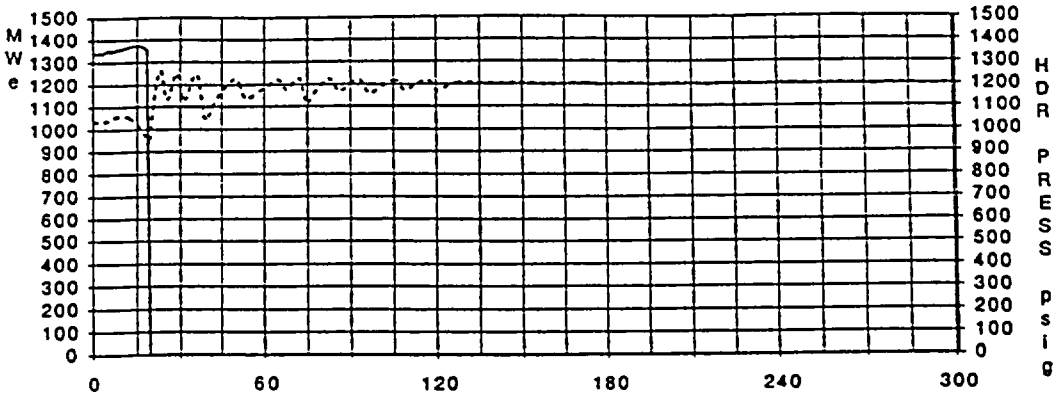

DELTA TCF
10
7.5
5
2.5
0
-2.5
-5
-7.5
-10
0
60
120
180
240
300

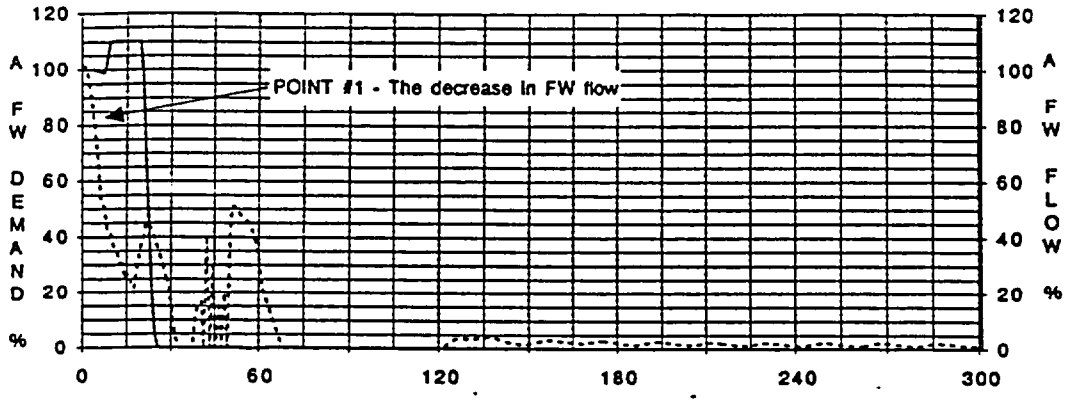

NEUTRON ERROR %
10
7.5
5
2.5
0
-2.5
-5
-7.5
-10
0
60
120
180
240
300
POINT # 2 - The rapid decrease in neutron error

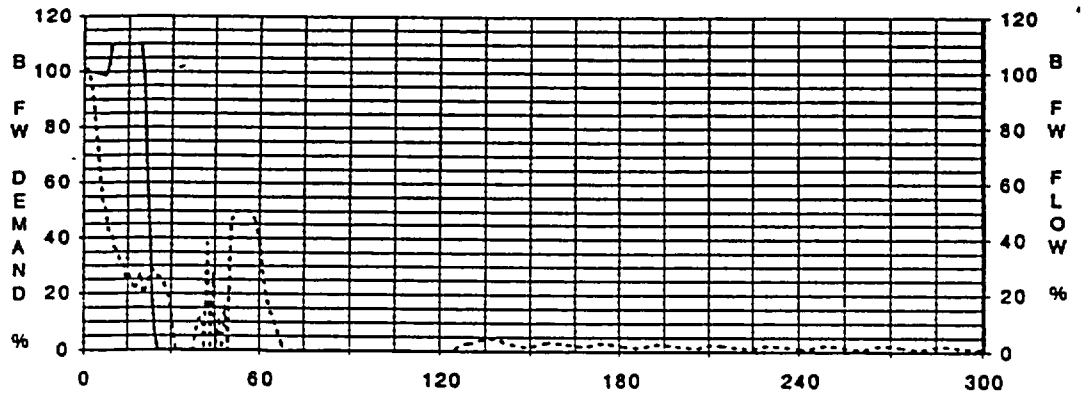

FLOW ERROR %
TRACK X LMTS
10
7.5
5
2.5
0
-2.5
-5
-7.5
-10
0
60
120
180
240
300

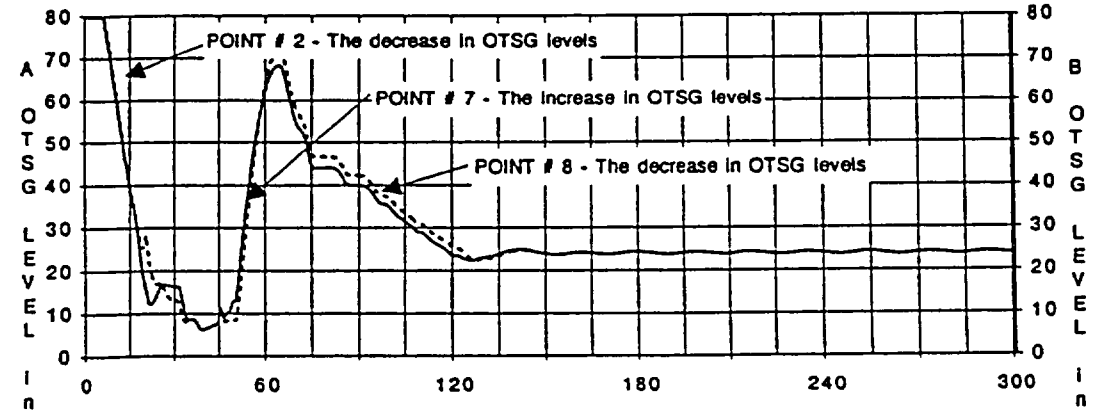
Transient - 14

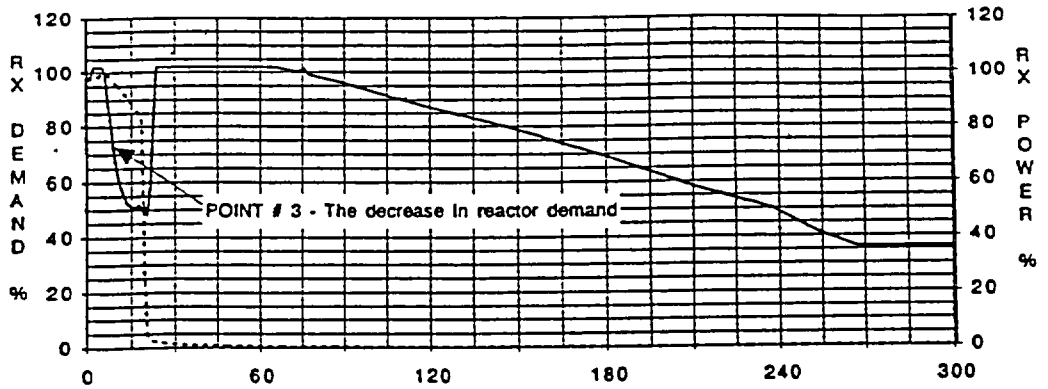
T_h Fails Low - BTU Limits Enabled

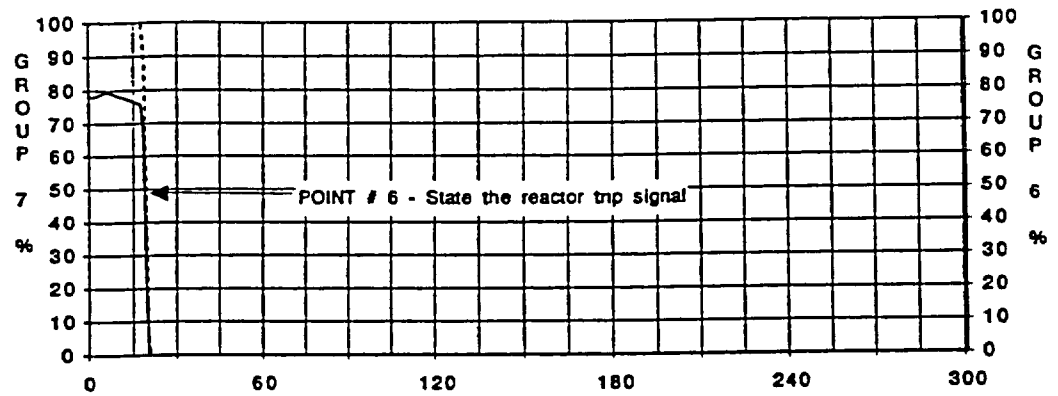

MWe
HDR PRESS psig
0
60
120
180
240
300

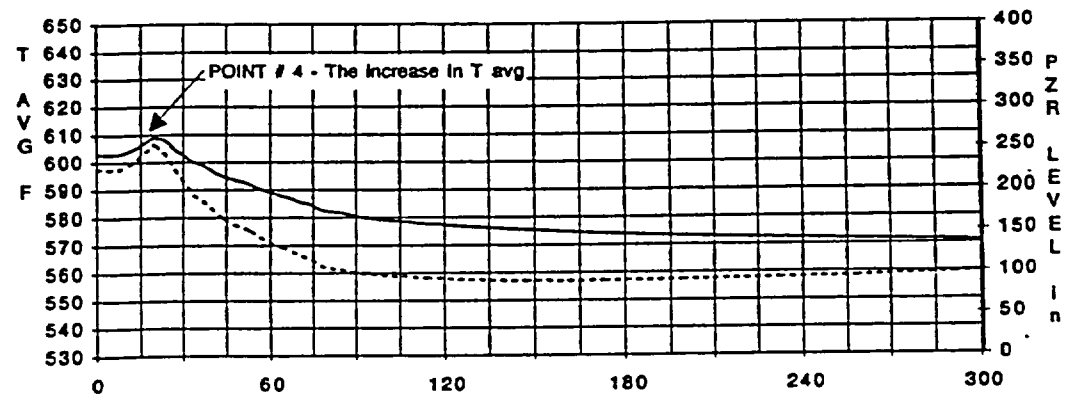

A FW DEMAND %
A FW FLOW %
POINT #1 - The decrease in FW flow
0
60
120
180
240
300

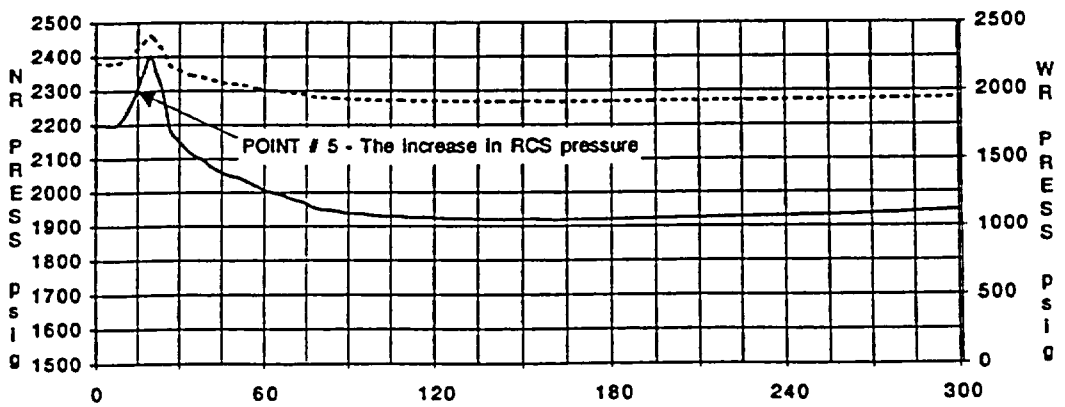

B FW DEMAND %
B FW FLOW %
0
60
120
180
240
300

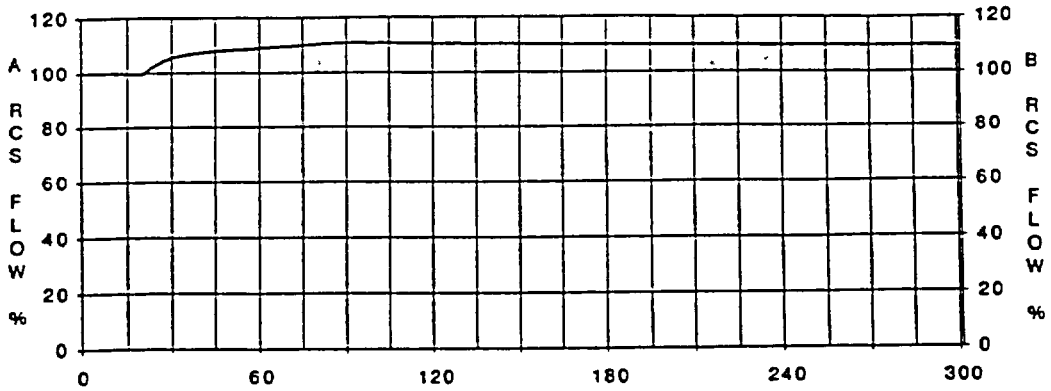

A OTSG LEVEL in
B OTSG LEVEL in
POINT # 2 - The decrease in OTSG levels
POINT # 7 - The increase in OTSG levels
POINT # 8 - The decrease in OTSG levels
0
60
120
180
240
300

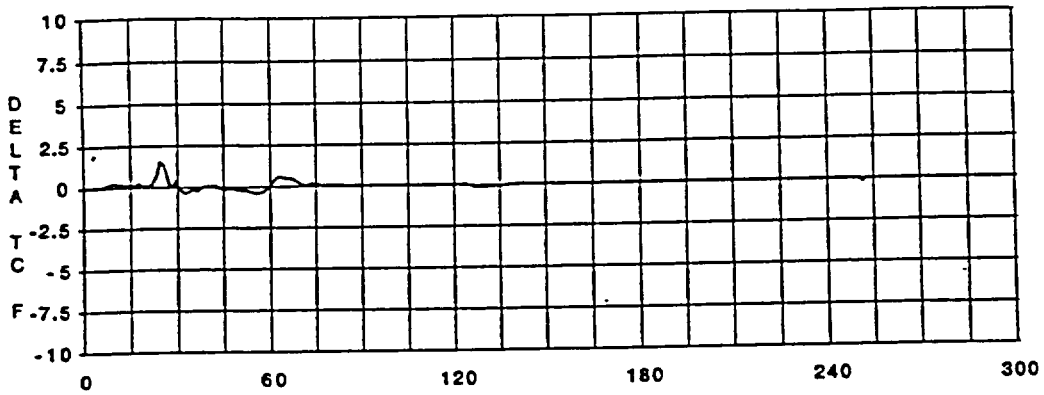

RX DEMAND %
RX POWER %
POINT # 3 - The decrease in reactor demand
0
60
120
180
240
300

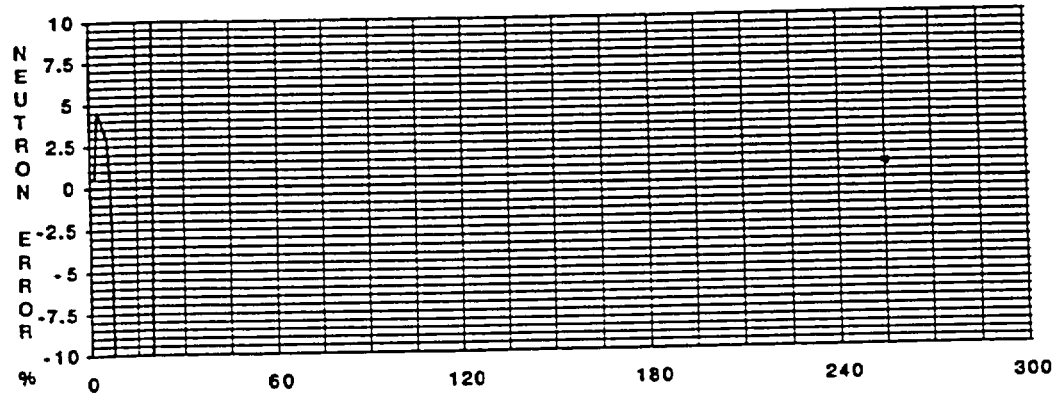

GROUP 7 %
GROUP 6 %
POINT # 6 - State the reactor trip signal
0
60
120
180
240
300

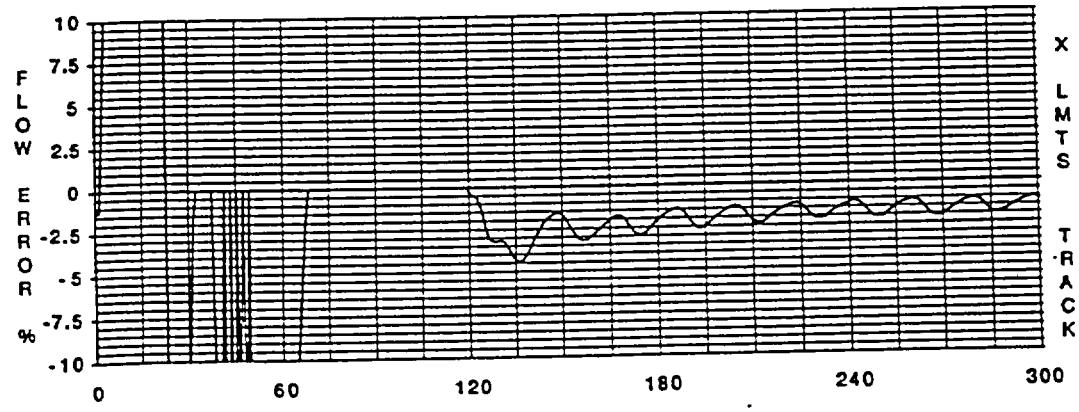

T AVG F
PZR LEVEL in
POINT # 4 - The increase in T avg
0
60
120
180
240
300


NR PRESS psig
WR PRESS psig
POINT # 5 - The increase in RCS pressure
0
60
120
180
240
300


A RCS FLOW %
B RCS FLOW %
120
100
80
60
40
20
0
0
60
120
180
240
300

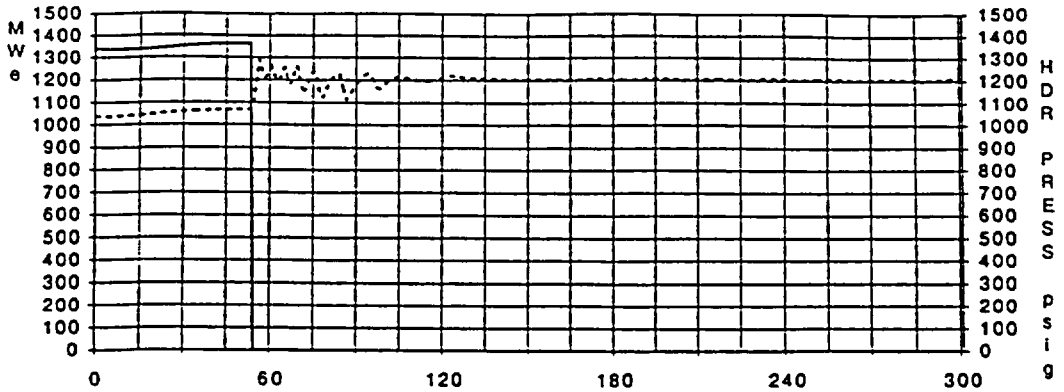

DELTA TCF
10
7.5
5
2.5
0
-2.5
-5
-7.5
-10
0
60
120
180
240
300

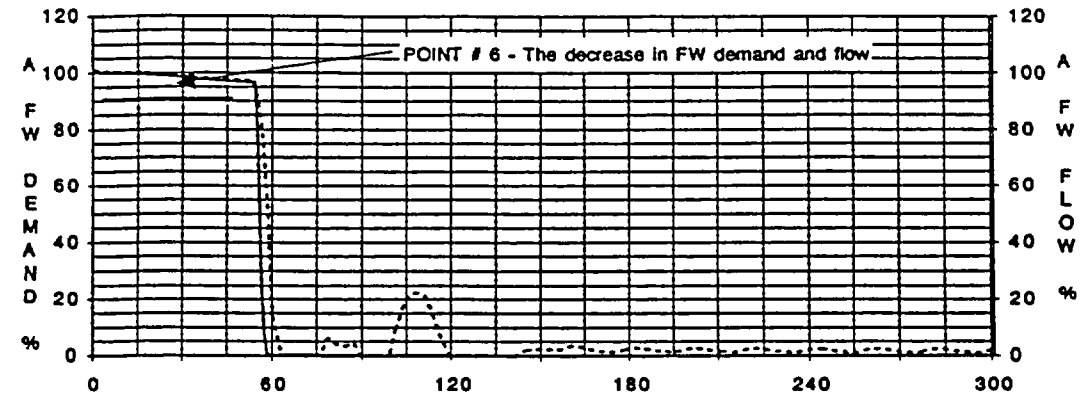

NEUTRON ERROR %
10
7.5
5
2.5
0
-2.5
-5
-7.5
-10
0
60
120
180
240
300

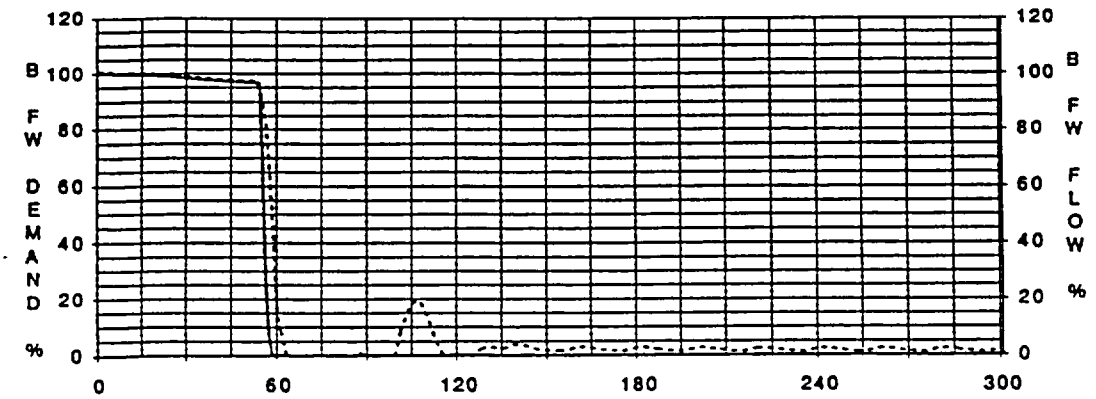

FLOW ERROR %
X LMTS TRACK
10
7.5
5
2.5
0
-2.5
-5
-7.5
-10
0
60
120
180
240
300

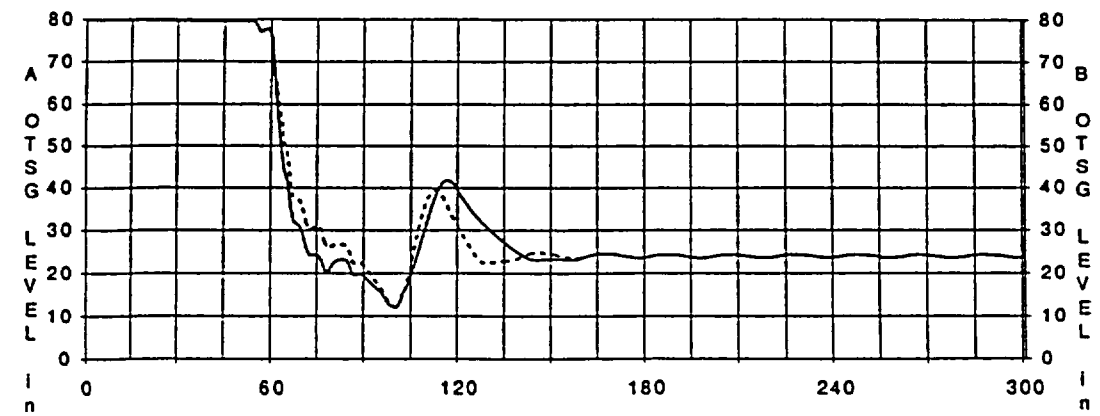
Transient 15

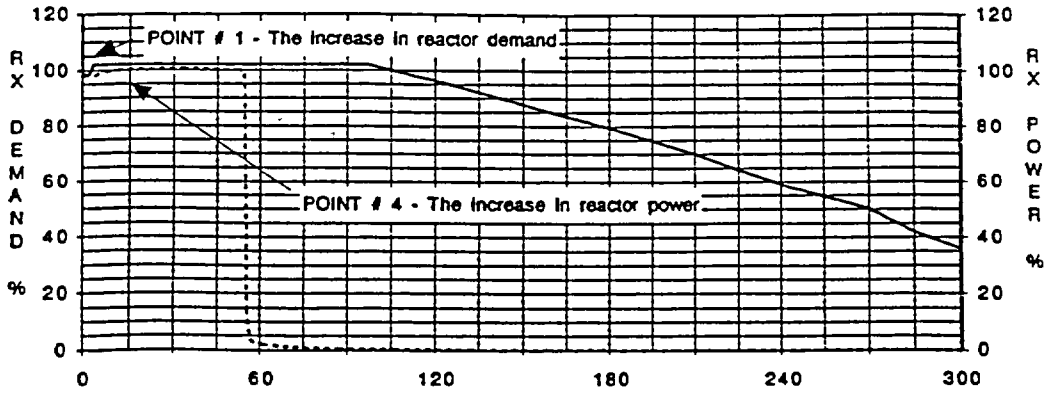
T_h Hot Fails Low

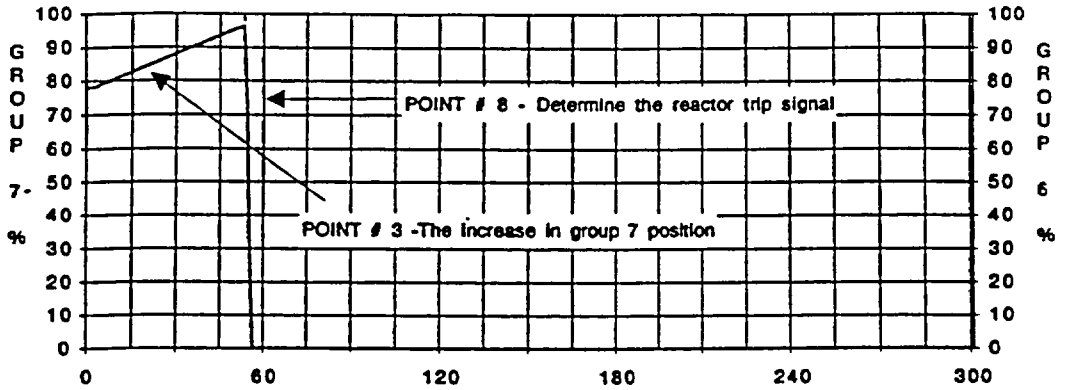

MWe
HDR PRESS psig
1500
1400
1300
1200
1100
1000
900
800
700
600
500
400
300
200
100
0
0
60
120
180
240
300

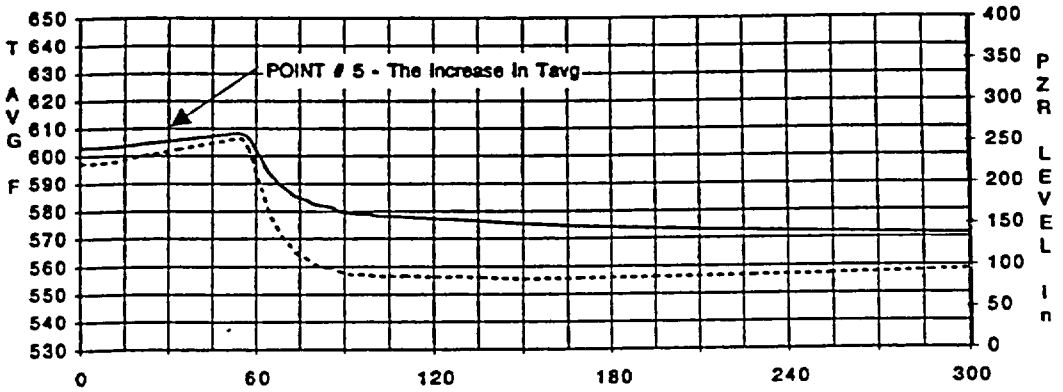

POINT # 6 - The decrease in FW demand and flow
A FW DEMAND %
A FW FLOW %
120
100
80
60
40
20
0
0
60
120
180
240
300

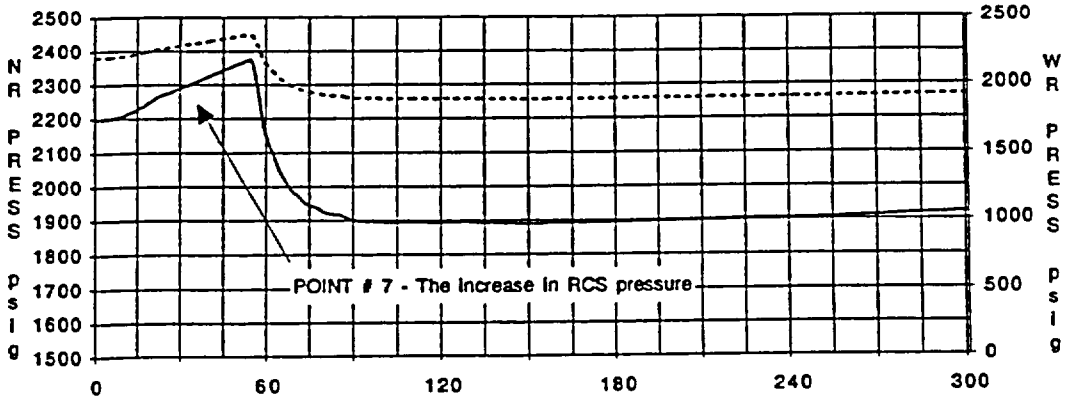

B FW DEMAND %
B FW FLOW %
120
100
80
60
40
20
0
0
60
120
180
240
300

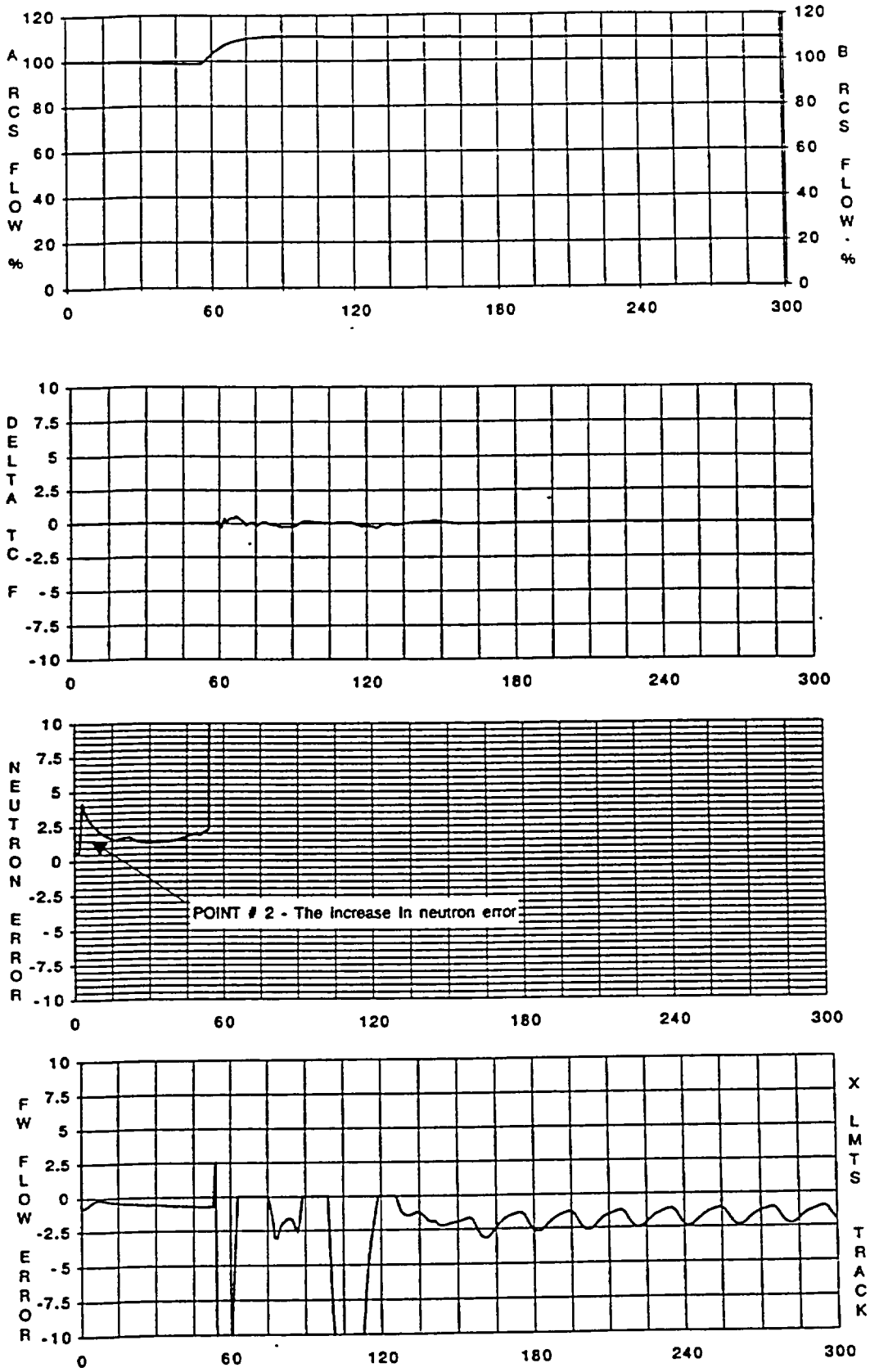

A OTSG LEVEL in
B OTSG LEVEL in
80
70
60
50
40
30
20
10
0
0
60
120
180
240
300


POINT # 1 - The increase in reactor demand
POINT # 4 - The increase in reactor power
RX DEMAND %
RX POWER %
0
20
40
60
80
100
120
0
60
120
180
240
300


POINT # 8 - Determine the reactor trip signal
POINT # 3 -The increase in group 7 position
GROUP 7- %
GROUP 6 %
0
10
20
30
40
50
60
70
80
90
100
0
60
120
180
240
300

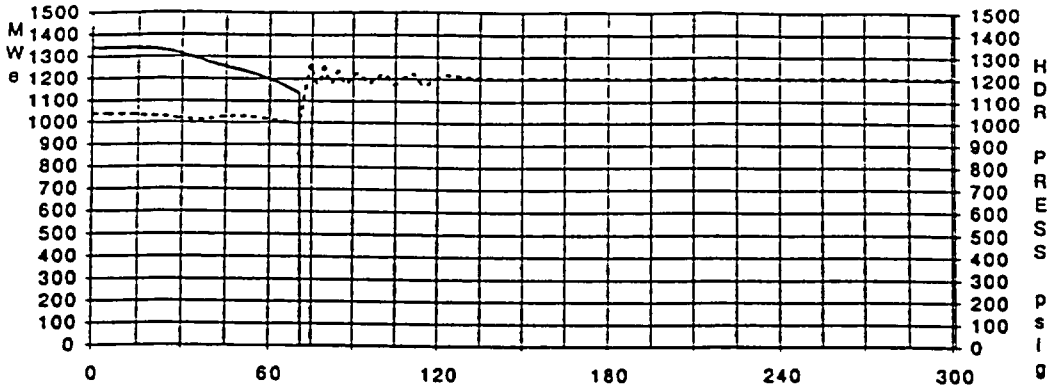

POINT # 5 - The increase in Tavg
TAVG F
PZR LEVEL in
530
540
550
560
570
580
590
600
610
620
630
640
650
0
50
100
150
200
250
300
350
400
0
60
120
180
240
300

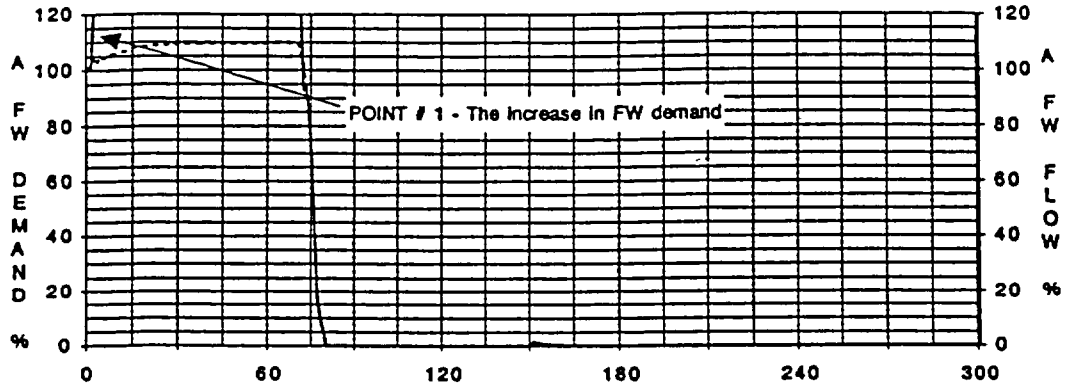

POINT # 7 - The increase in RCS pressure
NR PRESS psig
WR PRESS psig
1500
1600
1700
1800
1900
2000
2100
2200
2300
2400
2500
0
500
1000
1500
2000
2500
0
60
120
180
240
300

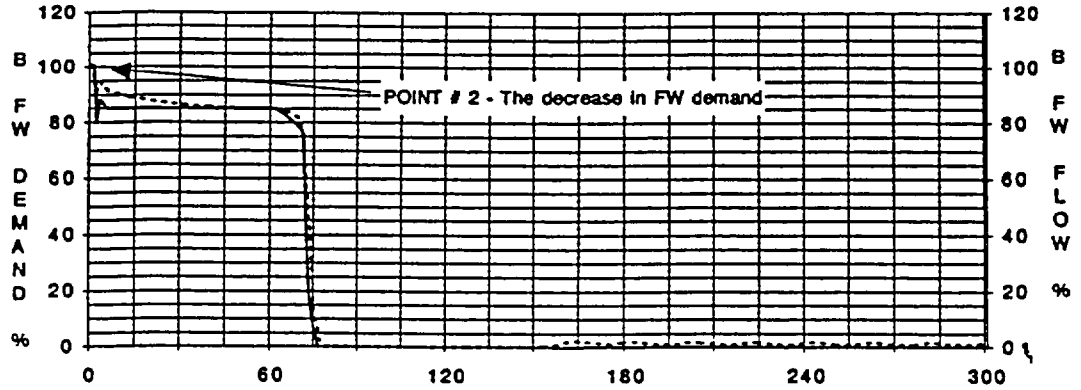

A RCS FLOW %
B RCS FLOW - %
DELTA TCF
NEUTRON ERROR
POINT # 2 - The increase in neutron error
FW FLOW ERROR
X LMTS TRACK
0
60
120
180
240
300

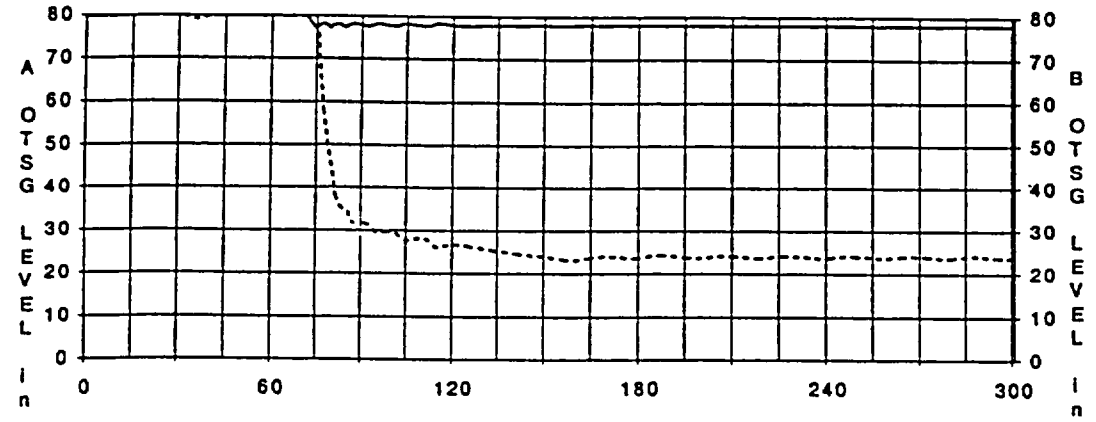
Transient 16

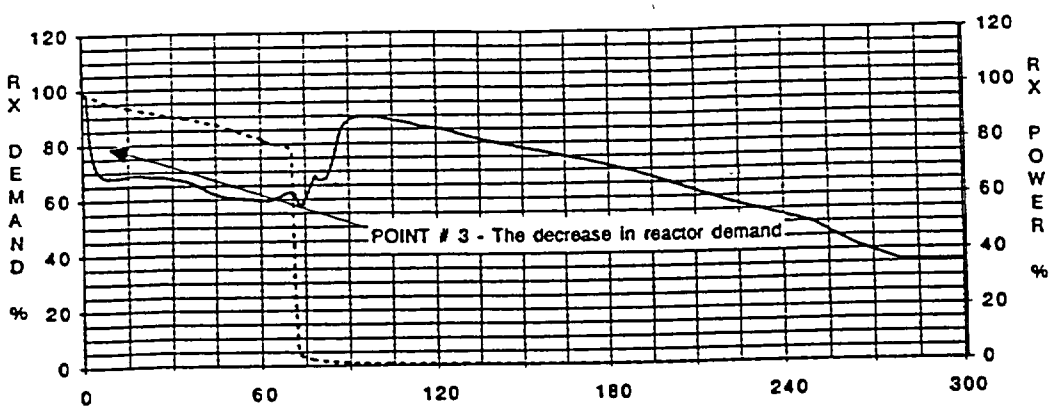
Cold Leg Temperature Fails High

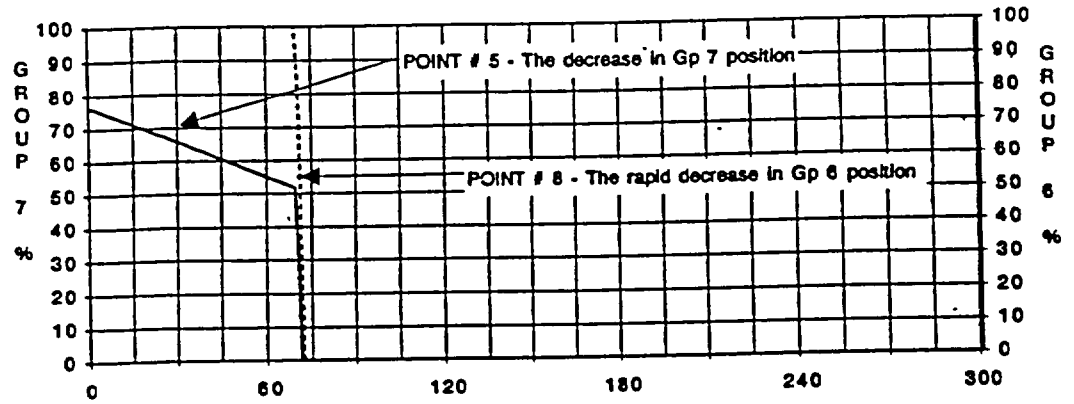

MWe
HDR PRESS psig
1500
1400
1300
1200
1100
1000
900
800
700
600
500
400
300
200
100
0
0
60
120
180
240
300

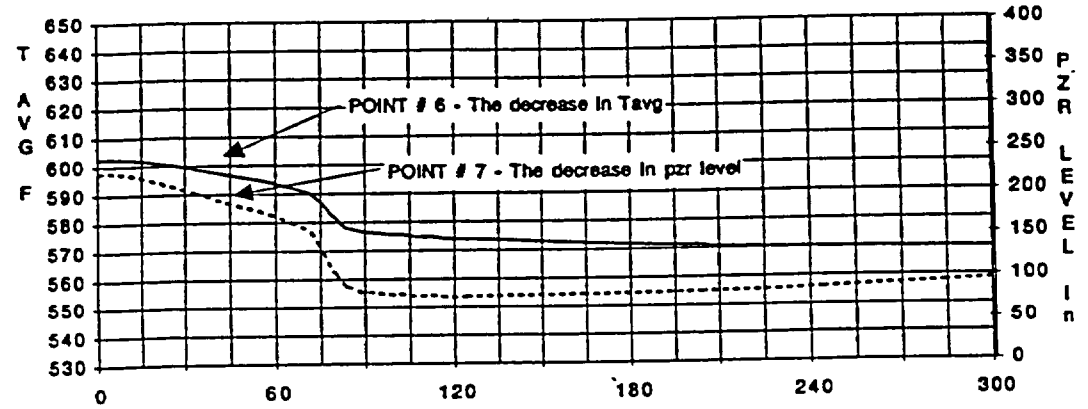

A FW DEMAND %
A FW FLOW %
POINT # 1 - The increase in FW demand
120
100
80
60
40
20
0
0
60
120
180
240
300

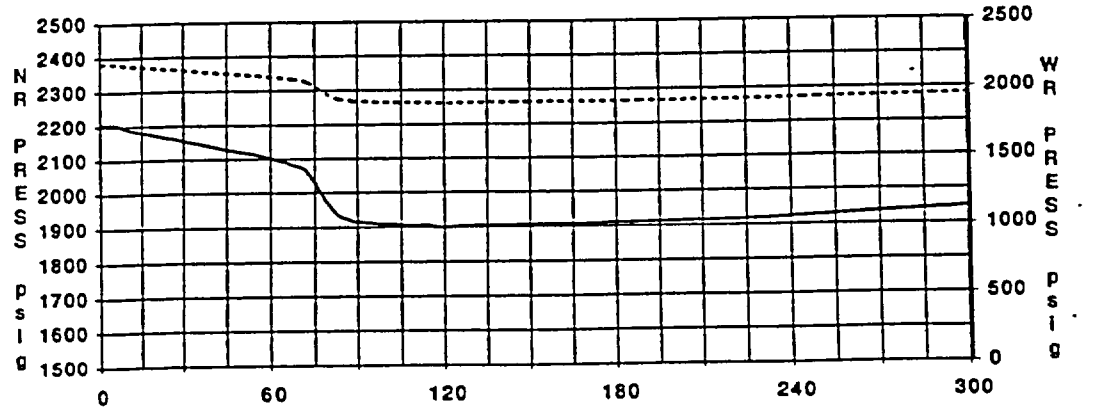

B FW DEMAND %
B FW FLOW %
POINT # 2 - The decrease in FW demand
120
100
80
60
40
20
0
0
60
120
180
240
300

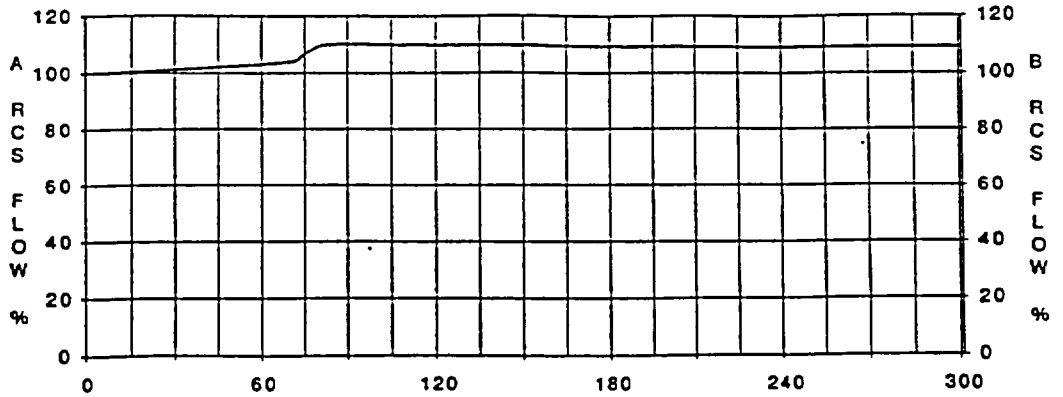

A OTSG LEVEL in
B OTSG LEVEL in
80
70
60
50
40
30
20
10
0
0
60
120
180
240
300

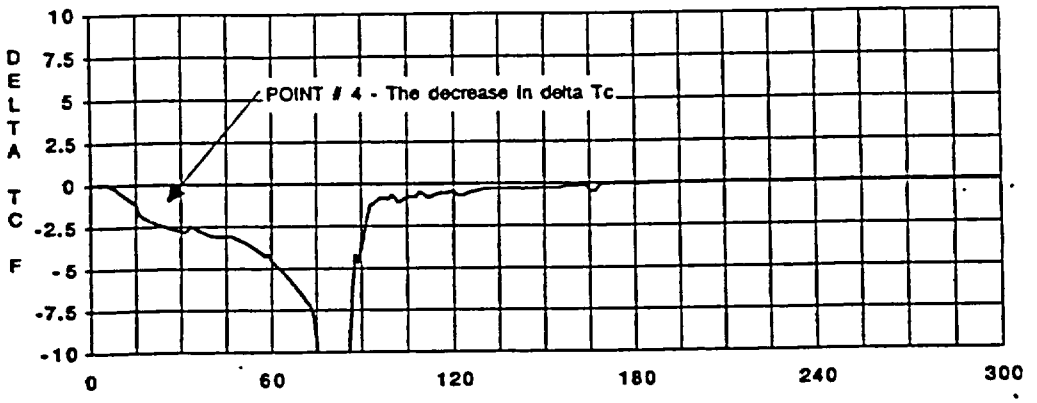

RX DEMAND %
RX POWER %
POINT # 3 - The decrease in reactor demand
0
60
120
180
240
300

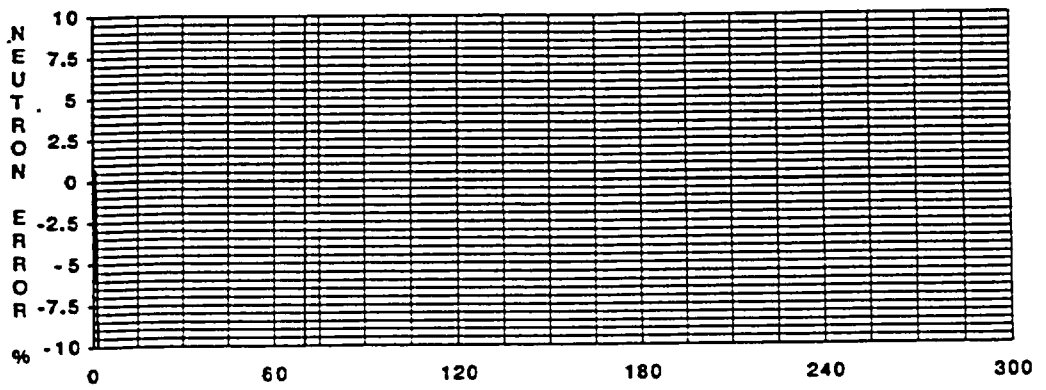

GROUP 7 %
GROUP 6 %
POINT # 5 - The decrease in Gp 7 position
POINT # 8 - The rapid decrease in Gp 6 position
0
60
120
180
240
300

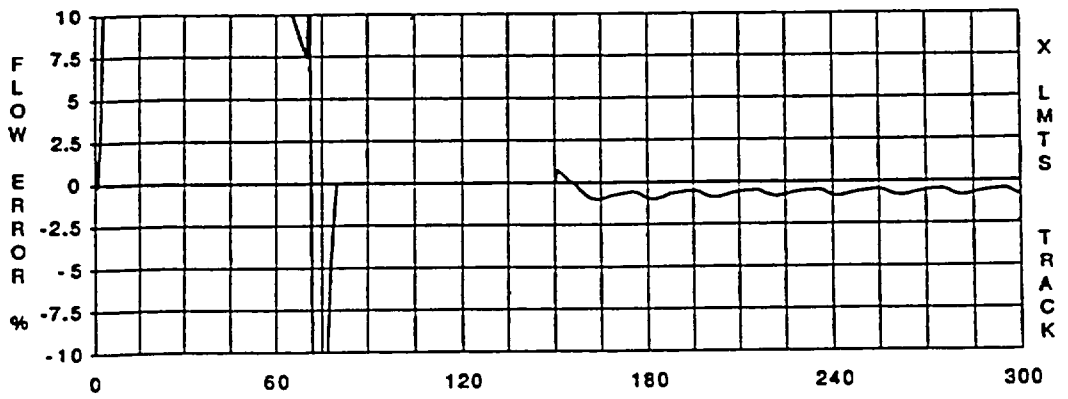

T AVG F
PZR LEVEL in
POINT # 6 - The decrease in Tavg
POINT # 7 - The decrease in pzr level
0
60
120
180
240
300


NR PRESS psig
WR PRESS psig
0
60
120
180
240
300


A RCS FLOW %
B RCS FLOW %
120
100
80
60
40
20
0
0
60
120
180
240
300

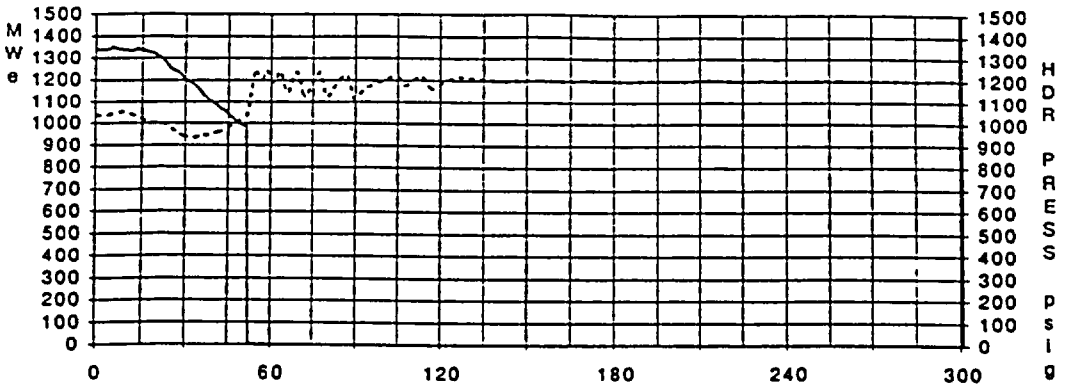

DELTA TC F
POINT # 4 - The decrease in delta Tc
10
7.5
5
2.5
0
-2.5
-5
-7.5
-10
0
60
120
180
240
300

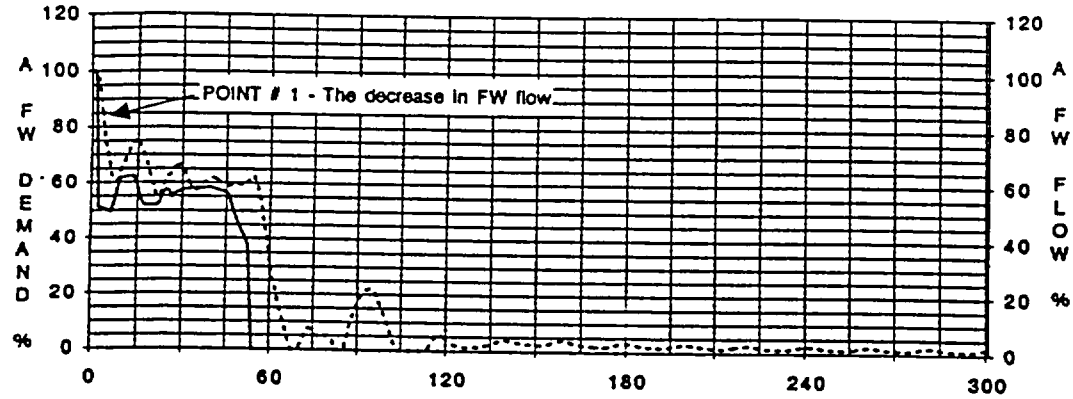

NEUTRON ERROR %
10
7.5
5
2.5
0
-2.5
-5
-7.5
-10
0
60
120
180
240
300

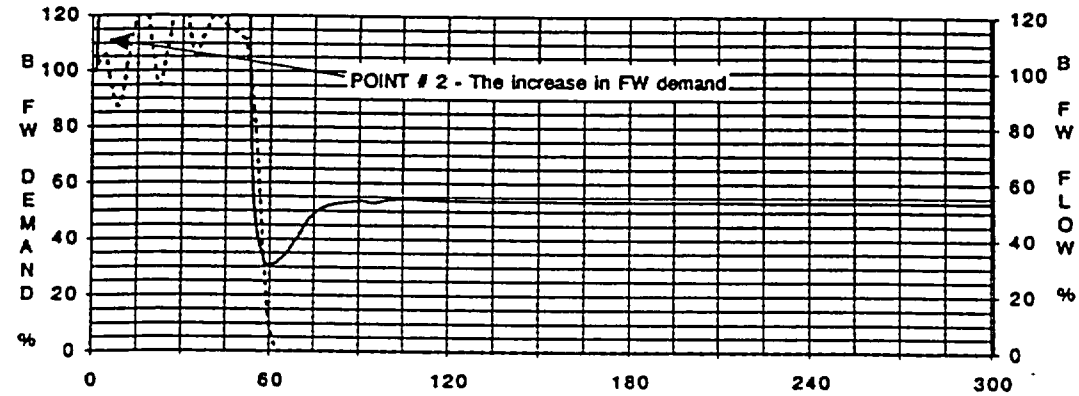

FLOW ERROR %
X LMTS TRACK
10
7.5
5
2.5
0
-2.5
-5
-7.5
-10
0
60
120
180
240
300

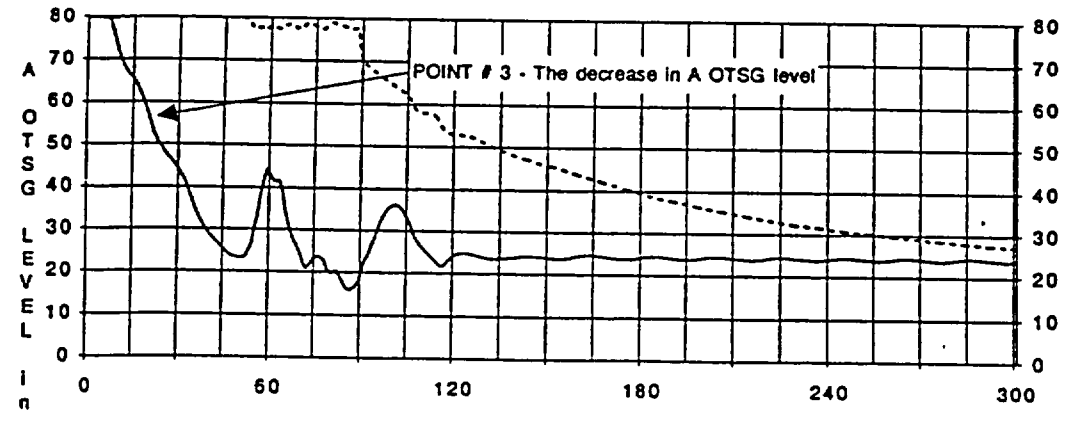
Transient 17

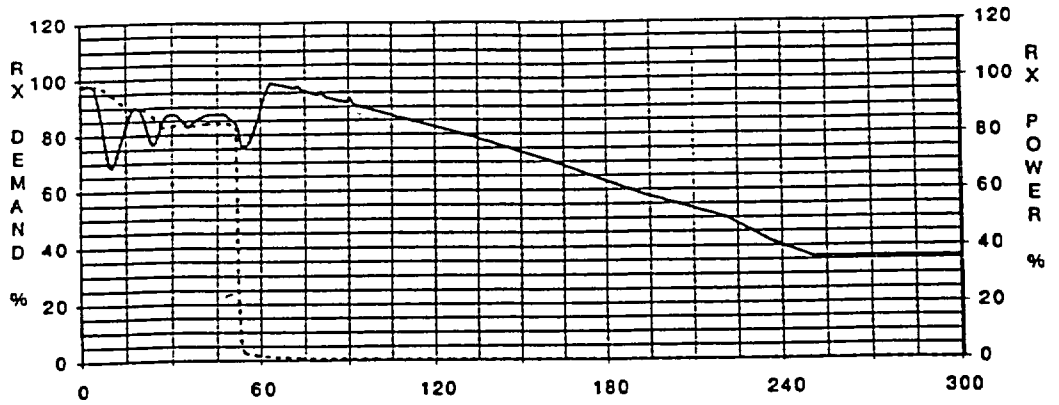
RCS Flow Transmitter Fails Low

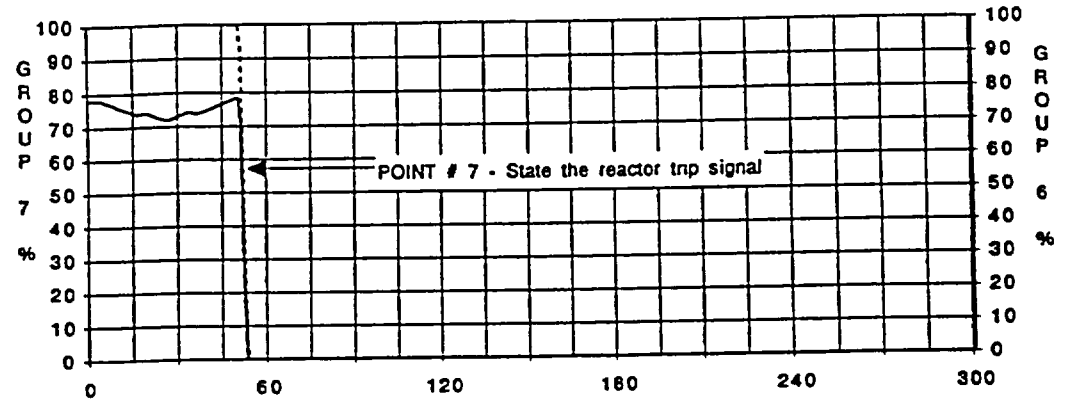

MWe
HDR PRESS psig
0
60
120
180
240
300

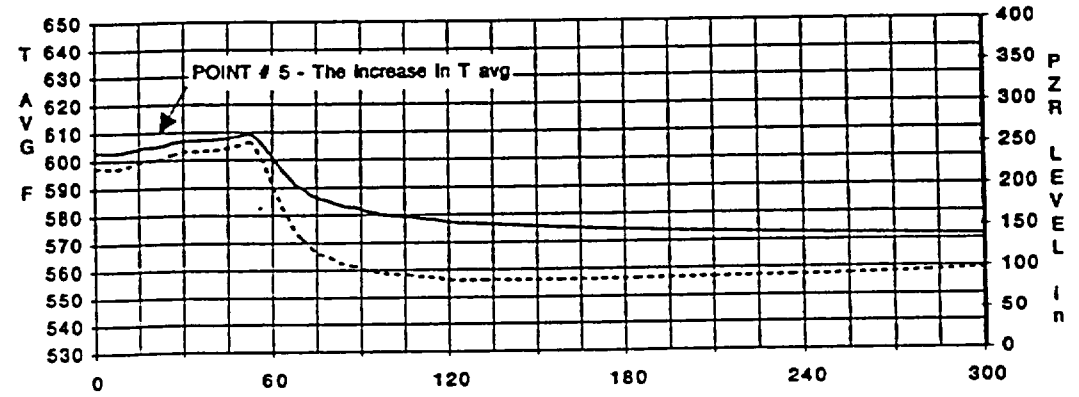

A FW DEMAND %
A FW FLOW %
POINT # 1 - The decrease in FW flow
0
60
120
180
240
300

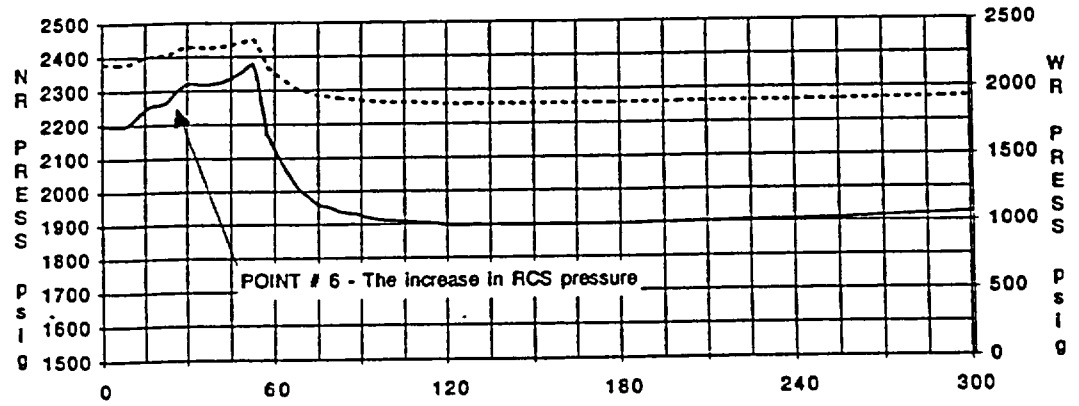

B FW DEMAND %
B FW FLOW %
POINT # 2 - The increase in FW demand
0
60
120
180
240
300

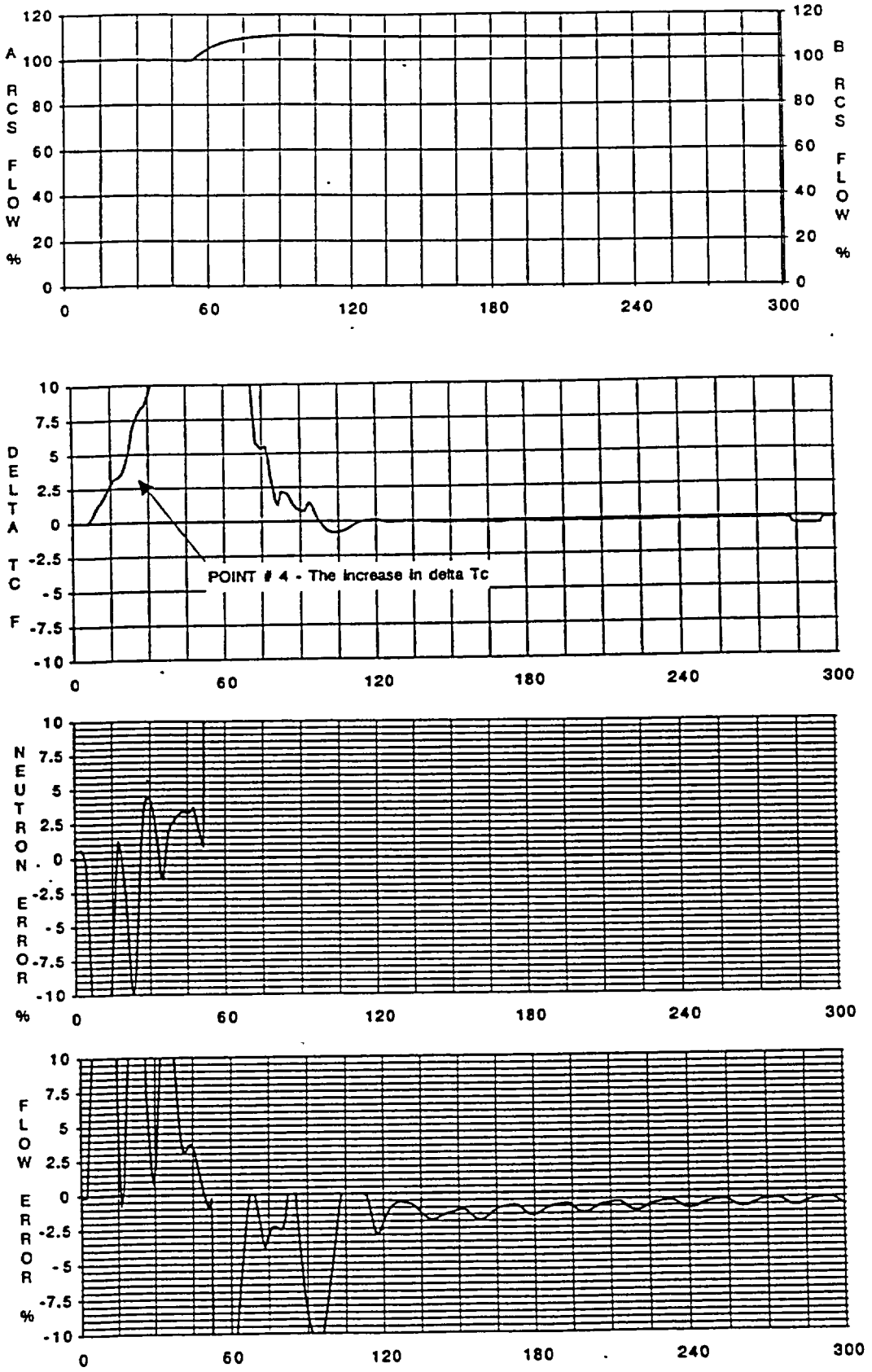

A OTSG LEVEL in
B OTSG LEVEL in
POINT # 3 - The decrease in A OTSG level
0
60
120
180
240
300


RX DEMAND %
RX POWER %
0
60
120
180
240
300


GROUP 7 %
GROUP 6 %
POINT # 7 - State the reactor trip signal
0
60
120
180
240
300

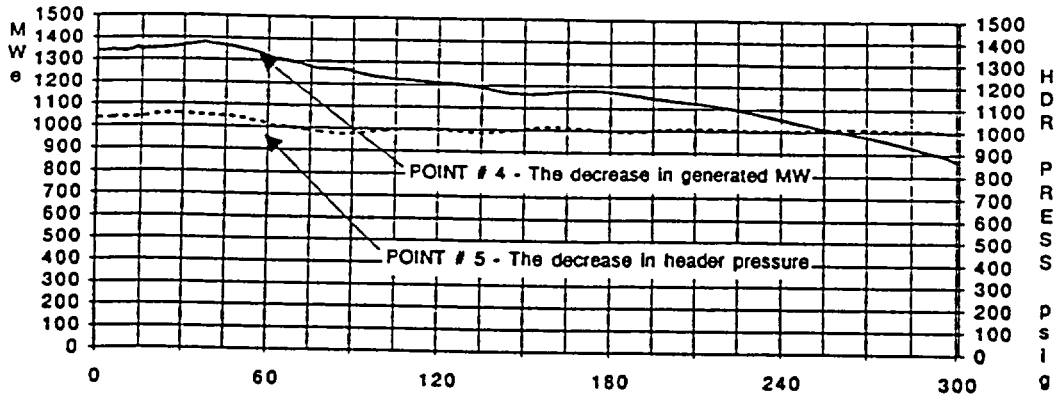

T AVG F
PZR LEVEL in
POINT # 5 - The increase in T avg
0
60
120
180
240
300

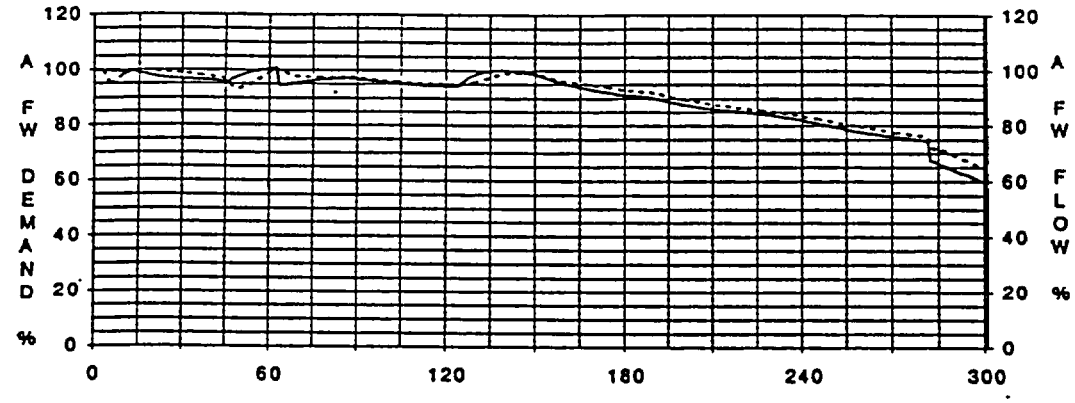

NR PRESS psig
WR PRESS psig
POINT # 6 - The increase in RCS pressure
0
60
120
180
240
300

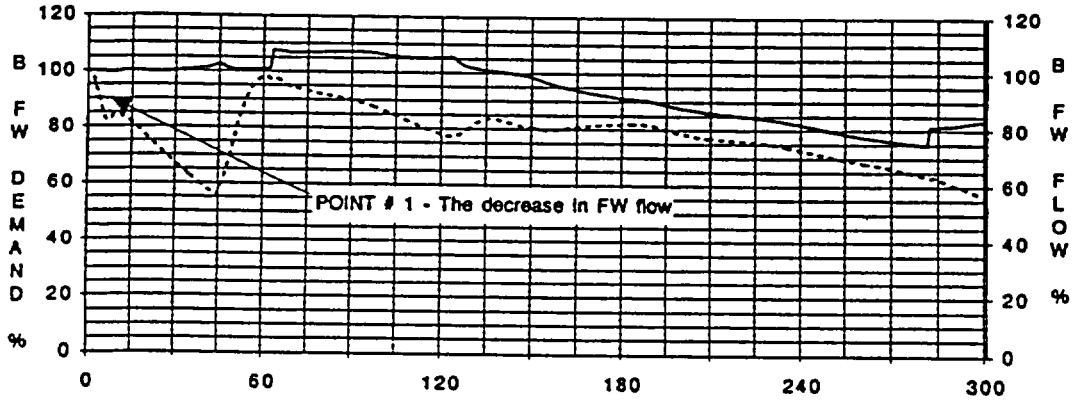

A RCS FLOW %
B RCS FLOW %
DELTA TC F
POINT # 4 - The increase in delta Tc
NEUTRON ERROR %
FLOW ERROR %
0
60
120
180
240
300

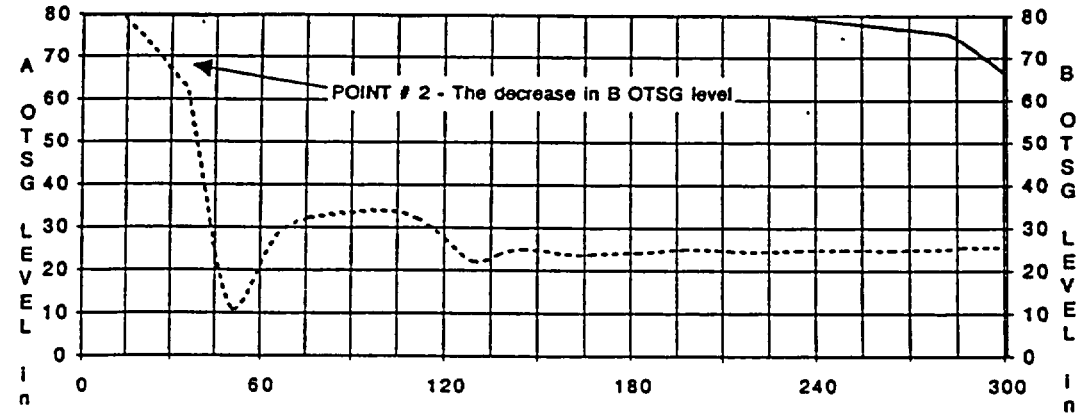
Transient 18

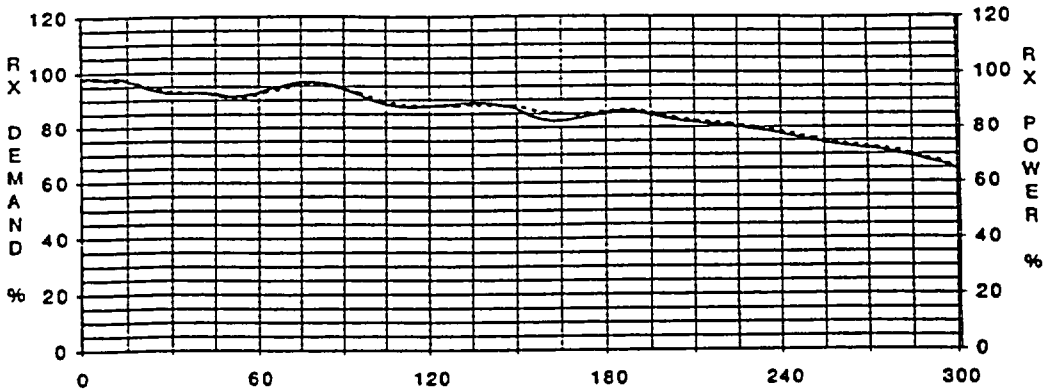
FW Flow Transmitter Fails High

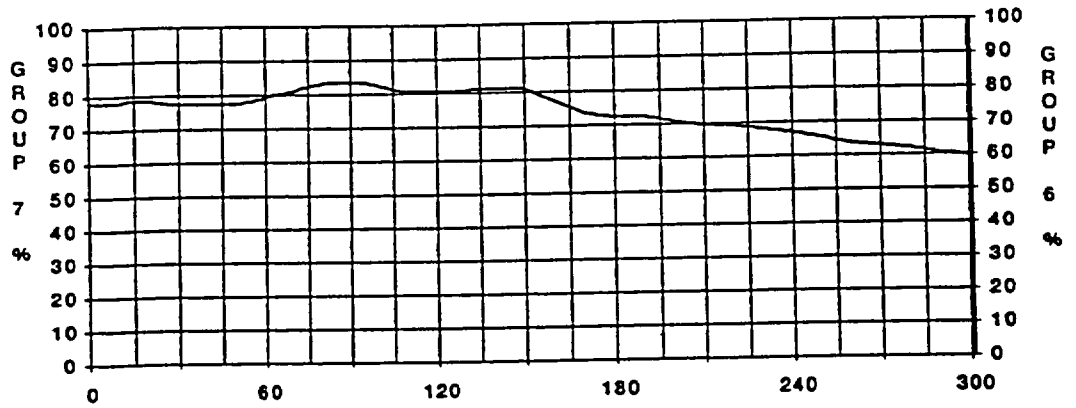

MWe
HDR PRESS psig
POINT # 4 - The decrease in generated MW
POINT # 5 - The decrease in header pressure

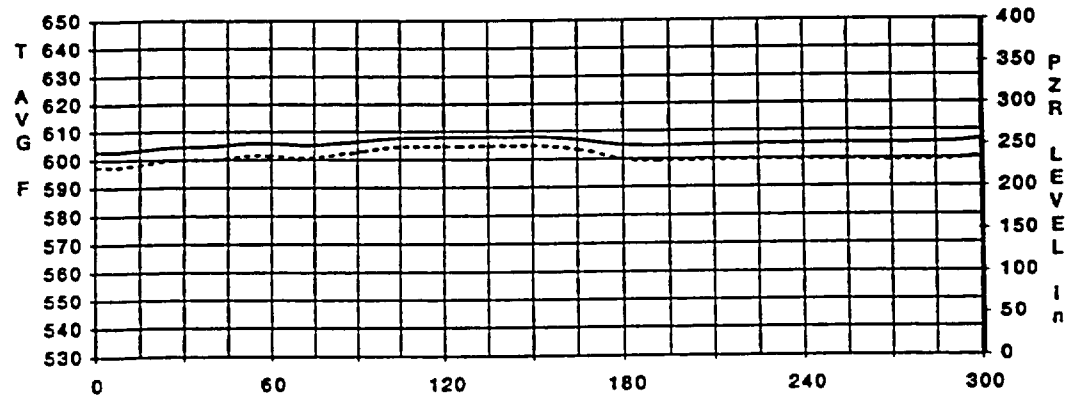

A FW DEMAND %
A FW FLOW %

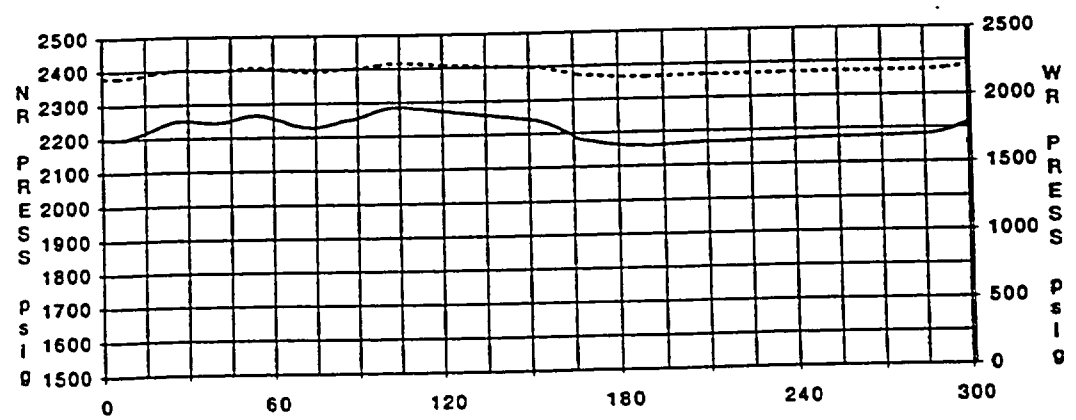

B FW DEMAND %
B FW FLOW %
POINT # 1 - The decrease in FW flow

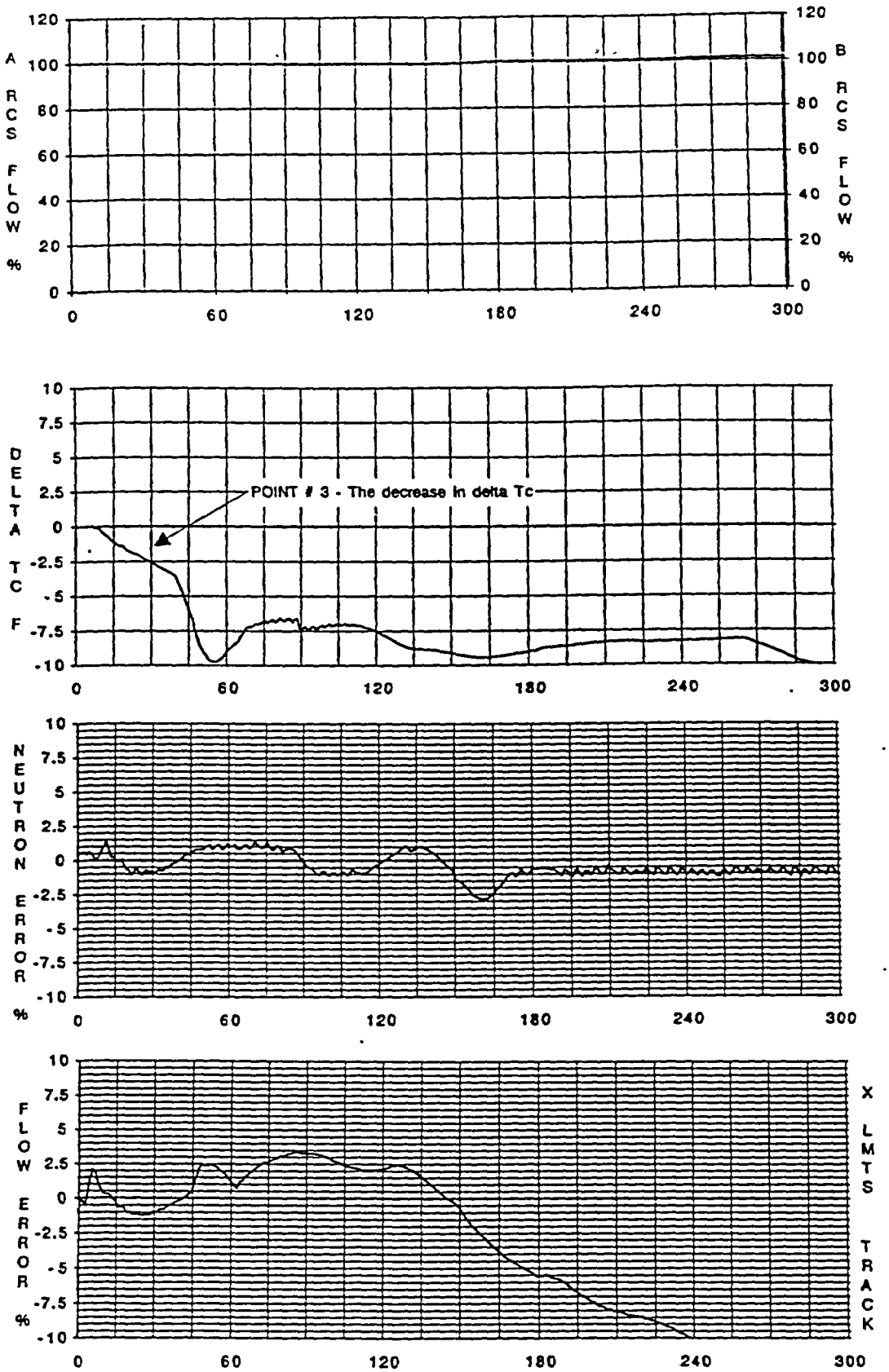

A OTSG LEVEL in
B OTSG LEVEL in
POINT # 2 - The decrease in B OTSG level


RX DEMAND %
RX POWER %
0
20
40
60
80
100
120
0
60
120
180
240
300


GROUP 7 %
GROUP 6 %
0
10
20
30
40
50
60
70
80
90
100
0
60
120
180
240
300

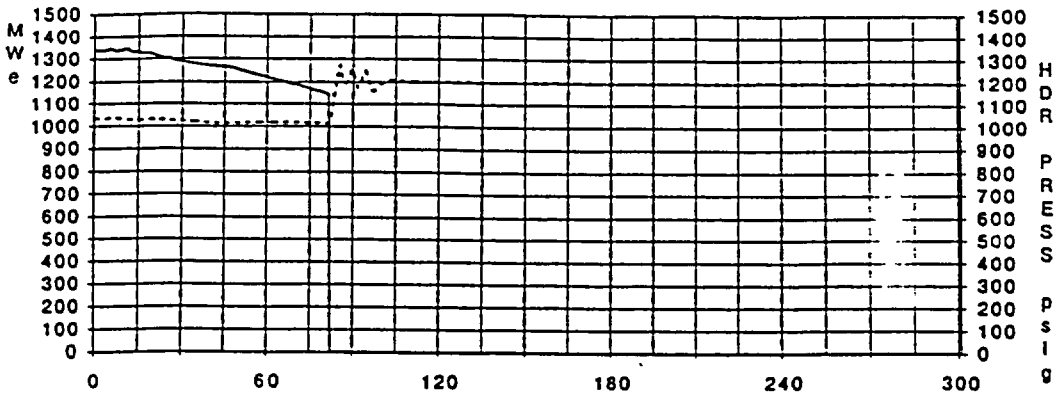

T AVG F
PZR LEVEL in
530
540
550
560
570
580
590
600
610
620
630
640
650
0
50
100
150
200
250
300
350
400
0
60
120
180
240
300

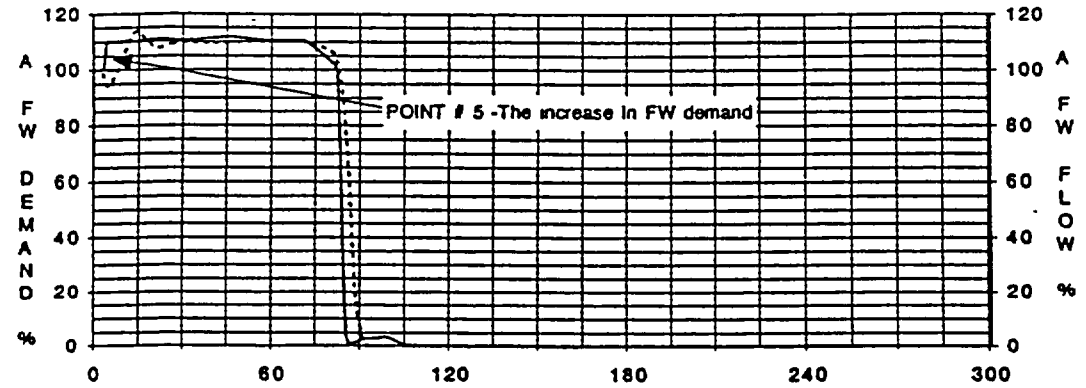

NR PRESS psig
WR PRESS psig
1500
1600
1700
1800
1900
2000
2100
2200
2300
2400
2500
0
500
1000
1500
2000
2500
0
60
120
180
240
300

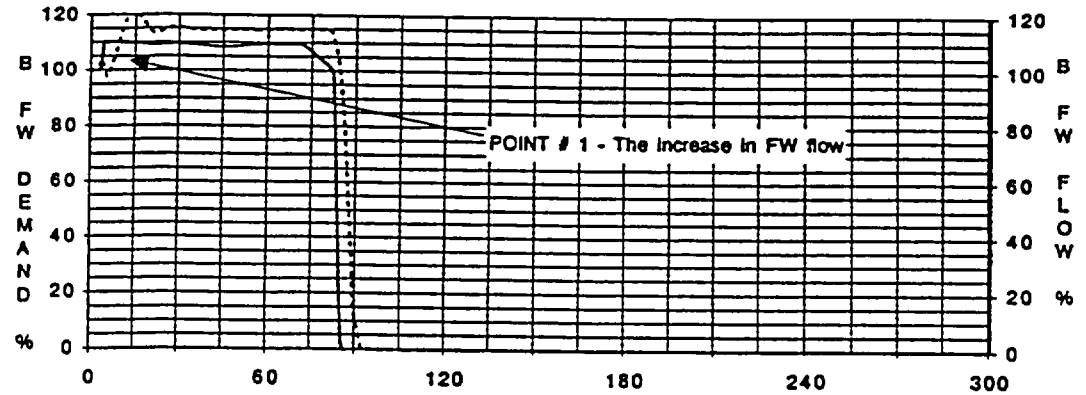

A RCS FLOW %
B RCS FLOW %
DELTA TCF
POINT # 3 - The decrease in delta Tc
NEUTRON ERROR %
FLOW ERROR %
X LMTS TRACK
0
60
120
180
240
300

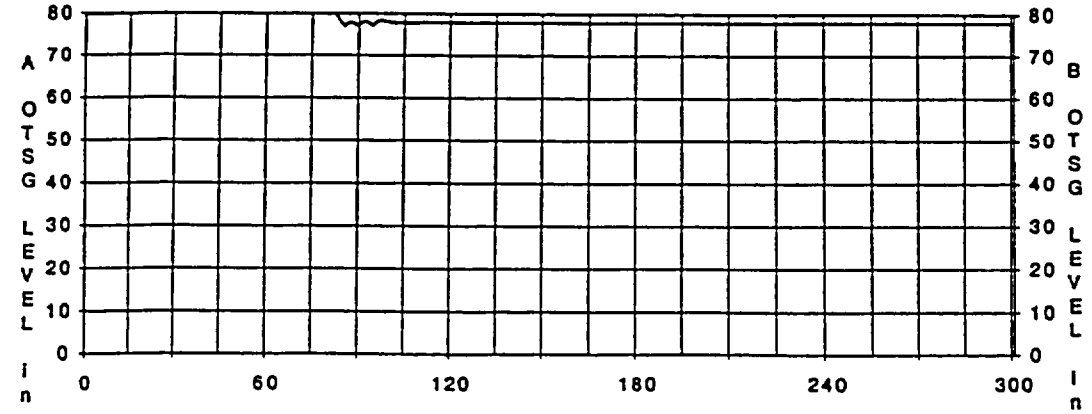
Transient 19

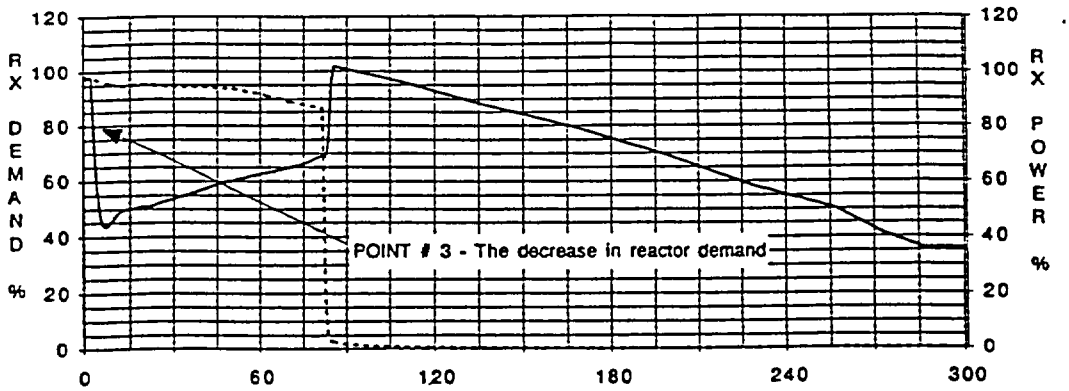
FW Flow Transmitter Fails Low

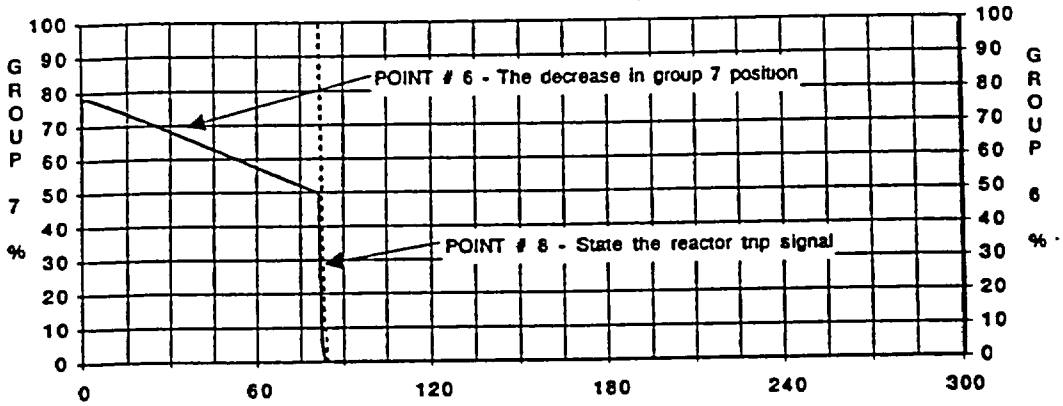

MWe
HDR PRESS psig
1500
1400
1300
1200
1100
1000
900
800
700
600
500
400
300
200
100
0
0
60
120
180
240
300

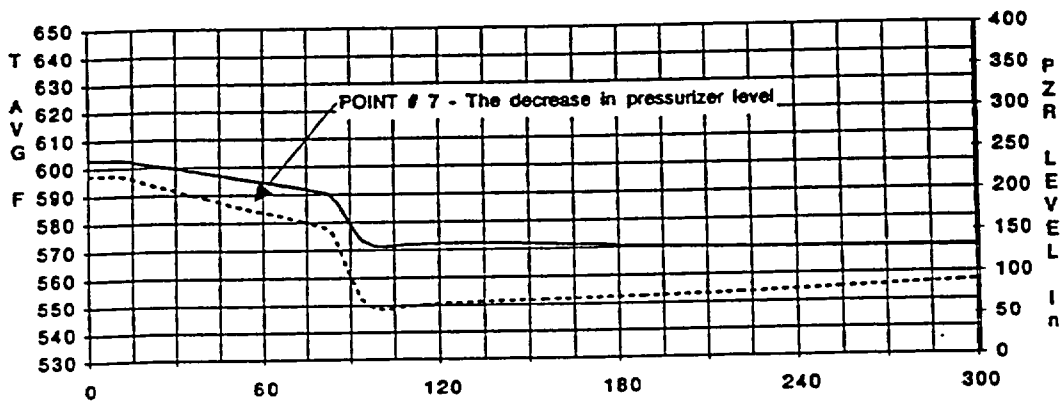

A FW DEMAND %
A FW FLOW %
POINT # 5 -The increase in FW demand
120
100
80
60
40
20
0
0
60
120
180
240
300

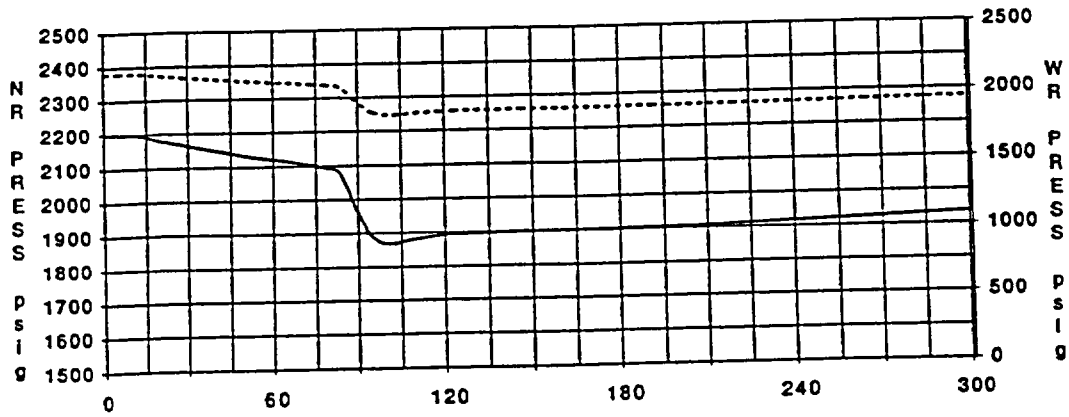

B FW DEMAND %
B FW FLOW %
POINT # 1 - The increase in FW flow
120
100
80
60
40
20
0
0
60
120
180
240
300

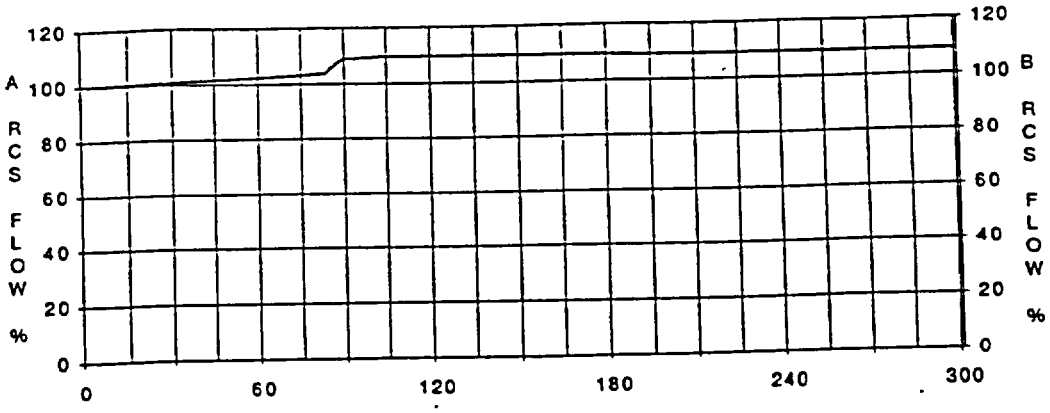

A OTSG LEVEL in
B OTSG LEVEL in
80
70
60
50
40
30
20
10
0
0
60
120
180
240
300

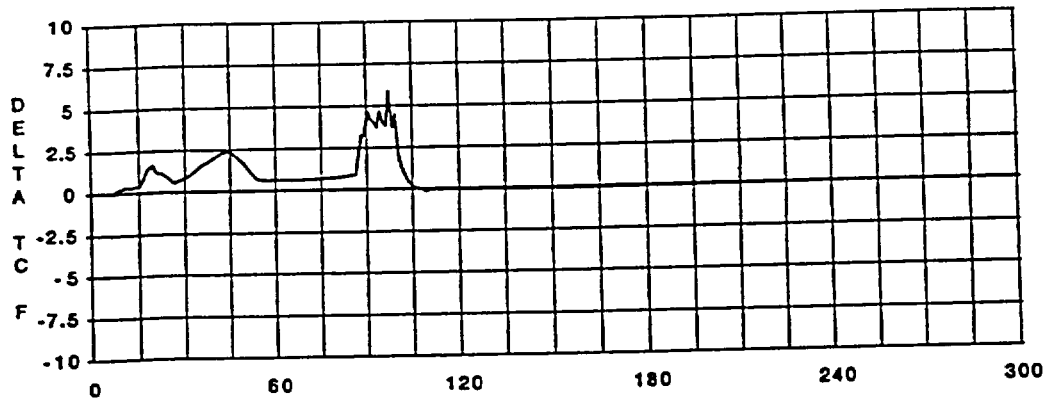

RX DEMAND %
RX POWER %
POINT # 3 - The decrease in reactor demand
0
60
120
180
240
300

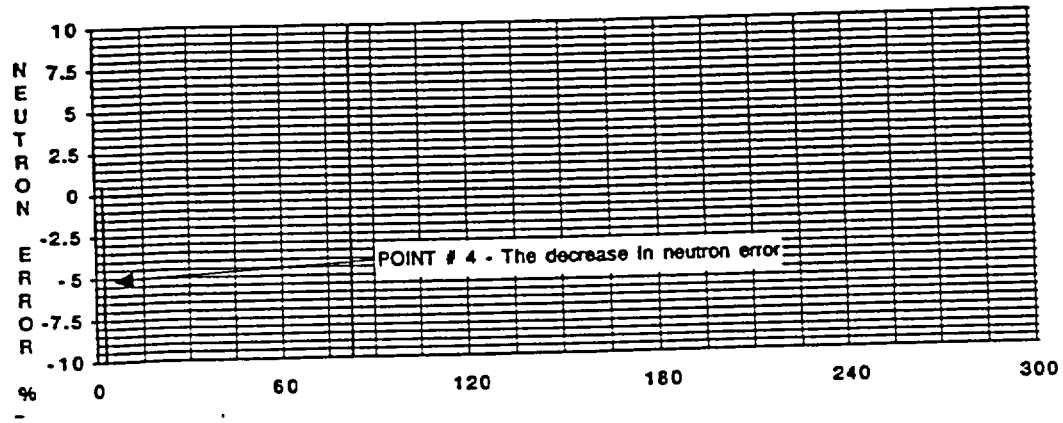

GROUP 7 %
GROUP 6 %
POINT # 6 - The decrease in group 7 position
POINT # 8 - State the reactor trip signal
0
60
120
180
240
300

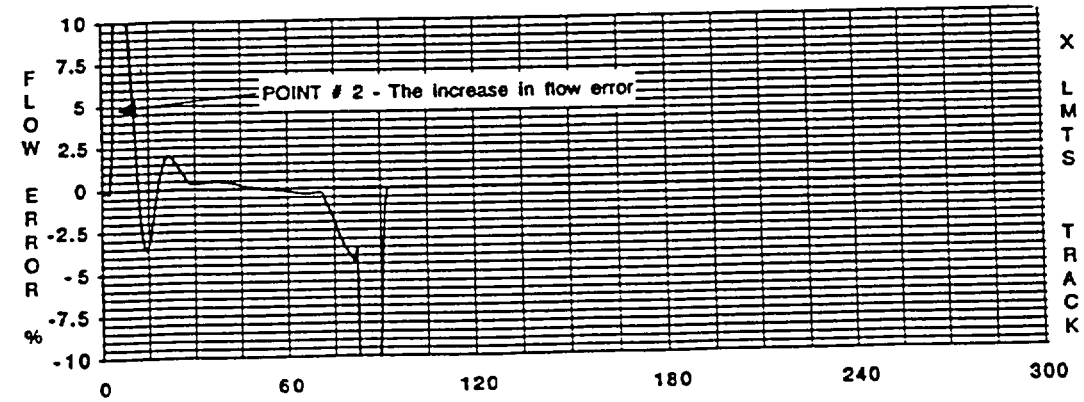

T AVG F
PZR LEVEL in
POINT # 7 - The decrease in pressurizer level
0
60
120
180
240
300


NR PRESS psig
WR PRESS psig
0
60
120
180
240
300


A RCS FLOW %
B RCS FLOW %
120
100
80
60
40
20
0
0
60
120
180
240
300

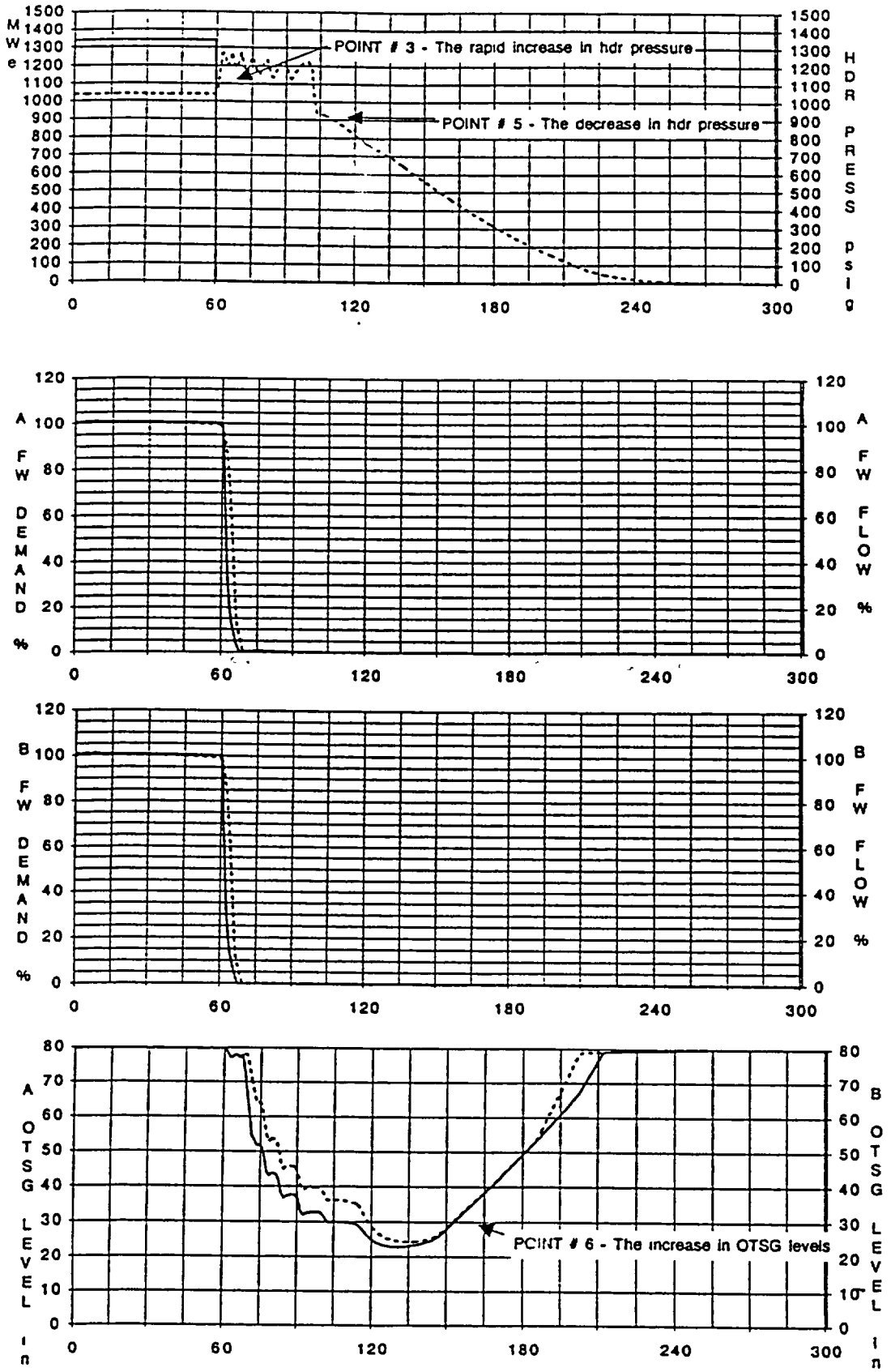

DELTA TCF
10
7.5
5
2.5
0
-2.5
-5
-7.5
-10
0
60
120
180
240
300

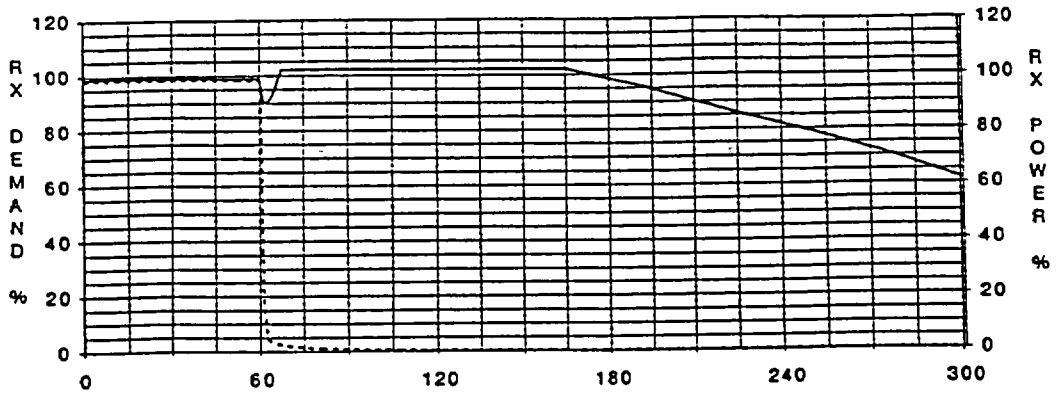

NEUTRON ERROR %
POINT # 4 - The decrease in neutron error
10
7.5
5
2.5
0
-2.5
-5
-7.5
-10
0
60
120
180
240
300

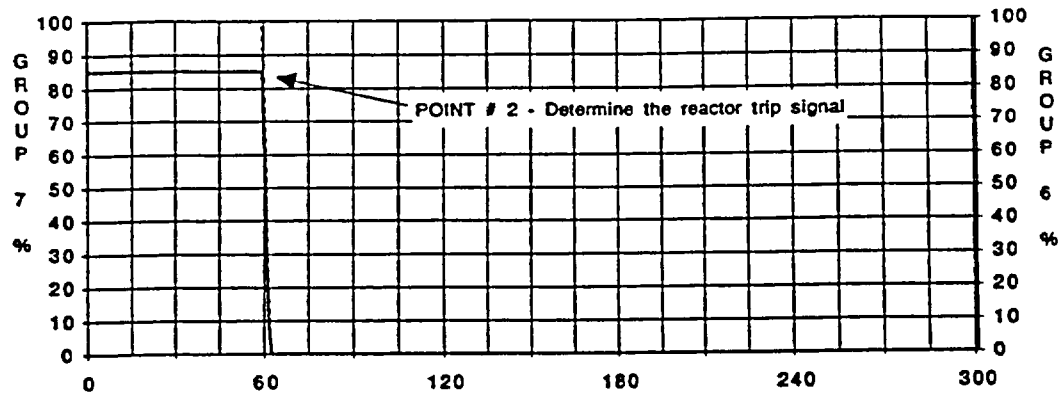

FLOW ERROR %
POINT # 2 - The increase in flow error
X LMTS TRACK
10
7.5
5
2.5
0
-2.5
-5
-7.5
-10
0
60
120
180
240
300

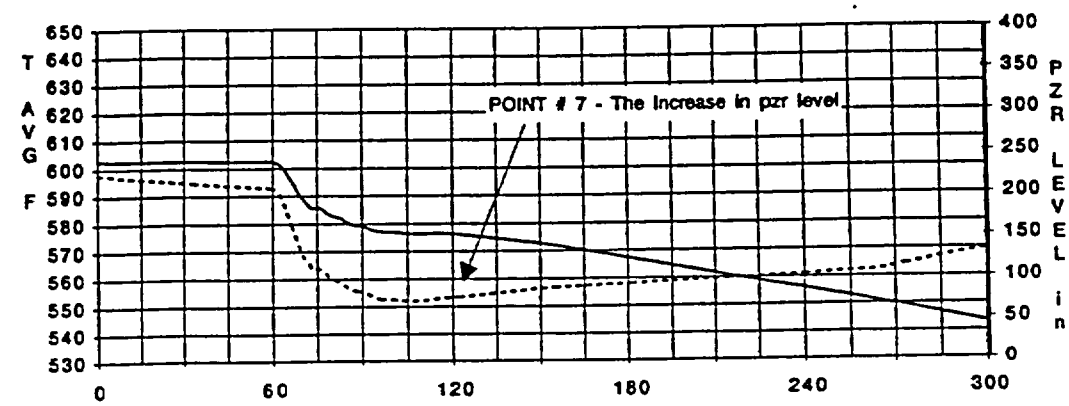
Transient 20

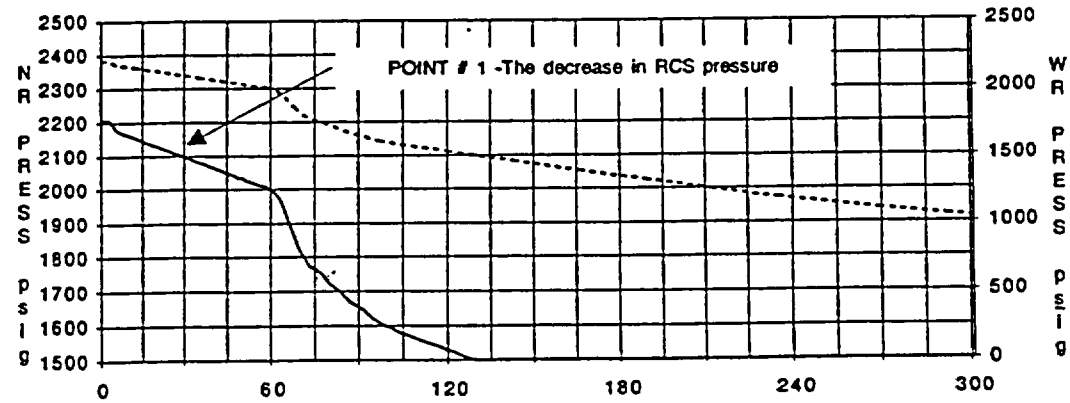
Pressurizer Pressure Transmitter Fails High

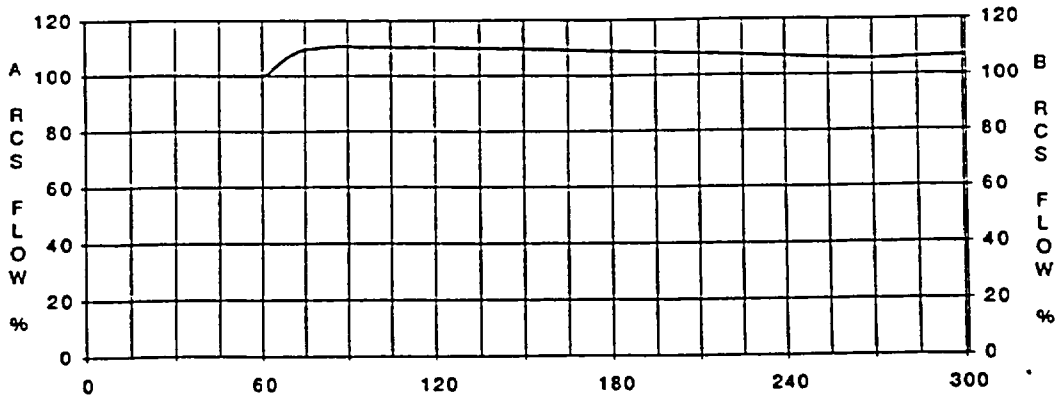

MWe
HDR PRESS psig
POINT # 3 - The rapid increase in hdr pressure
POINT # 5 - The decrease in hdr pressure
A FW DEMAND %
A FW FLOW %
B FW DEMAND %
B FW FLOW %
A OTSG LEVEL in
B OTSG LEVEL in
PCINT # 6 - The increase in OTSG levels

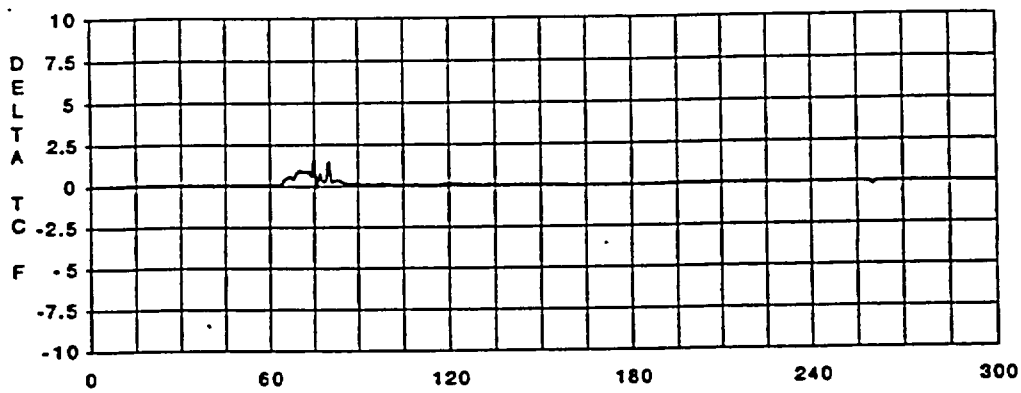

RX DEMAND %
RX POWER %
0
20
40
60
80
100
120
0
60
120
180
240
300

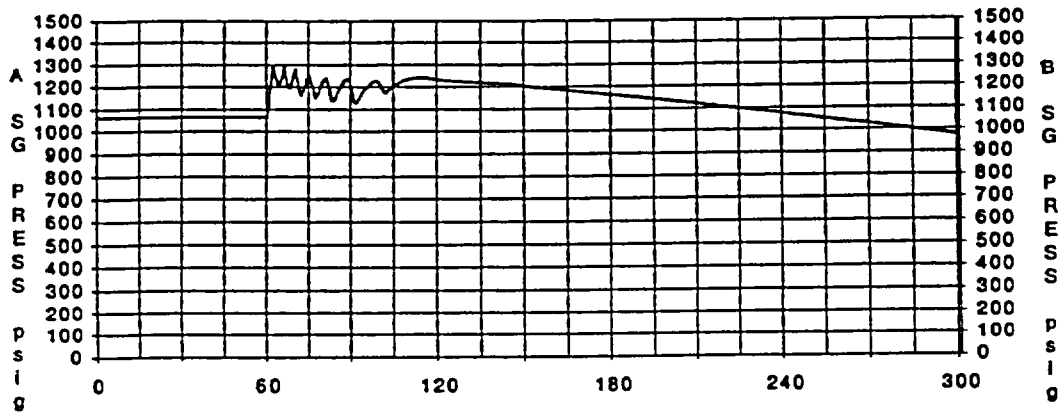

GROUP 7 %
GROUP 6 %
POINT # 2 - Determine the reactor trip signal
0
10
20
30
40
50
60
70
80
90
100
0
60
120
180
240
300

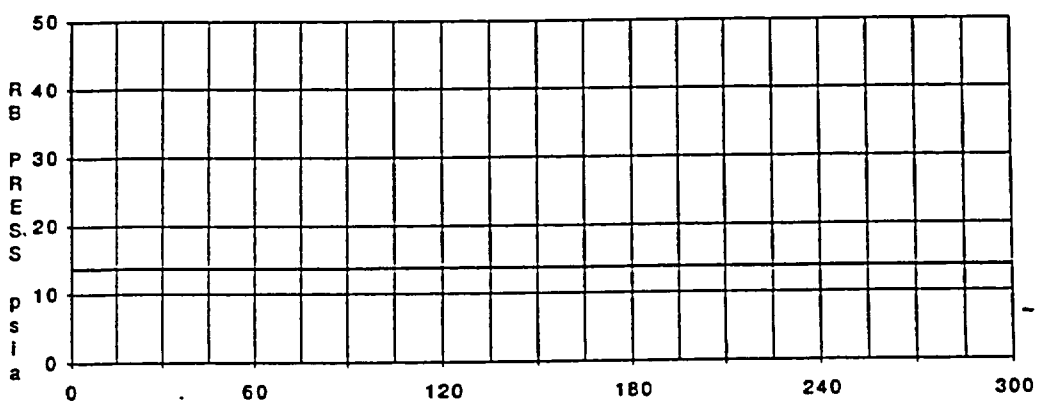

T AVG F
PZR LEVEL in
POINT # 7 - The Increase in pzr level
530
540
550
560
570
580
590
600
610
620
630
640
650
0
50
100
150
200
250
300
350
400
0
60
120
180
240
300

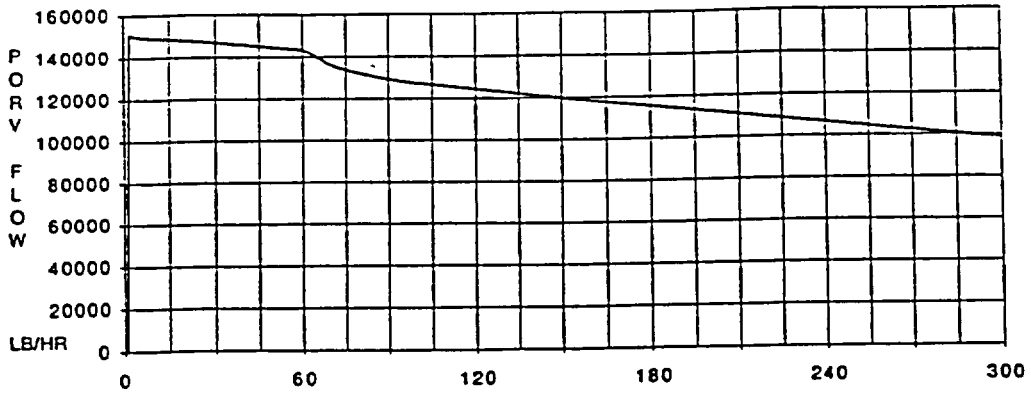

NR PRESS psig
WR PRESS psig
POINT # 1 -The decrease in RCS pressure
1500
1600
1700
1800
1900
2000
2100
2200
2300
2400
2500
0
500
1000
1500
2000
2500
0
60
120
180
240
300

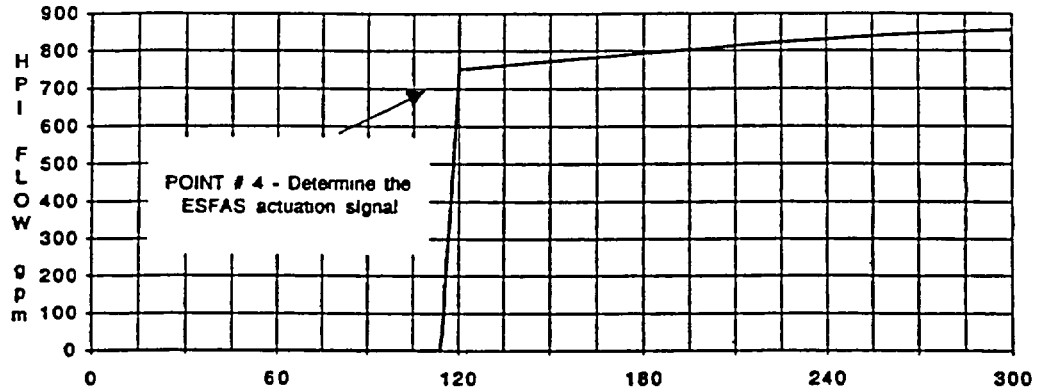

A RCS FLOW %
B RCS FLOW %
120
100
80
60
40
20
0
0
60
120
180
240
300

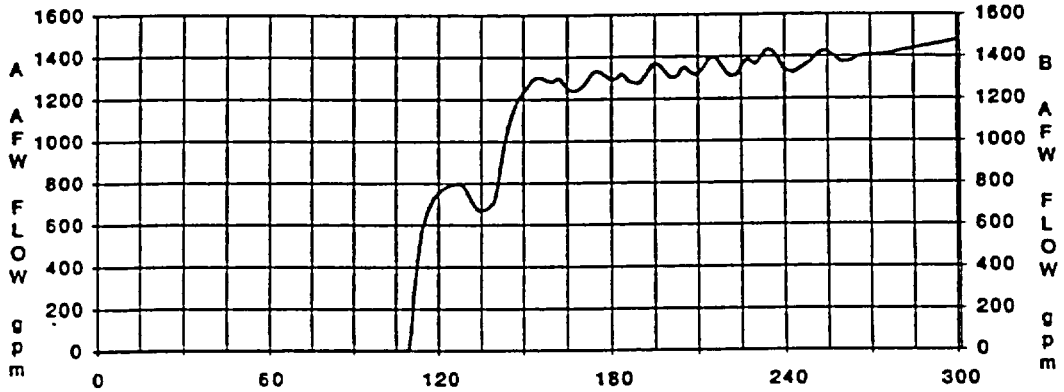

DELTA TCF
10
7.5
5
2.5
0
-2.5
-5
-7.5
-10
0
60
120
180
240
300


A SG PRESS psig
B SG PRESS psig
1500
1400
1300
1200
1100
1000
900
800
700
600
500
400
300
200
100
0
0
60
120
180
240
300


RB PRESS. psia
50
40
30
20
10
0
0
60
120
180
240
300


PORV FLOW
LB/HR
160000
140000
120000
100000
80000
60000
40000
20000
0
0
60
120
180
240
300


HPI FLOW gpm
900
800
700
600
500
400
300
200
100
0
POINT # 4 - Determine the ESFAS actuation signal
0
60
120
180
240
300


A AFW FLOW gpm
B AFW FLOW gpm
1600
1400
1200
1000
800
600
400
200
0
0
60
120
180
240
300

Transient 21

Classroom Exercise

TRANSIENT 21

Assume that the selected pressurizer level transmitter fails high, and provide answers to the following questions:

1. As a result of this input failure, what will be the position of the makeup flow control valve? (see Fig. 3.1)

2. What will be the rate of change of pressurizer level?

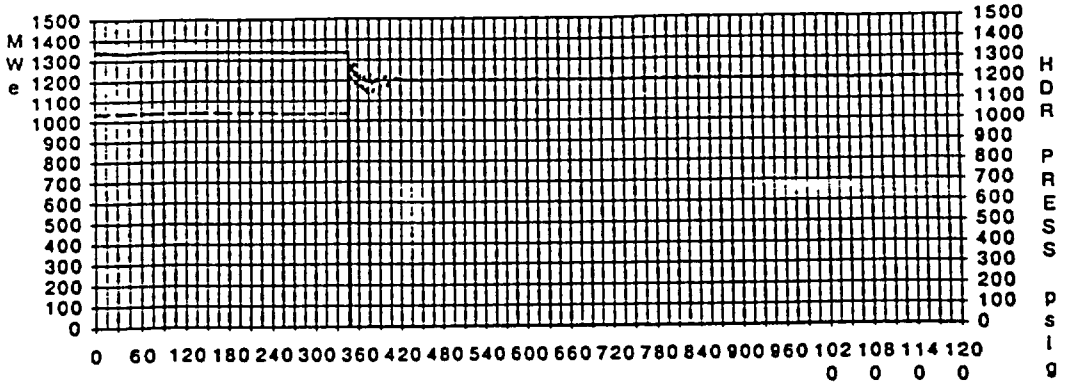
3. What will be the rate of change of makeup tank level?

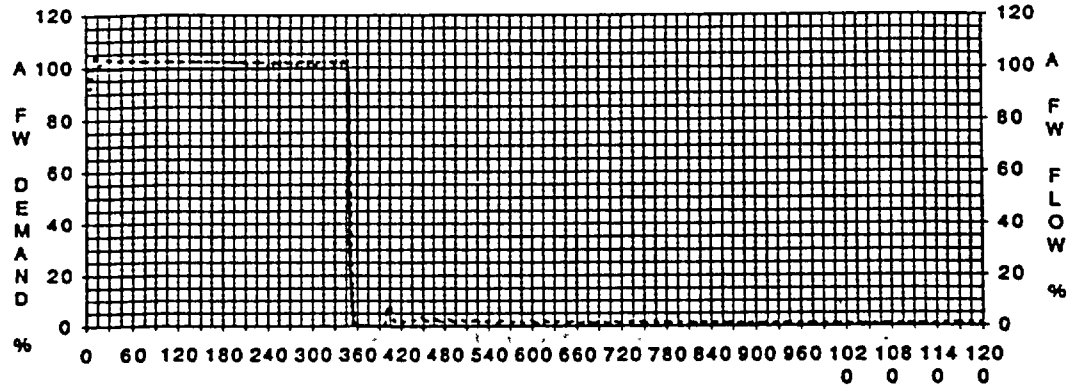
4. How many groups of pressurizer heaters will be energized?

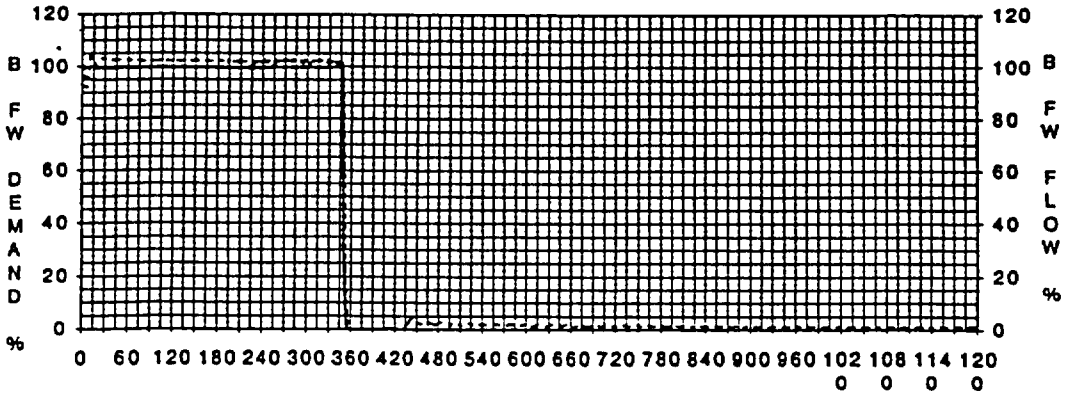
5. Will the reactor trip? If yes, what will be the cause of the trip?

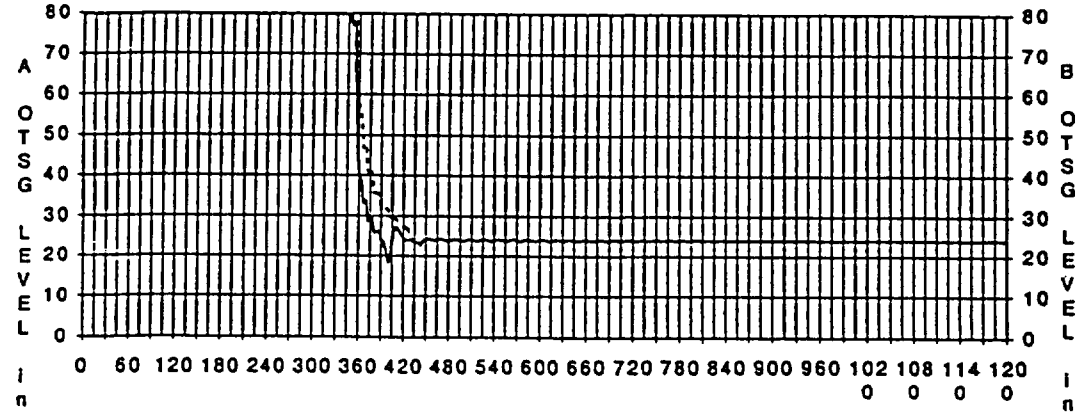
Transient 22

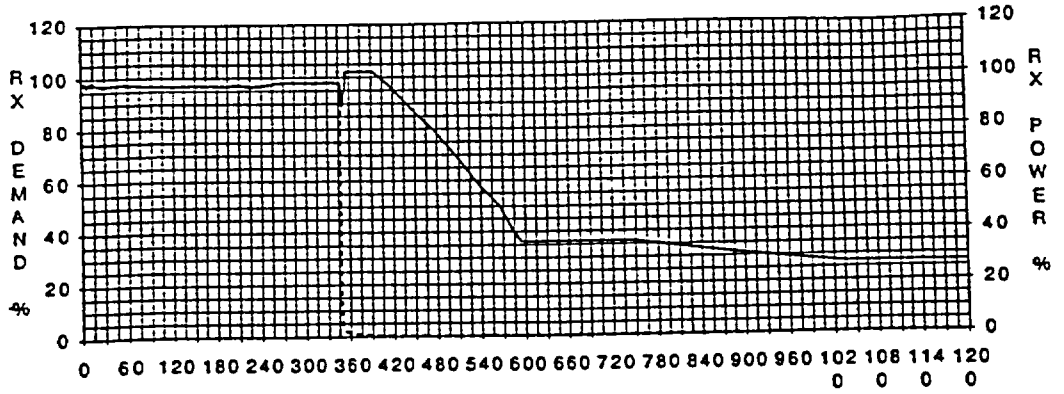
Pressurizer Level Fails Low

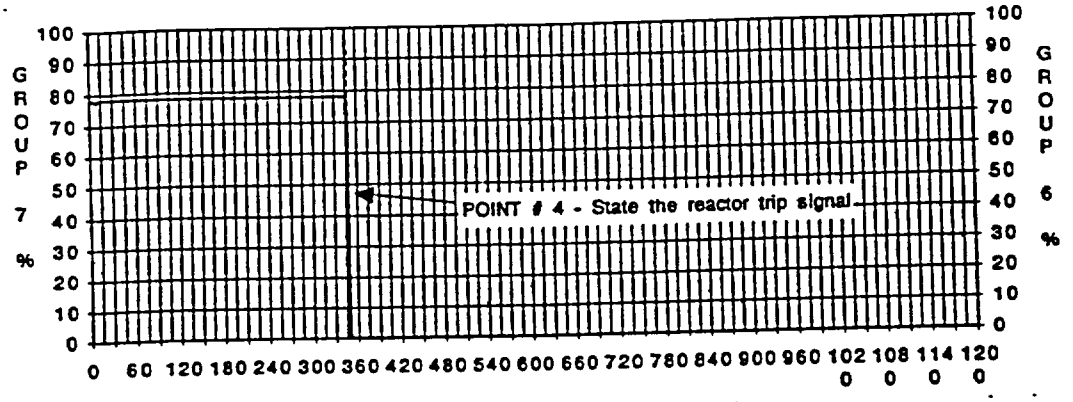

MWe
HDR PRESS psig
0 60 120 180 240 300 360 420 480 540 600 660 720 780 840 900 960 1020 1080 1140 1200

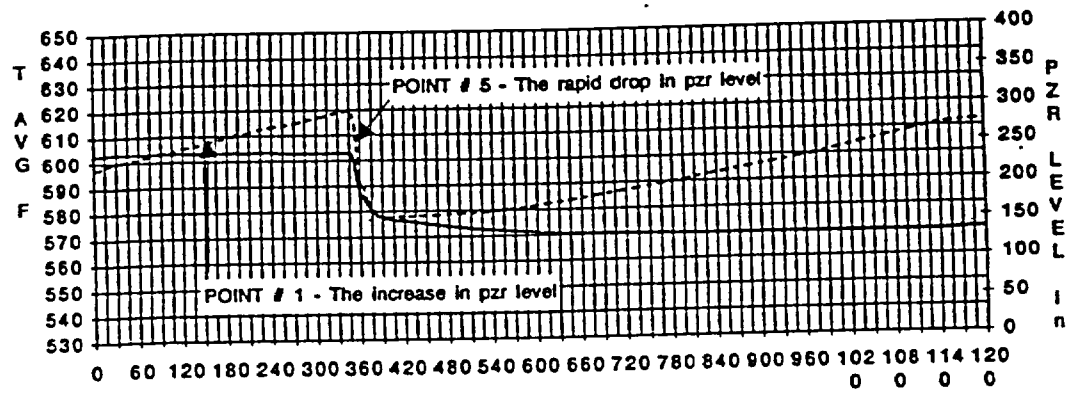

A FW DEMAND %
A FW FLOW %
0 60 120 180 240 300 360 420 480 540 600 660 720 780 840 900 960 1020 1080 1140 1200

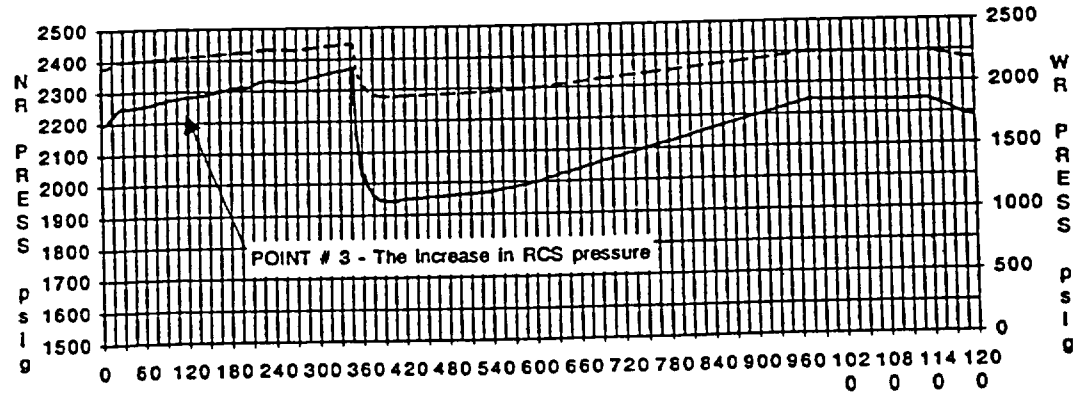

B FW DEMAND %
B FW FLOW %
0 60 120 180 240 300 360 420 480 540 600 660 720 780 840 900 960 1020 1080 1140 1200

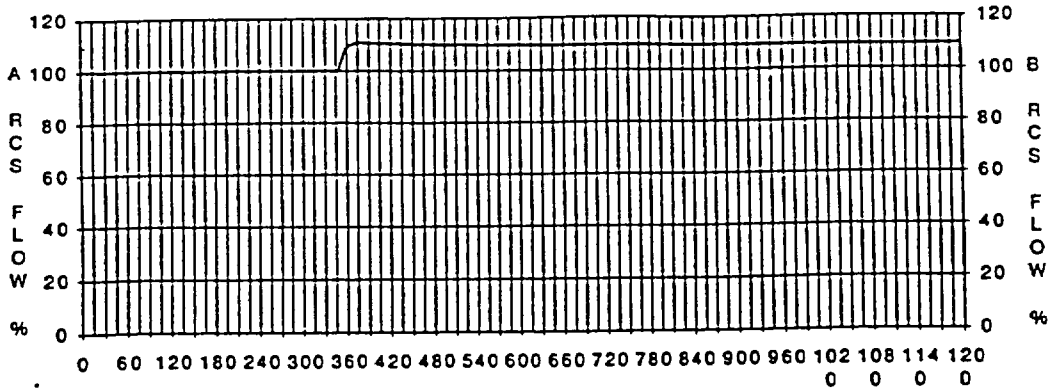

A OTSG LEVEL in
B OTSG LEVEL in
0 60 120 180 240 300 360 420 480 540 600 660 720 780 840 900 960 1020 1080 1140 1200

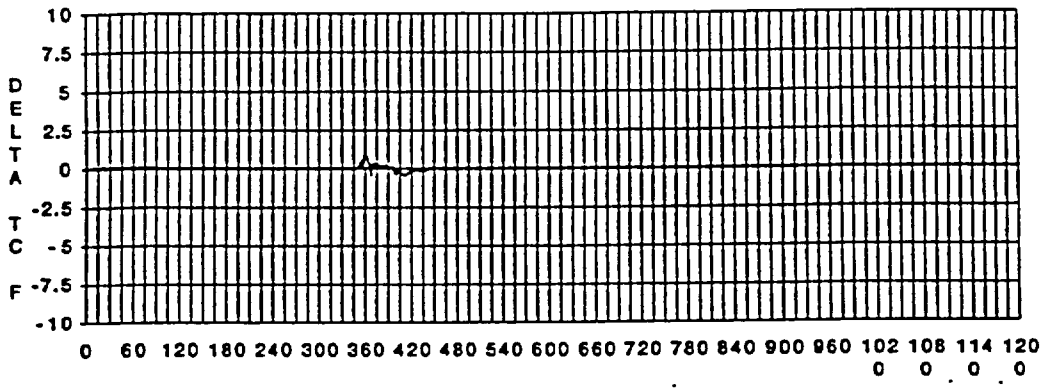

RX DEMAND %
RX POWER %
0 60 120 180 240 300 360 420 480 540 600 660 720 780 840 900 960 1020 1080 1140 1200

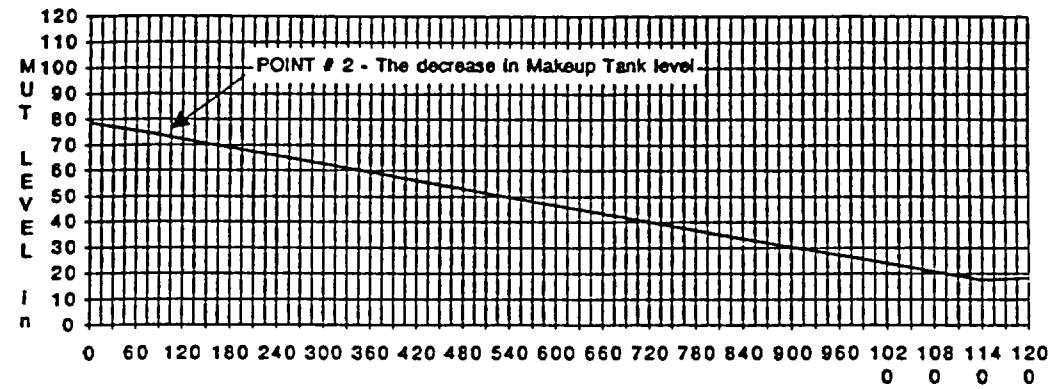

GROUP 7 %
GROUP 6 %
POINT # 4 - State the reactor trip signal
0 60 120 180 240 300 360 420 480 540 600 660 720 780 840 900 960 1020 1080 1140 1200


T AVG F
PZR LEVEL in
POINT # 5 - The rapid drop in pzr level
POINT # 1 - The increase in pzr level
0 60 120 180 240 300 360 420 480 540 600 660 720 780 840 900 960 1020 1080 1140 1200


NR PRESS psig
WR PRESS psig
POINT # 3 - The increase in RCS pressure
0 60 120 180 240 300 360 420 480 540 600 660 720 780 840 900 960 1020 1080 1140 1200

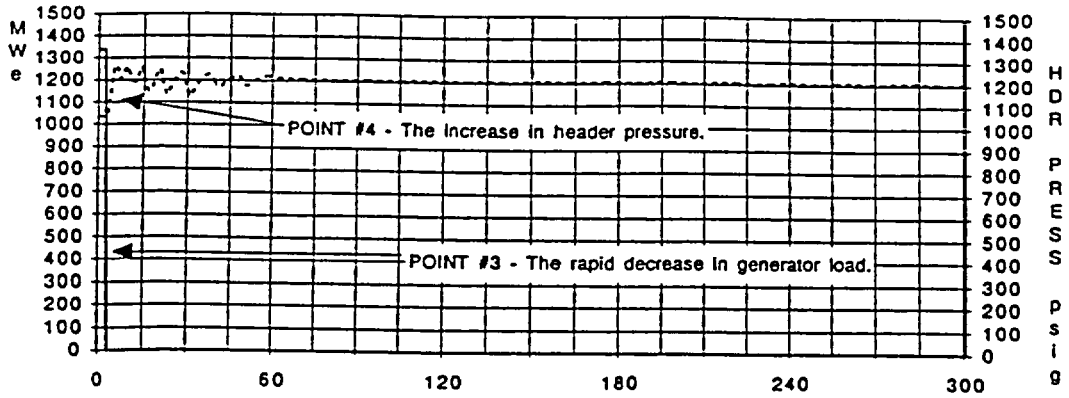

A RCS FLOW %
B RCS FLOW %
120
100
80
60
40
20
0
0 60 120 180 240 300 360 420 480 540 600 660 720 780 840 900 960 1020 1080 1140 1200

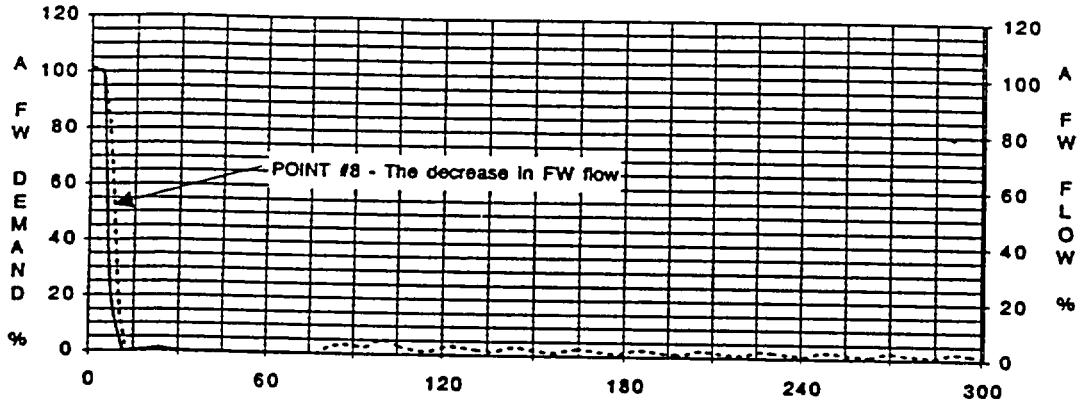

DELTA TC F
10
7.5
5
2.5
0
-2.5
-5
-7.5
-10
0 60 120 180 240 300 360 420 480 540 600 660 720 780 840 900 960 1020 1080 1140 1200

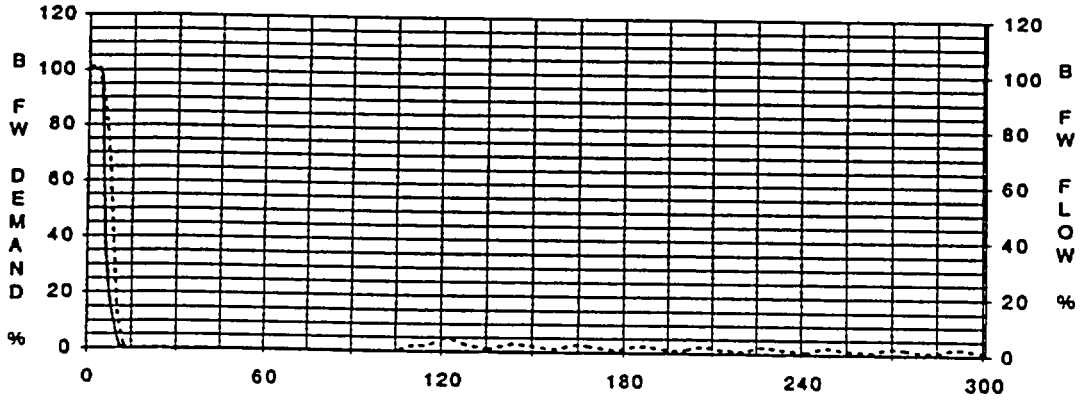

MUT LEVEL in
POINT # 2 - The decrease in Makeup Tank level
120
110
100
90
80
70
60
50
40
30
20
10
0
0 60 120 180 240 300 360 420 480 540 600 660 720 780 840 900 960 1020 1080 1140 1200

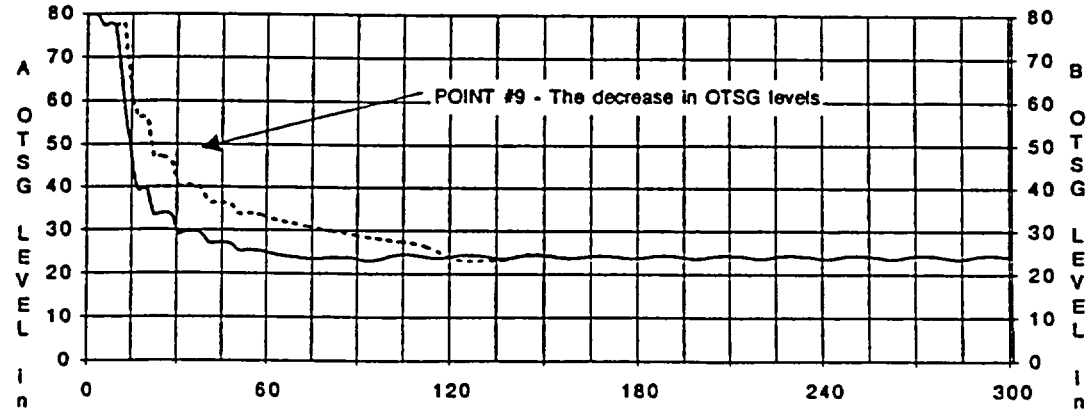
Transient 23

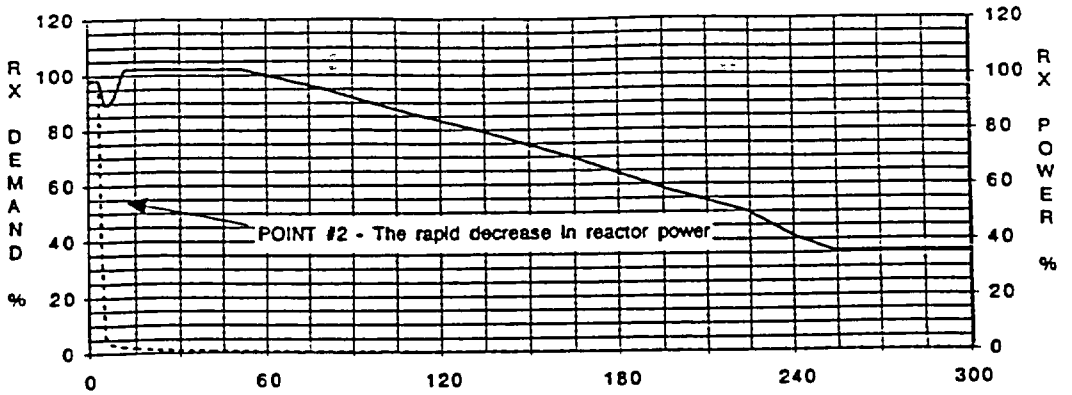
Manual Reactor Trip

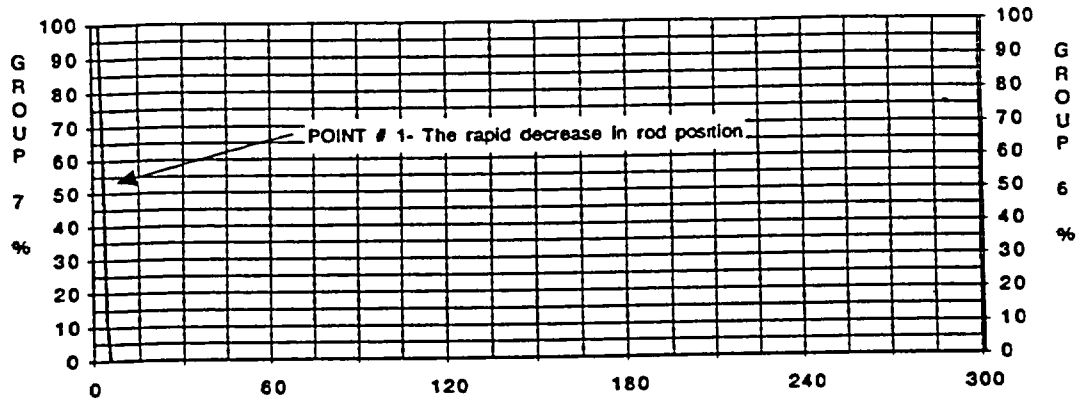

MWe
HDR PRESS psig
POINT #4 - The increase in header pressure.
POINT #3 - The rapid decrease in generator load.

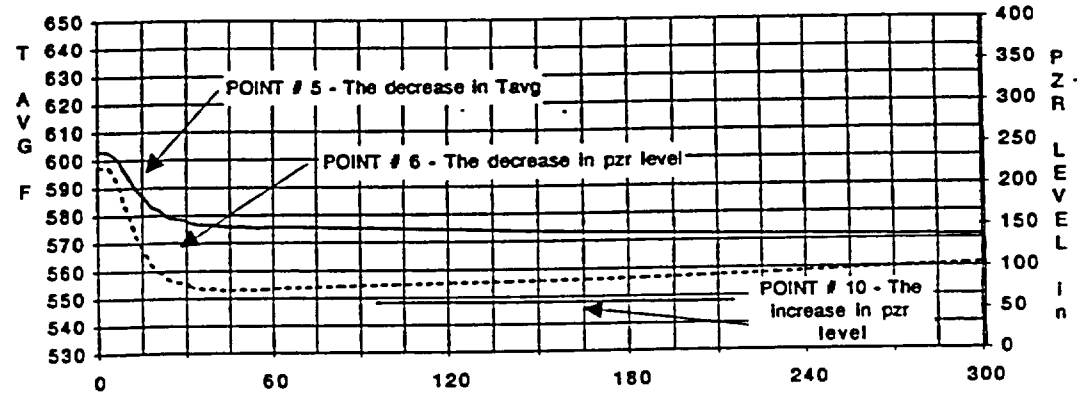

A FW DEMAND %
A FW FLOW %
POINT #8 - The decrease in FW flow

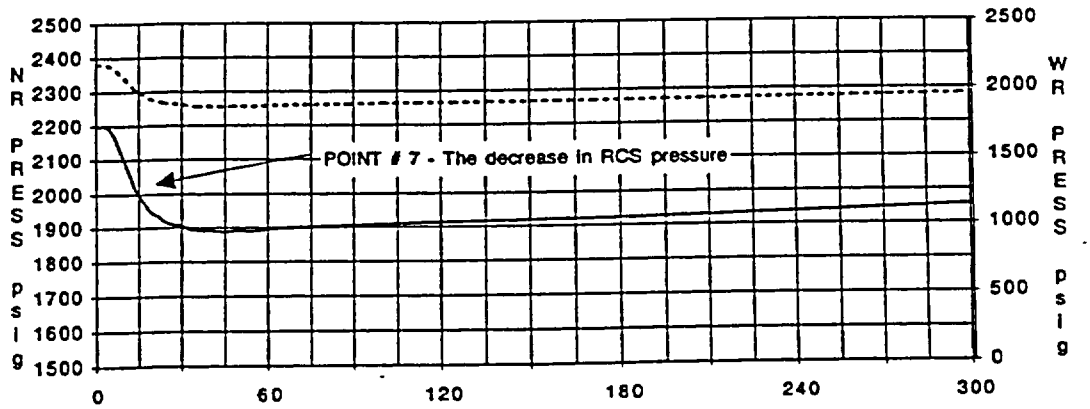

B FW DEMAND %
B FW FLOW %

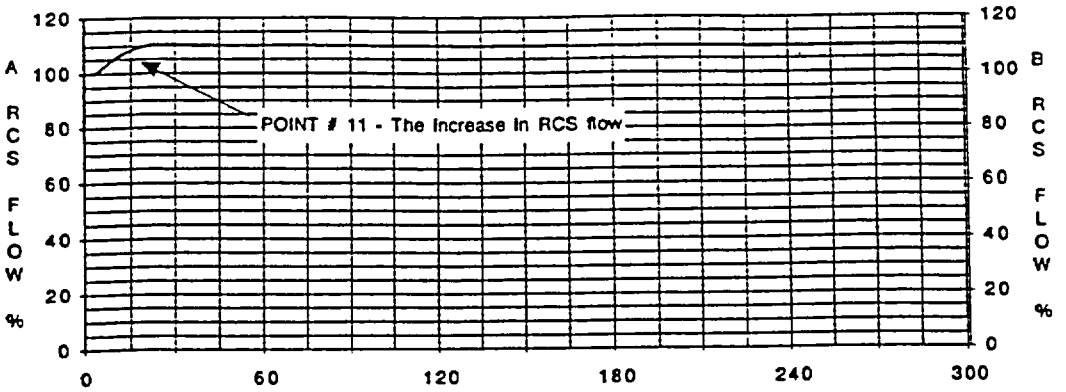

A OTSG LEVEL in
B OTSG LEVEL in
POINT #9 - The decrease in OTSG levels

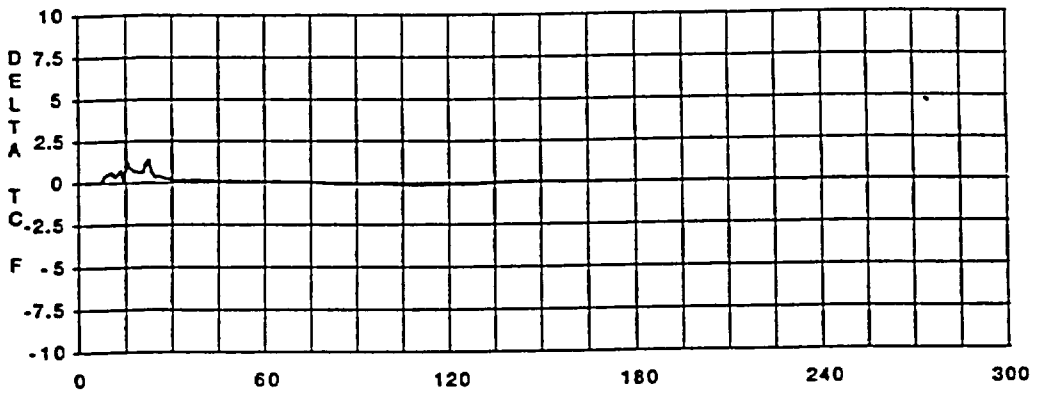

RX DEMAND %
RX POWER %
POINT #2 - The rapid decrease in reactor power
0
20
40
60
80
100
120
0
60
120
180
240
300


GROUP 7 %
GROUP 6 %
POINT # 1- The rapid decrease in rod position
0
10
20
30
40
50
60
70
80
90
100
0
60
120
180
240
300


TAVG F
PZR LEVEL in
POINT # 5 - The decrease in Tavg
POINT # 6 - The decrease in pzr level
POINT # 10 - The increase in pzr level
530
540
550
560
570
580
590
600
610
620
630
640
650
0
50
100
150
200
250
300
350
400
0
60
120
180
240
300

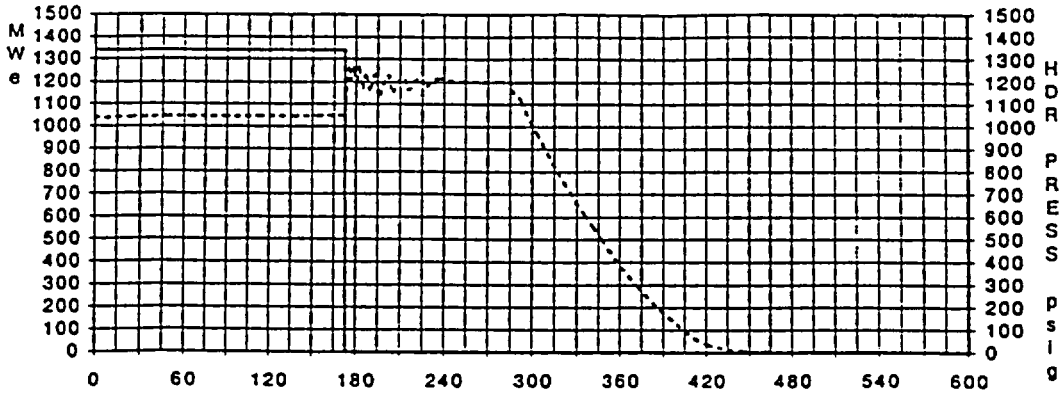

NR PRESS psig
WR PRESS psig
POINT # 7 - The decrease in RCS pressure
1500
1600
1700
1800
1900
2000
2100
2200
2300
2400
2500
0
500
1000
1500
2000
2500
0
60
120
180
240
300

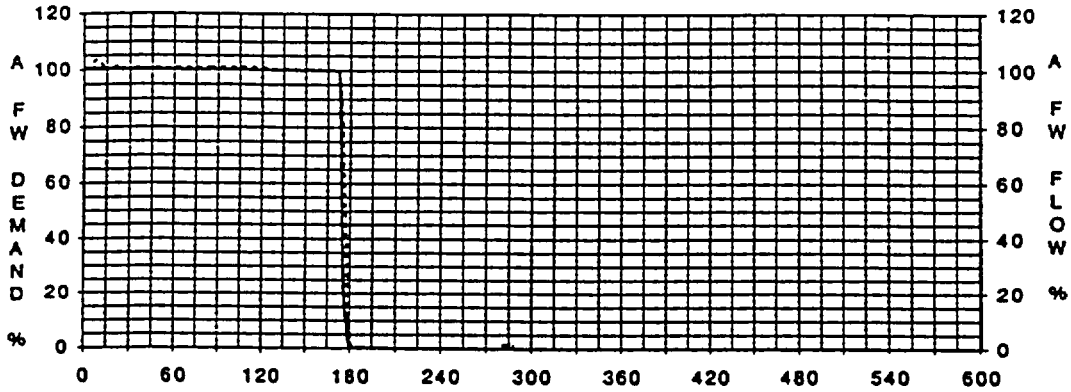

A RCS FLOW %
B RCS FLOW %
POINT # 11 - The Increase In RCS flow
120
100
80
60
40
20
0
0
60
120
180
240
300

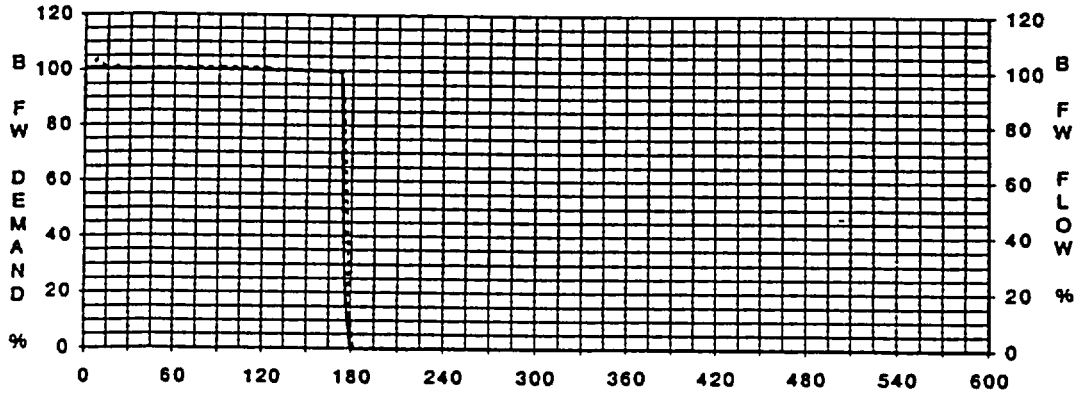

DELTA TC F
10
7.5
5
2.5
0
-2.5
-5
-7.5
-10
0
60
120
180
240
300

Transient 24

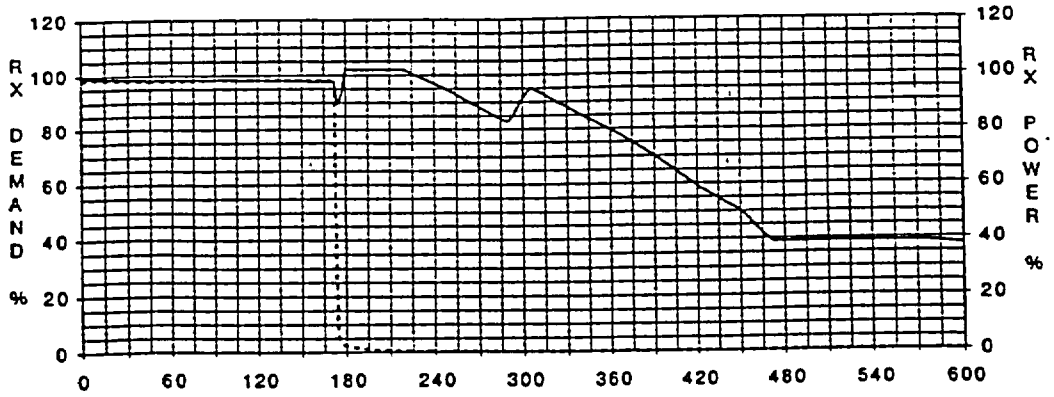
Small Break LOCA

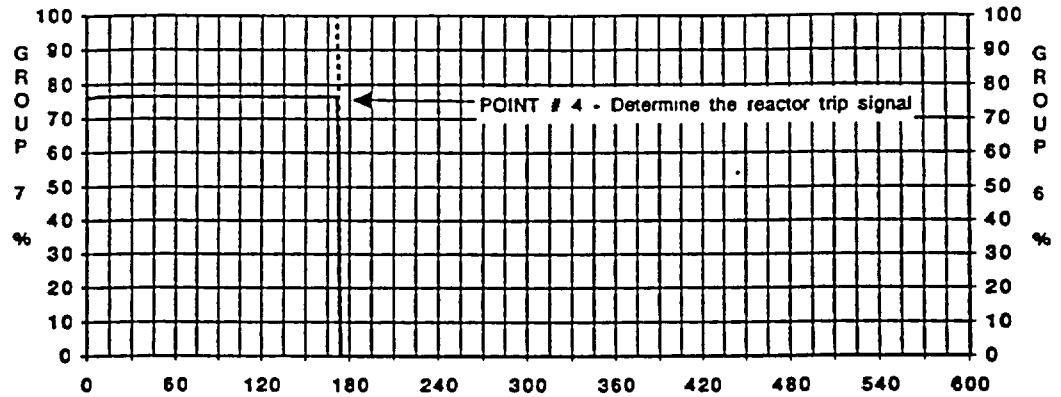

MWe
HDR PRESS psig
1500
1400
1300
1200
1100
1000
900
800
700
600
500
400
300
200
100
0
0 60 120 180 240 300 360 420 480 540 600

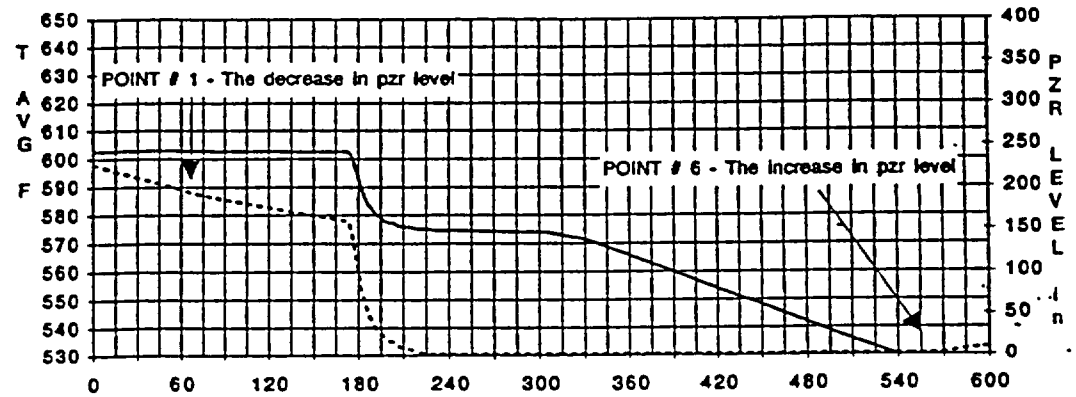

A FW DEMAND %
A FW FLOW %
120
100
80
60
40
20
0
0 60 120 180 240 300 360 420 480 540 600

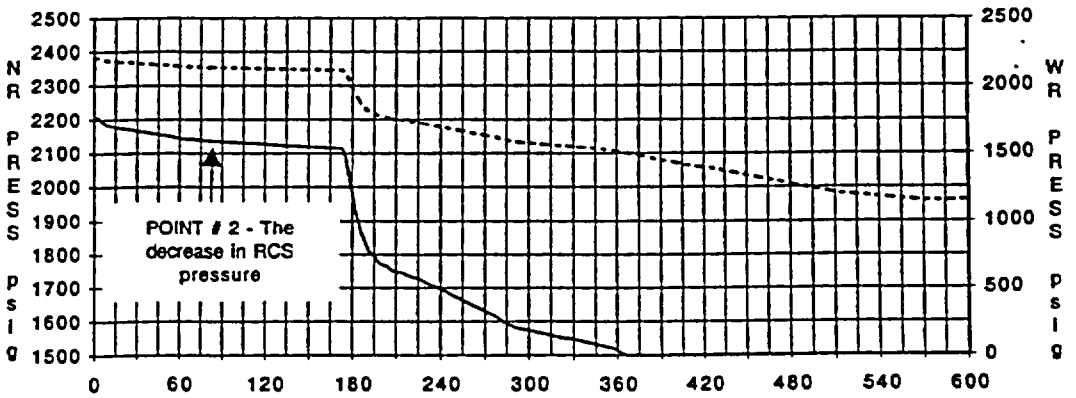

B FW DEMAND %
B FW FLOW %
120
100
80
60
40
20
0
0 60 120 180 240 300 360 420 480 540 600

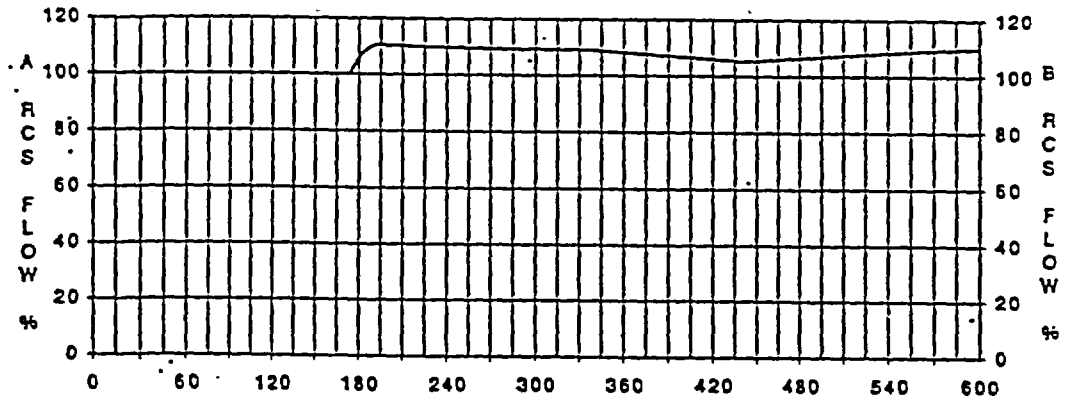
A OTSG LEVEL in
B OTSG LEVEL %
80
70
60
50
40
30
20
10
0
0 60 120 180 240 300 360 420 480 540 600

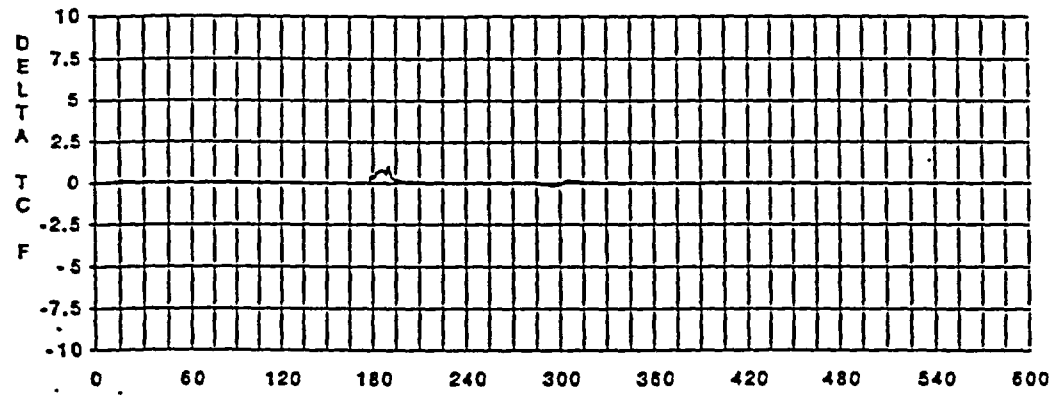

RX DEMAND %
RX POWER %
0 60 120 180 240 300 360 420 480 540 600

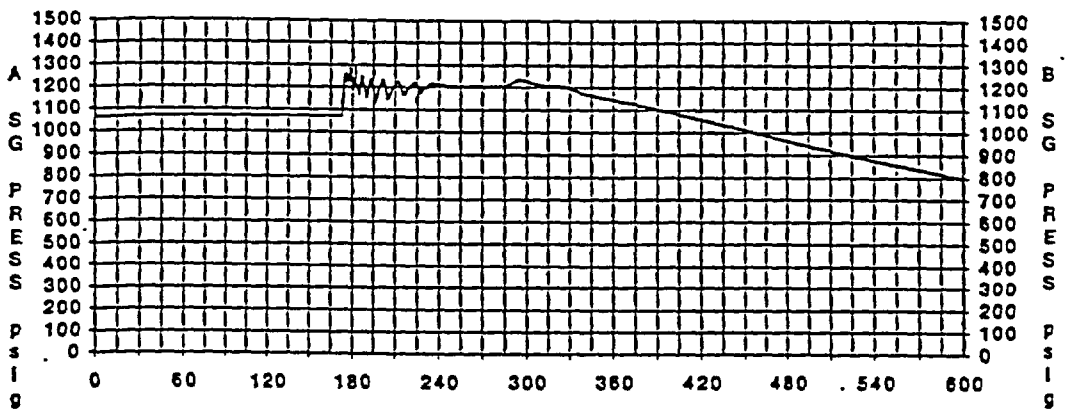

GROUP 7 %
GROUP 6 %
POINT # 4 - Determine the reactor trip signal
0 60 120 180 240 300 360 420 480 540 600

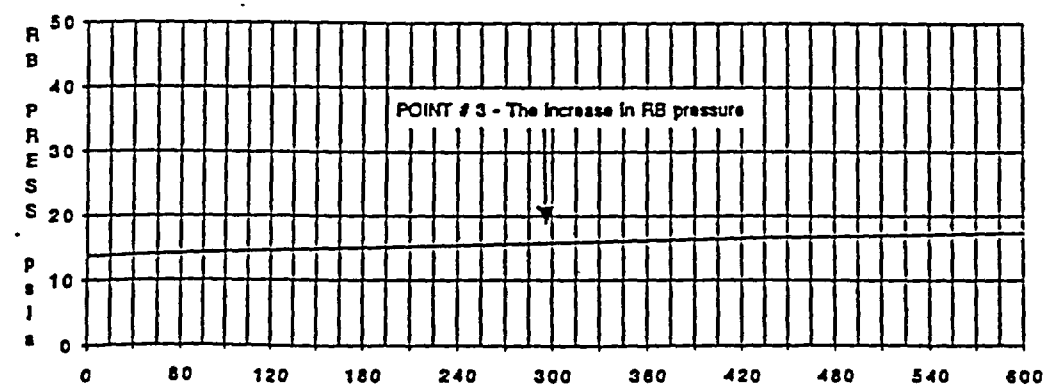

T AVG F
PZR LEVEL in
POINT # 1 - The decrease in pzr level
POINT # 6 - The increase in pzr level
0 60 120 180 240 300 360 420 480 540 600

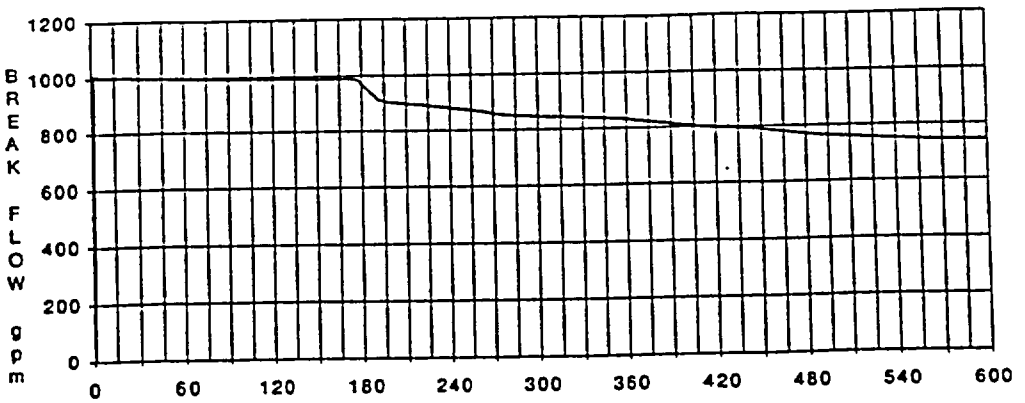

NR PRESS psig
WR PRESS psig
POINT # 2 - The decrease in RCS pressure
0 60 120 180 240 300 360 420 480 540 600

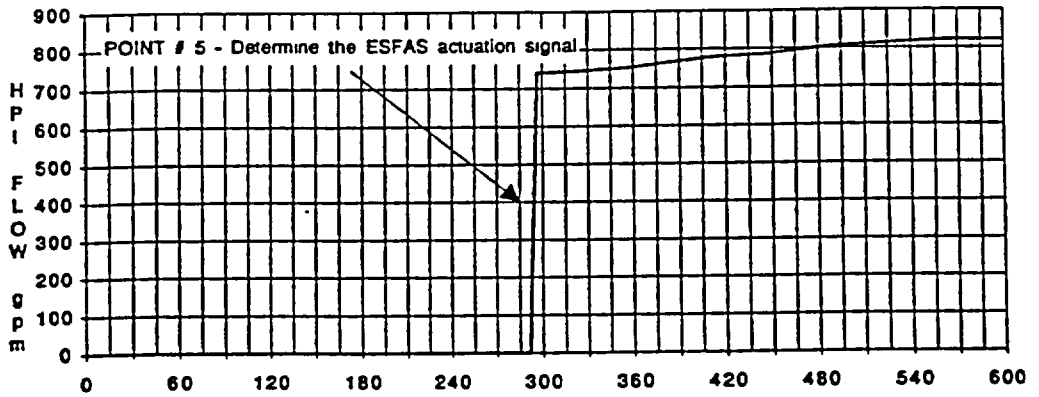

A RCS FLOW %
B RCS FLOW %
0 20 40 60 80 100 120
0 60 120 180 240 300 360 420 480 540 600

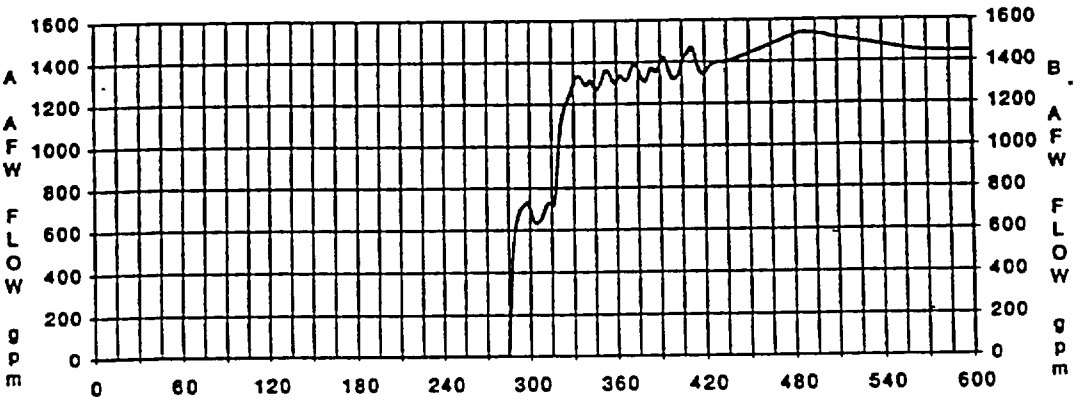

DELTA TCF
10 7.5 5 2.5 0 -2.5 -5 -7.5 -10
0 60 120 180 240 300 360 420 480 540 600


A SG PRESS psig
B SG PRESS psig
0 100 200 300 400 500 600 700 800 900 1000 1100 1200 1300 1400 1500
0 60 120 180 240 300 360 420 480 540 600


RB PRESS psia
0 10 20 30 40 50
POINT # 3 - The increase in RB pressure
0 60 120 180 240 300 360 420 480 540 600

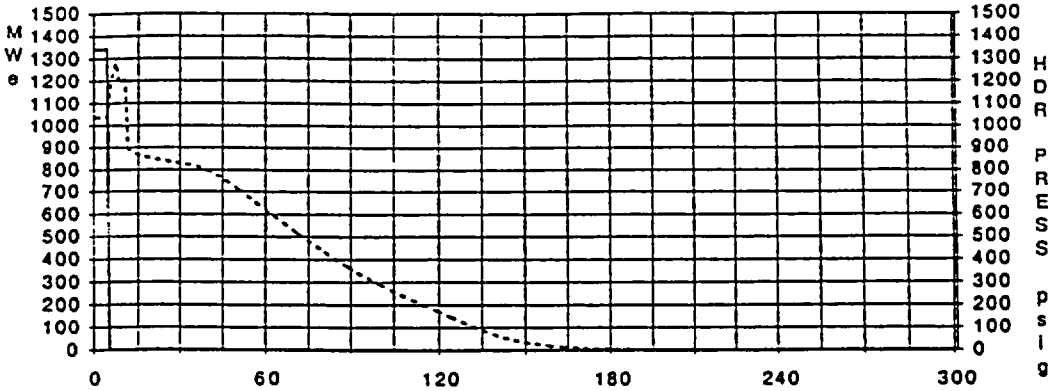

BREAK FLOW gpm
1200
1000
800
600
400
200
0
0
60
120
180
240
300
360
420
480
540
600

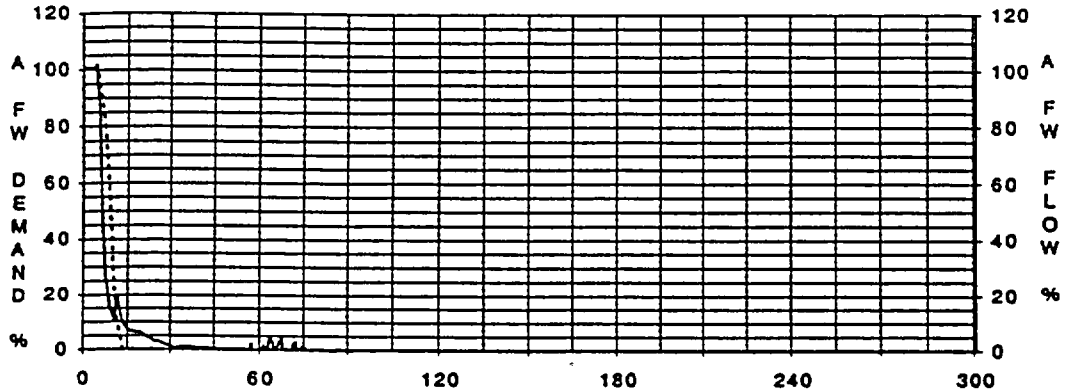

POINT # 5 - Determine the ESFAS actuation signal
HPI FLOW gpm
900
800
700
600
500
400
300
200
100
0
0
60
120
180
240
300
360
420
480
540
600

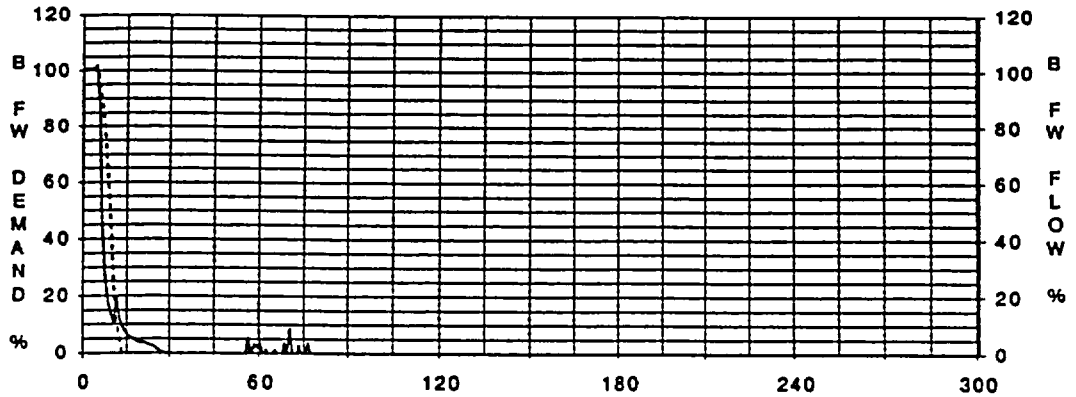

A AFW FLOW gpm
1600
1400
1200
1000
800
600
400
200
0
B AFW FLOW gpm
1600
1400
1200
1000
800
600
400
200
0
0
60
120
180
240
300
360
420
480
540
600

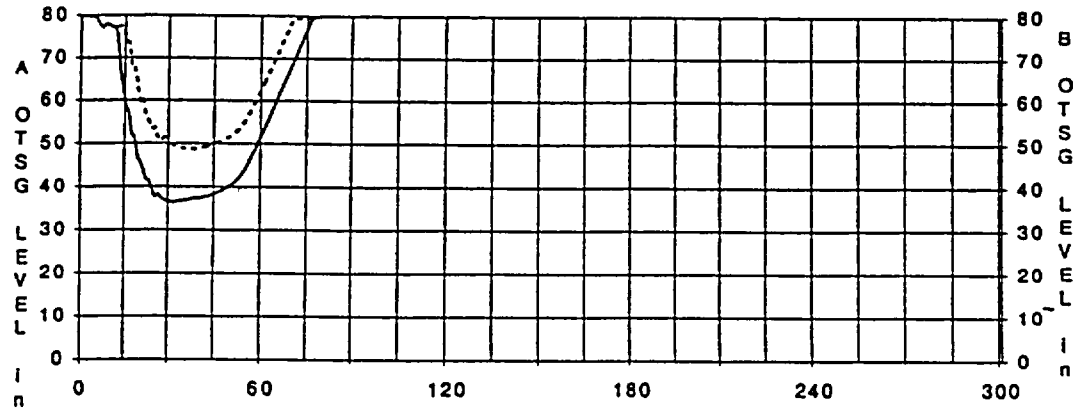
Transient 25

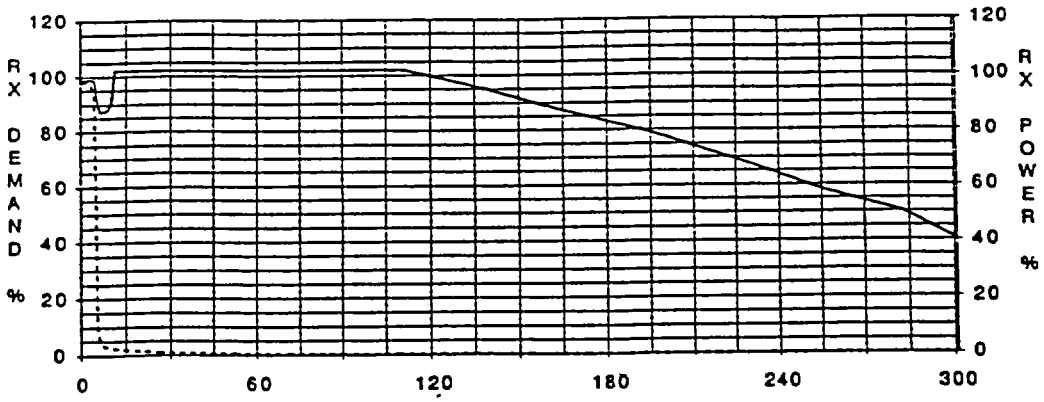
Large LOCA

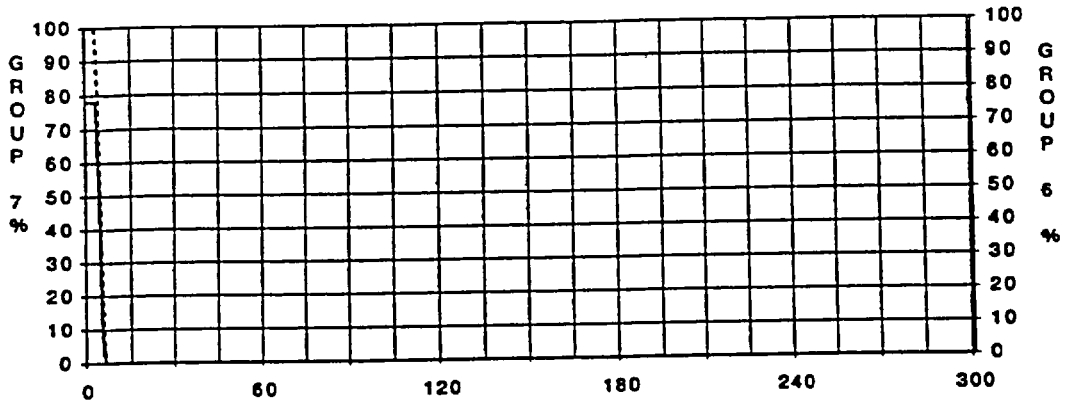

MWe
HDR PRESS psig
1500
1400
1300
1200
1100
1000
900
800
700
600
500
400
300
200
100
0
0
60
120
180
240
300

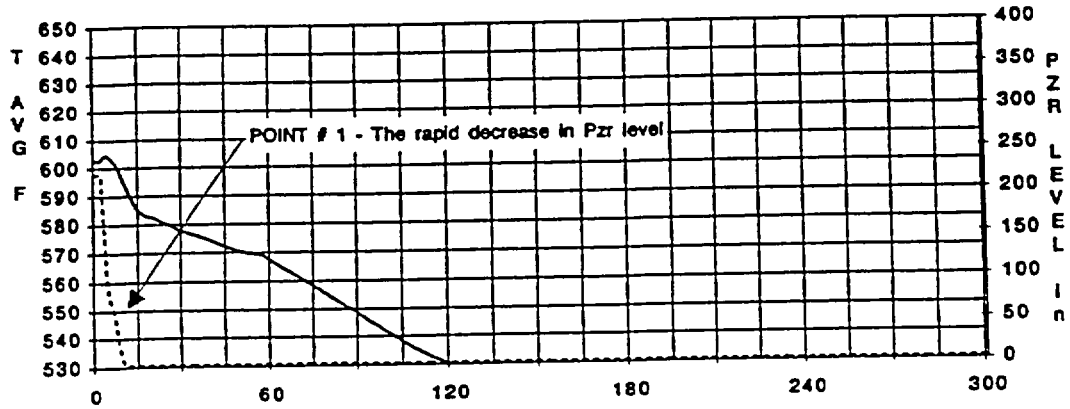

A FW DEMAND %
A FW FLOW %
120
100
80
60
40
20
0
0
60
120
180
240
300

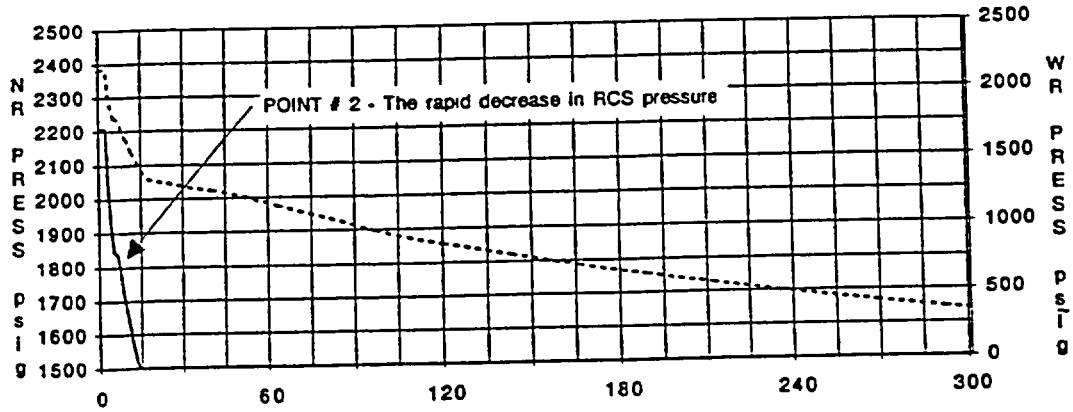

B FW DEMAND %
B FW FLOW %
120
100
80
60
40
20
0
0
60
120
180
240
300

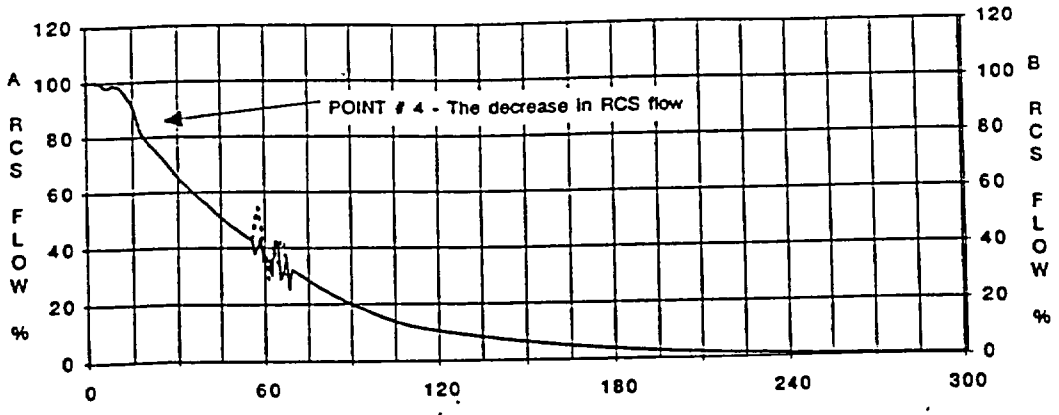

A OTSG LEVEL in
B OTSG LEVEL in
80
70
60
50
40
30
20
10
0
0
60
120
180
240
300

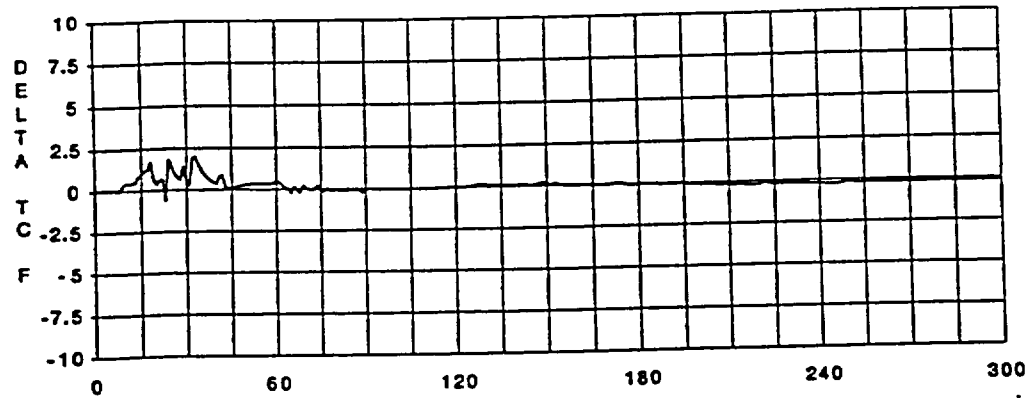

RX DEMAND %
RX POWER %
0
20
40
60
80
100
120
0
60
120
180
240
300

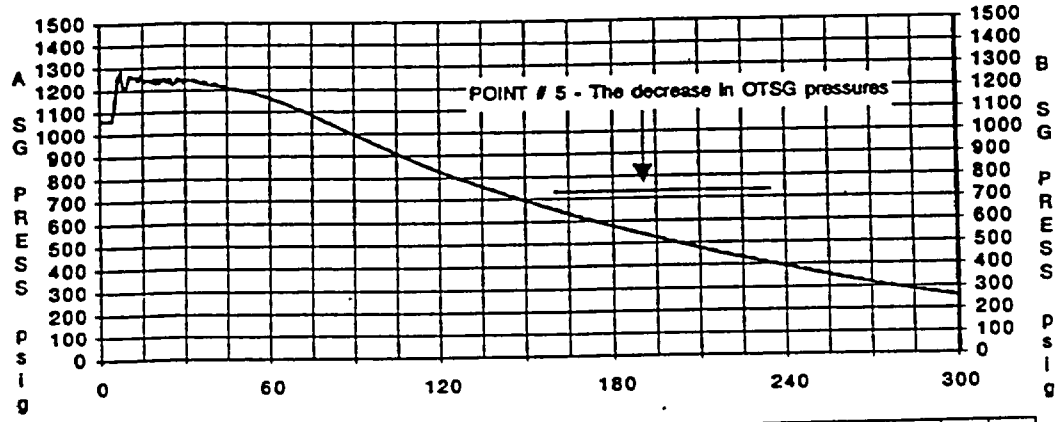

GROUP 7 %
GROUP 6 %
0
10
20
30
40
50
60
70
80
90
100
0
60
120
180
240
300

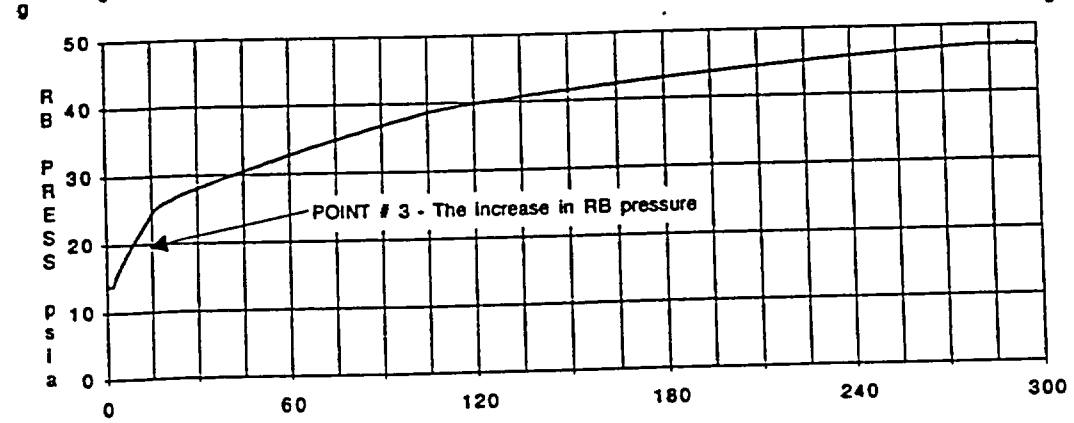

T AVG F
PZR LEVEL in
POINT # 1 - The rapid decrease in Pzr level
530
540
550
560
570
580
590
600
610
620
630
640
650
0
50
100
150
200
250
300
350
400
0
60
120
180
240
300


NR PRESS psig
WR PRESS psig
POINT # 2 - The rapid decrease in RCS pressure
1500
1600
1700
1800
1900
2000
2100
2200
2300
2400
2500
0
500
1000
1500
2000
2500
0
60
120
180
240
300


A RCS FLOW %
B RCS FLOW %
POINT # 4 - The decrease in RCS flow
0
20
40
60
80
100
120
0
60
120
180
240
300

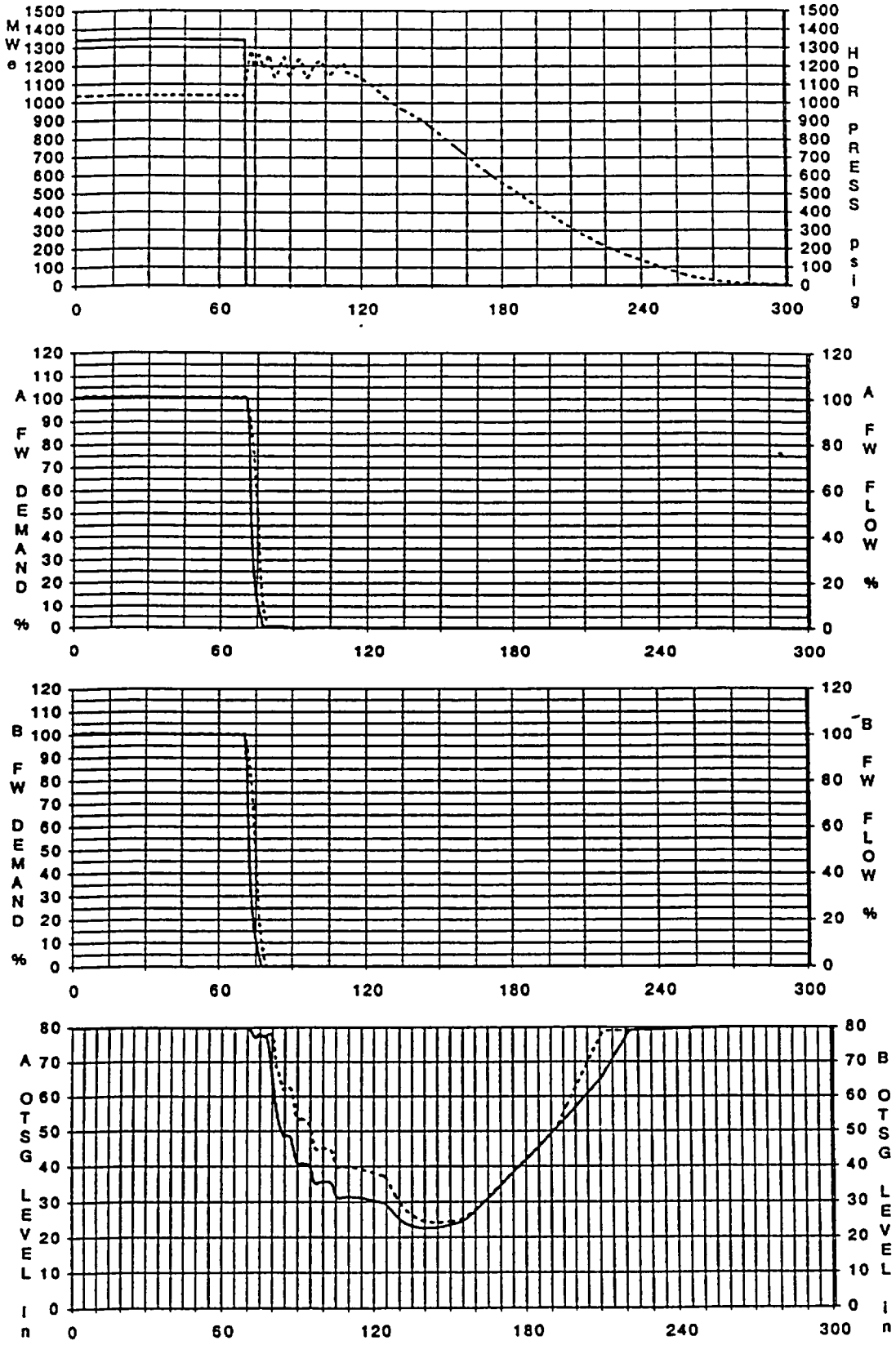

DELTA TCF
10
7.5
5
2.5
0
-2.5
-5
-7.5
-10
0
60
120
180
240
300

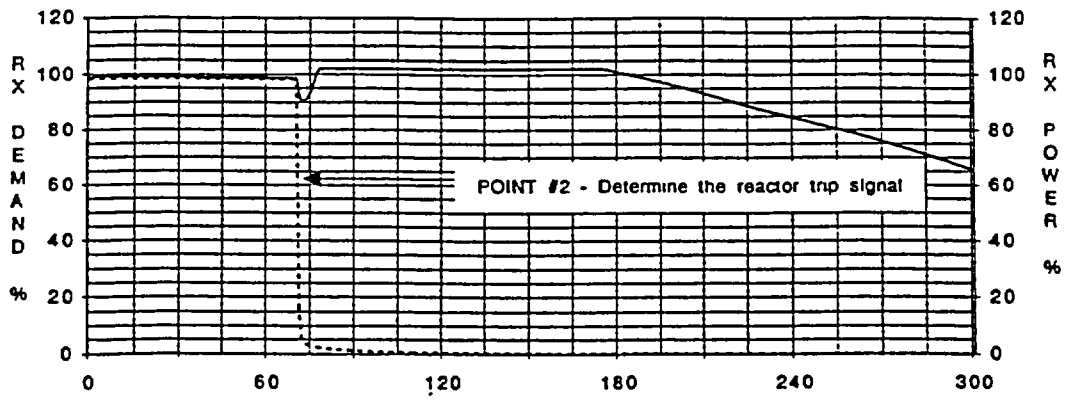

A SG PRESS psig
B SG PRESS psig
POINT # 5 - The decrease in OTSG pressures
0
100
200
300
400
500
600
700
800
900
1000
1100
1200
1300
1400
1500
0
60
120
180
240
300

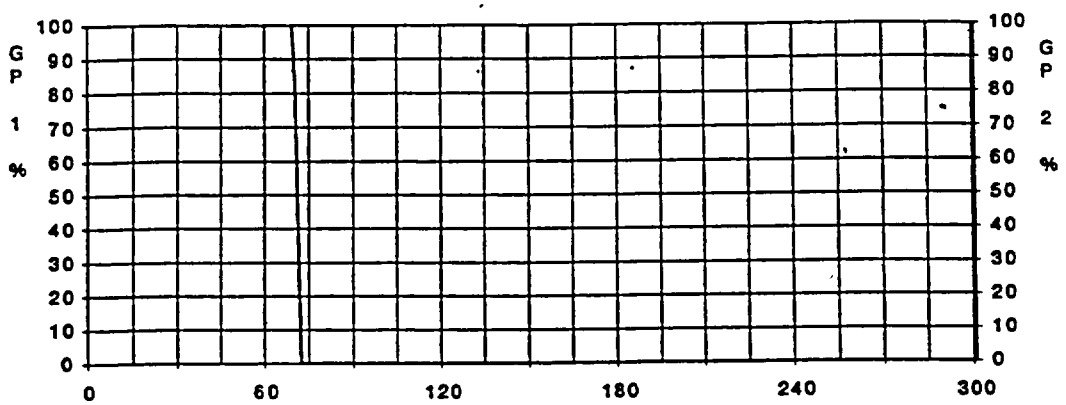

RB PRESS psia
POINT # 3 - The increase in RB pressure
0
10
20
30
40
50
0
60
120
180
240
300

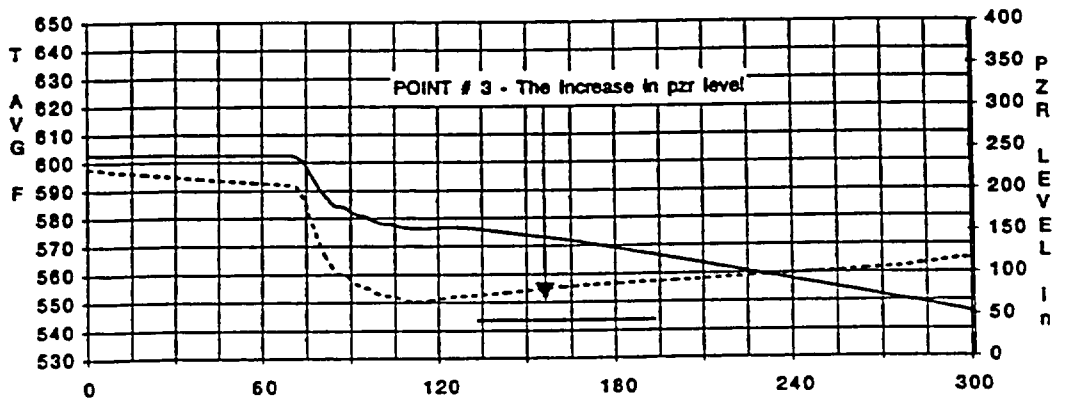
Transient 26

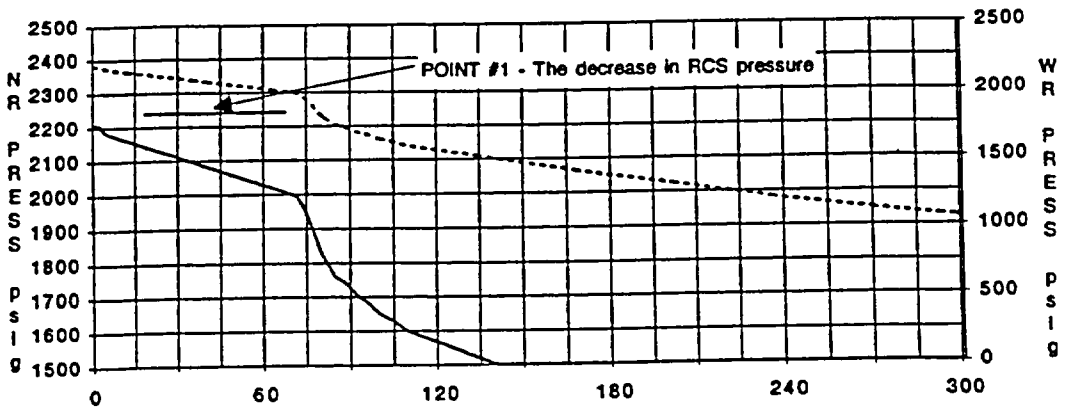
Failed Open PORV

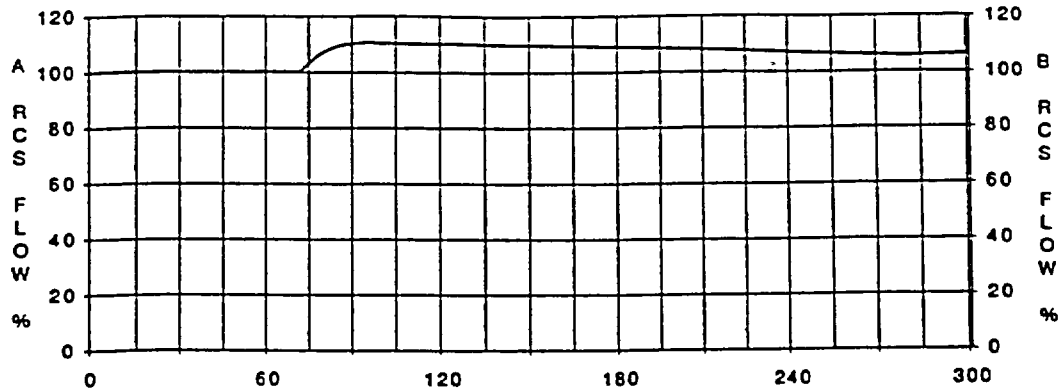

MWe
HDR PRESS psig
A FW DEMAND %
A FW FLOW %
B FW DEMAND %
B FW FLOW %
A OTSG LEVEL in
B OTSG LEVEL in
0
60
120
180
240
300

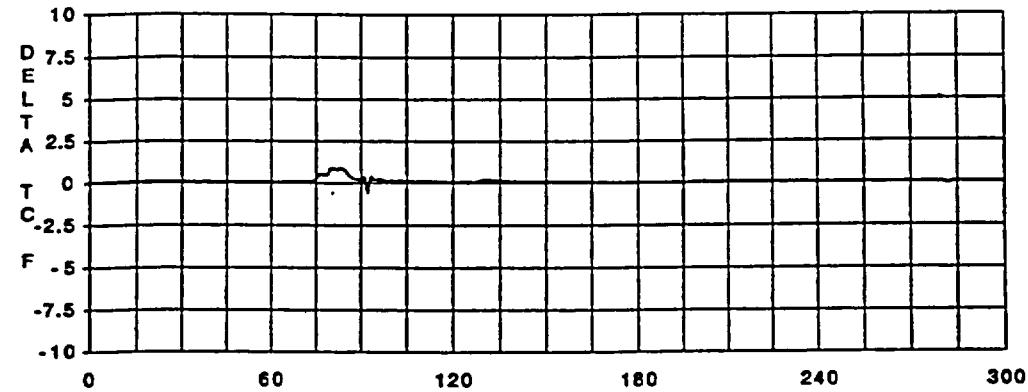

RX DEMAND %
RX POWER %
POINT #2 - Determine the reactor trip signal
0
20
40
60
80
100
120
0
60
120
180
240
300

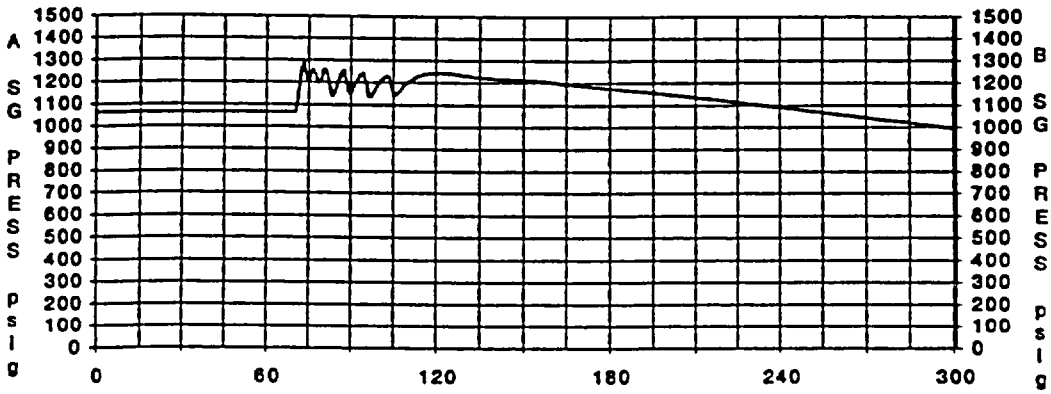

GP 1 %
GP 2 %
0
10
20
30
40
50
60
70
80
90
100
0
60
120
180
240
300

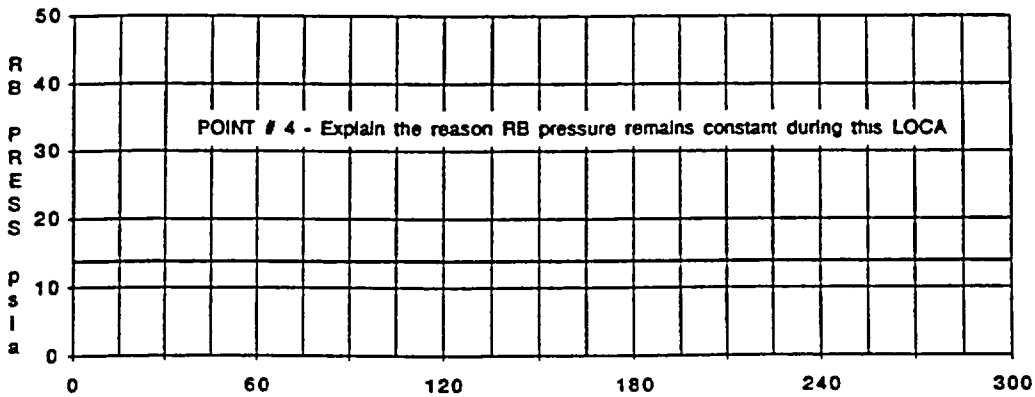

T AVG F
PZR LEVEL in
POINT # 3 - The increase in pzr level
530
540
550
560
570
580
590
600
610
620
630
640
650
0
50
100
150
200
250
300
350
400
0
60
120
180
240
300


NR PRESS psig
WR PRESS psig
POINT #1 - The decrease in RCS pressure
1500
1600
1700
1800
1900
2000
2100
2200
2300
2400
2500
0
500
1000
1500
2000
2500
0
60
120
180
240
300

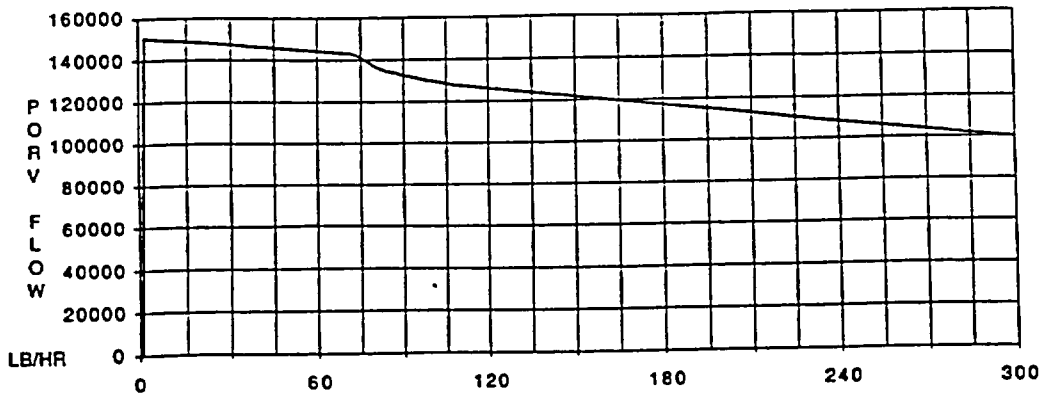


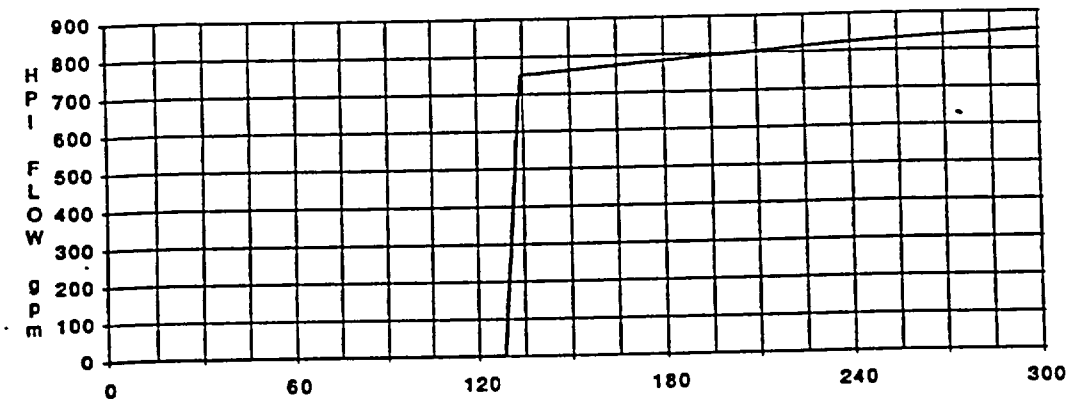


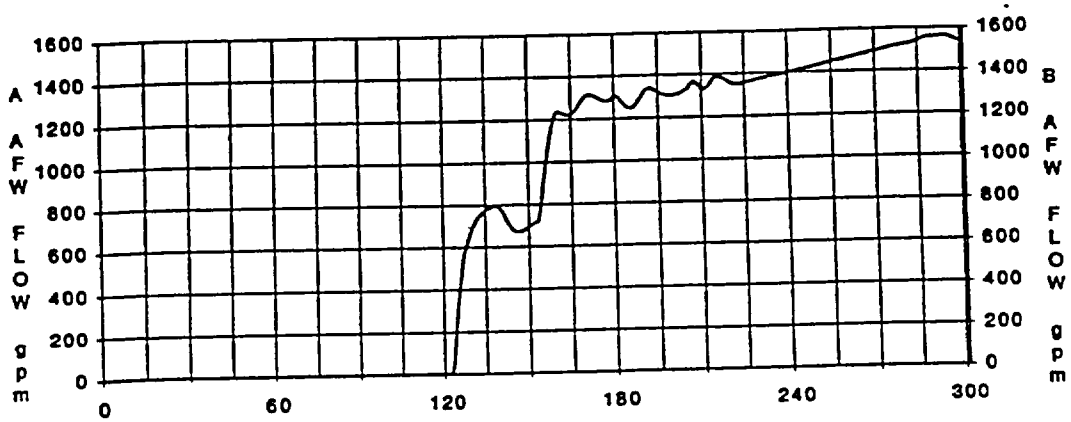




POINT # 4 - Explain the reason RB pressure remains constant during this LOCA

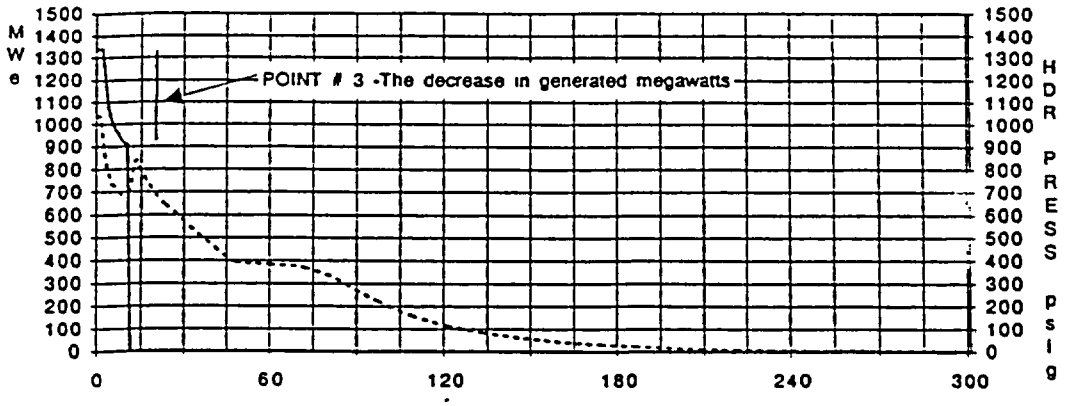

160000
140000
120000
100000
80000
60000
40000
20000
0
PORV FLOW
LB/HR
0
60
120
180
240
300

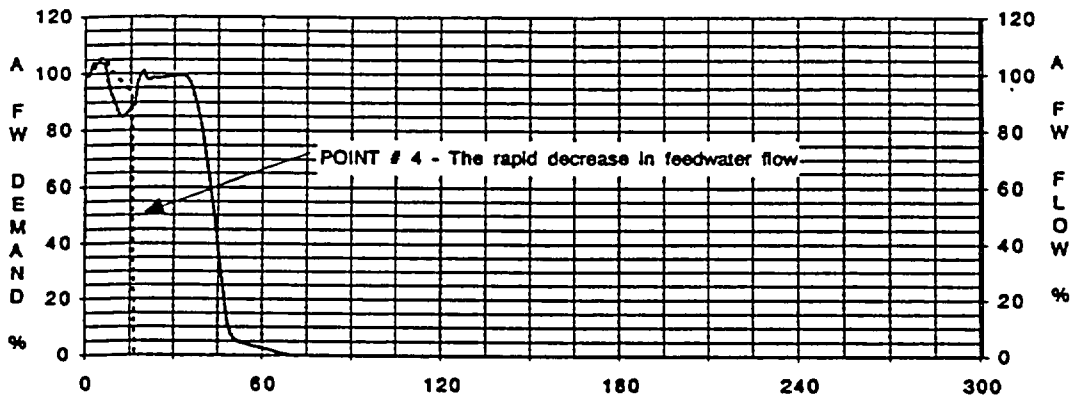

900
800
700
600
500
400
300
200
100
0
HPI FLOW gpm
0
60
120
180
240
300

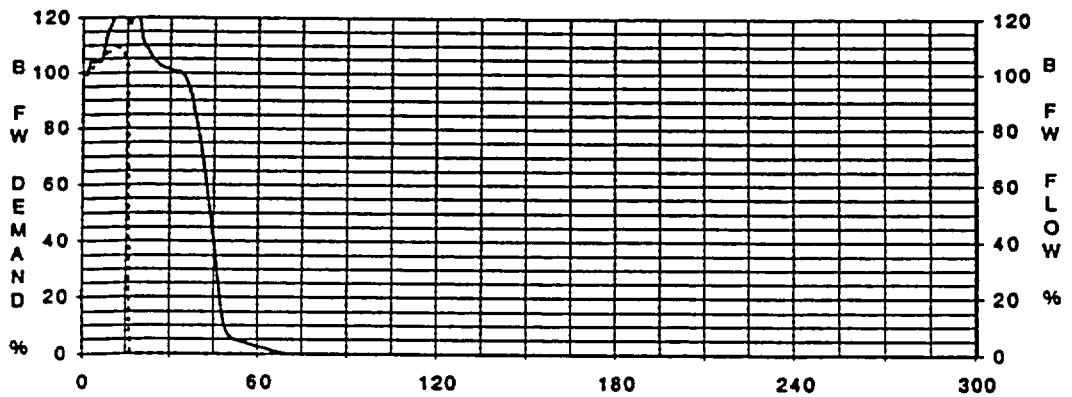

1600
1400
1200
1000
800
600
400
200
0
A AFW FLOW gpm
B AFW FLOW gpm
0
60
120
180
240
300

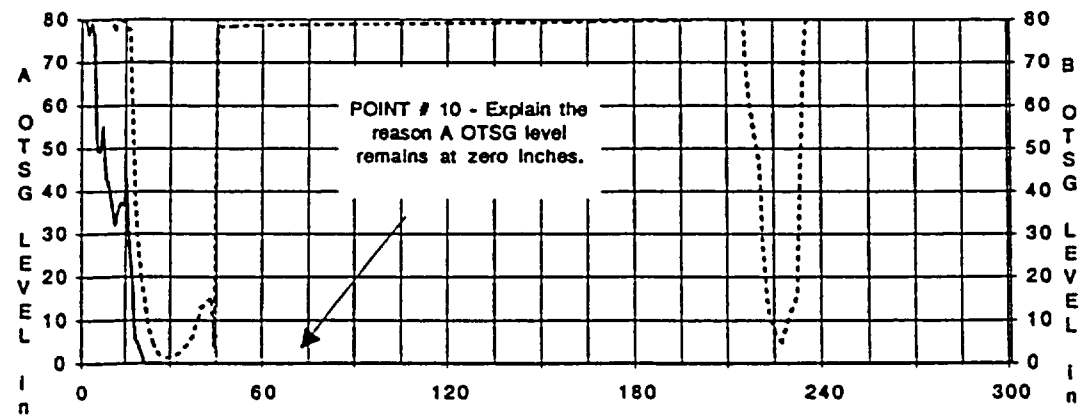
Transient 27

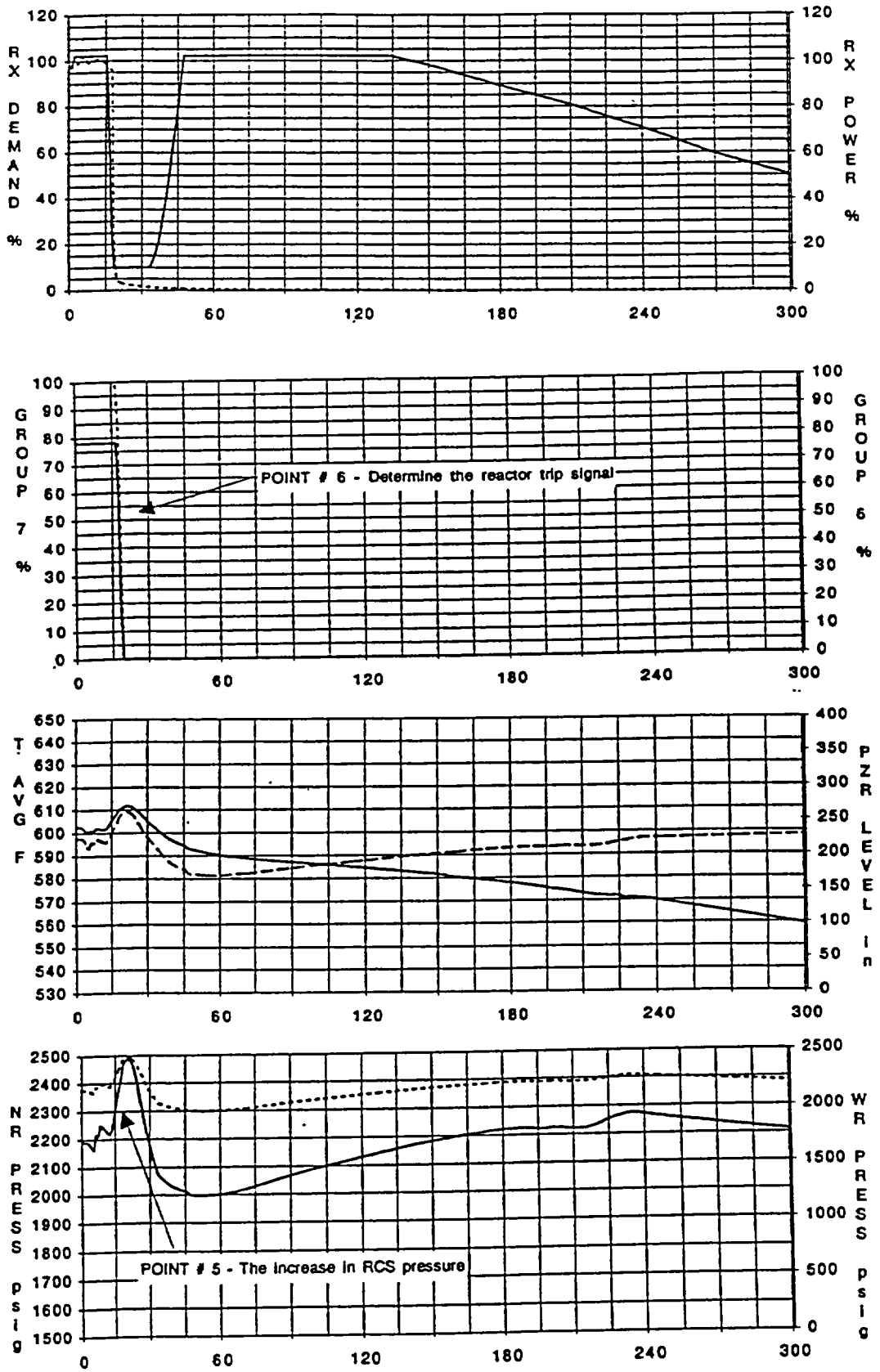
Main Steam Break Inside of the Reactor Building

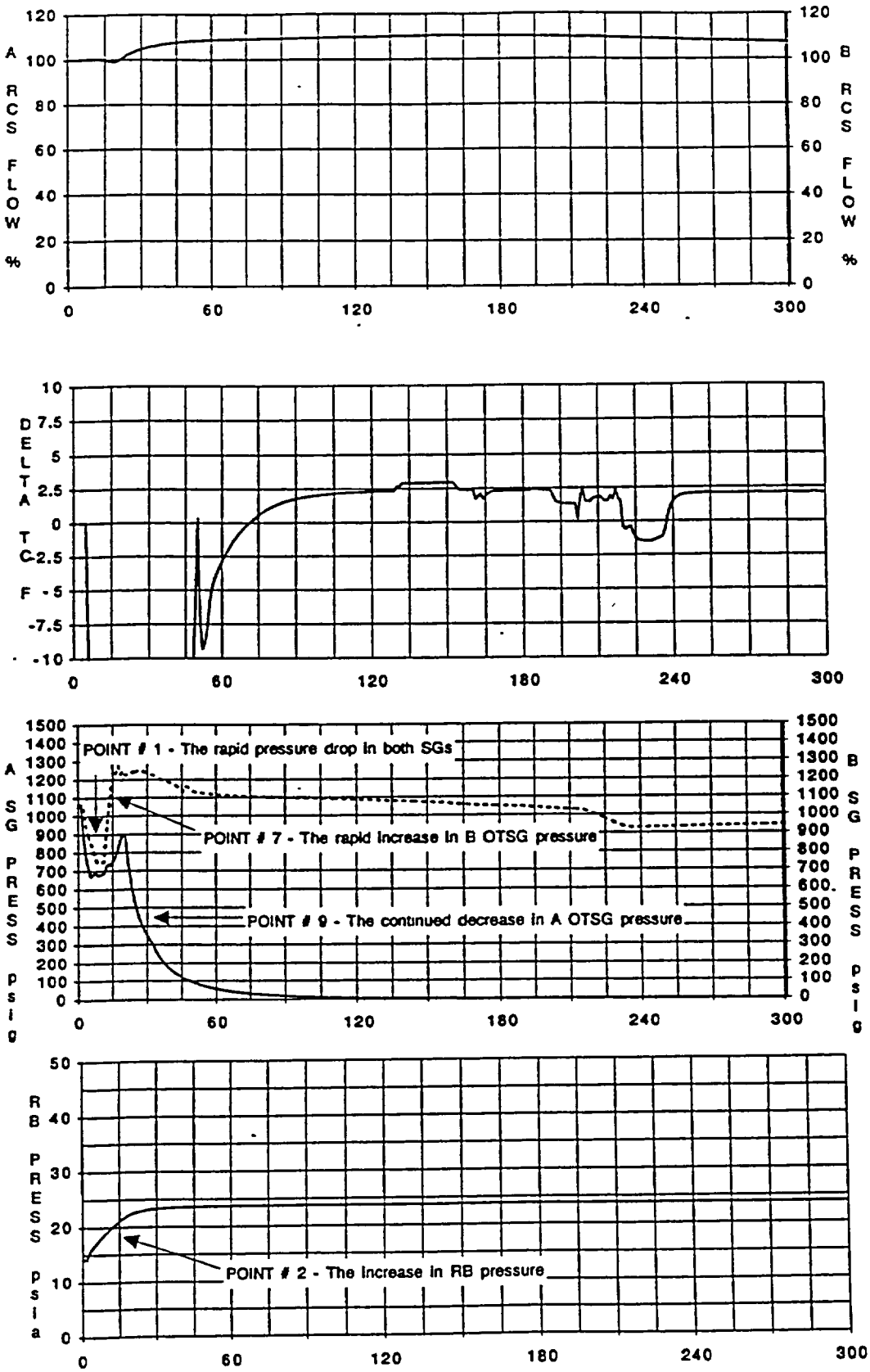

MWe
HDR PRESS psig
POINT # 3 -The decrease in generated megawatts
0 60 120 180 240 300

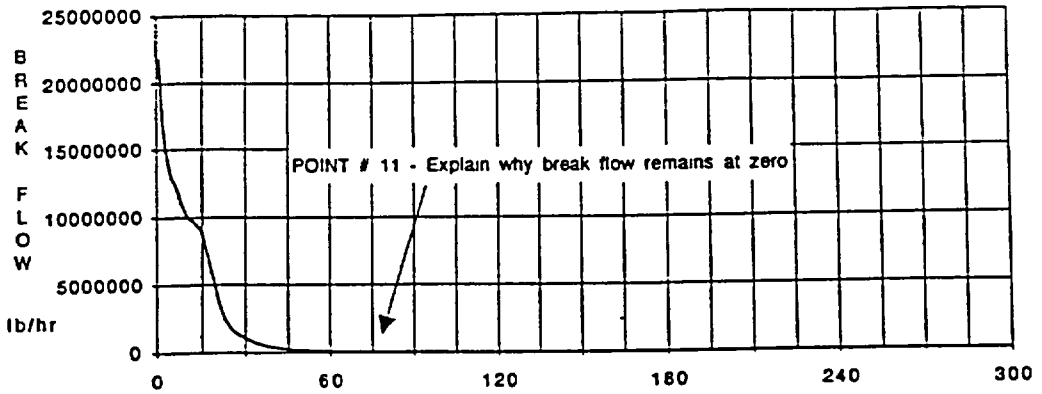

A FW DEMAND %
A FW FLOW %
POINT # 4 - The rapid decrease in feedwater flow
0 60 120 180 240 300

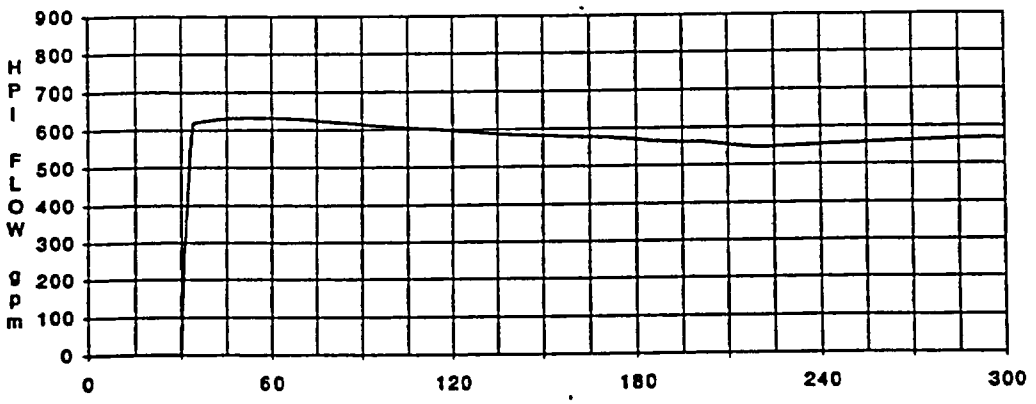

B FW DEMAND %
B FW FLOW %
0 60 120 180 240 300

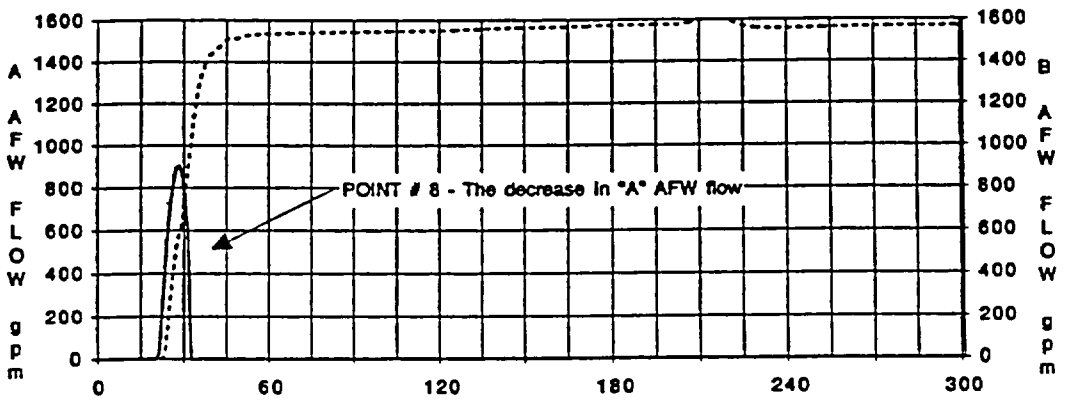

A OTSG LEVEL in
B OTSG LEVEL in
POINT # 10 - Explain the reason A OTSG level remains at zero inches.
0 60 120 180 240 300


RX DEMAND %
RX POWER %
GROUP 7 %
GROUP 6 %
POINT # 6 - Determine the reactor trip signal
T AVG F
PZR LEVEL in
NR PRESS psig
WR PRESS psig
POINT # 5 - The increase in RCS pressure


A RCS FLOW %
B RCS FLOW %
DELTA TC F
A SG PRESS psig
B SG PRESS psig
RB PRESS psia
POINT # 1 - The rapid pressure drop in both SGs
POINT # 7 - The rapid increase in B OTSG pressure
POINT # 9 - The continued decrease in A OTSG pressure
POINT # 2 - The increase in RB pressure

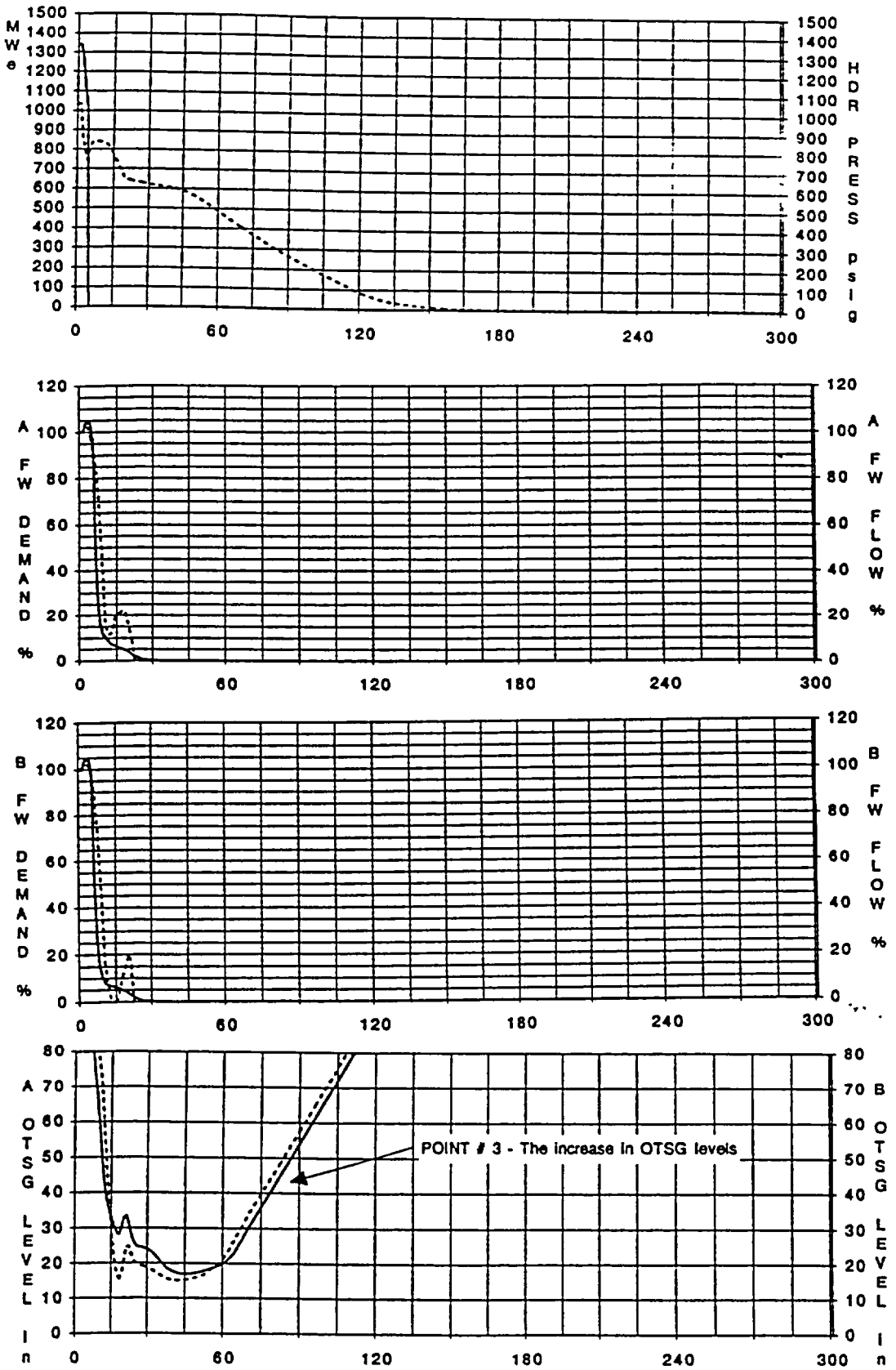

BREAK FLOW lb/hr
25000000
20000000
15000000
10000000
5000000
0
0
60
120
180
240
300
POINT # 11 - Explain why break flow remains at zero

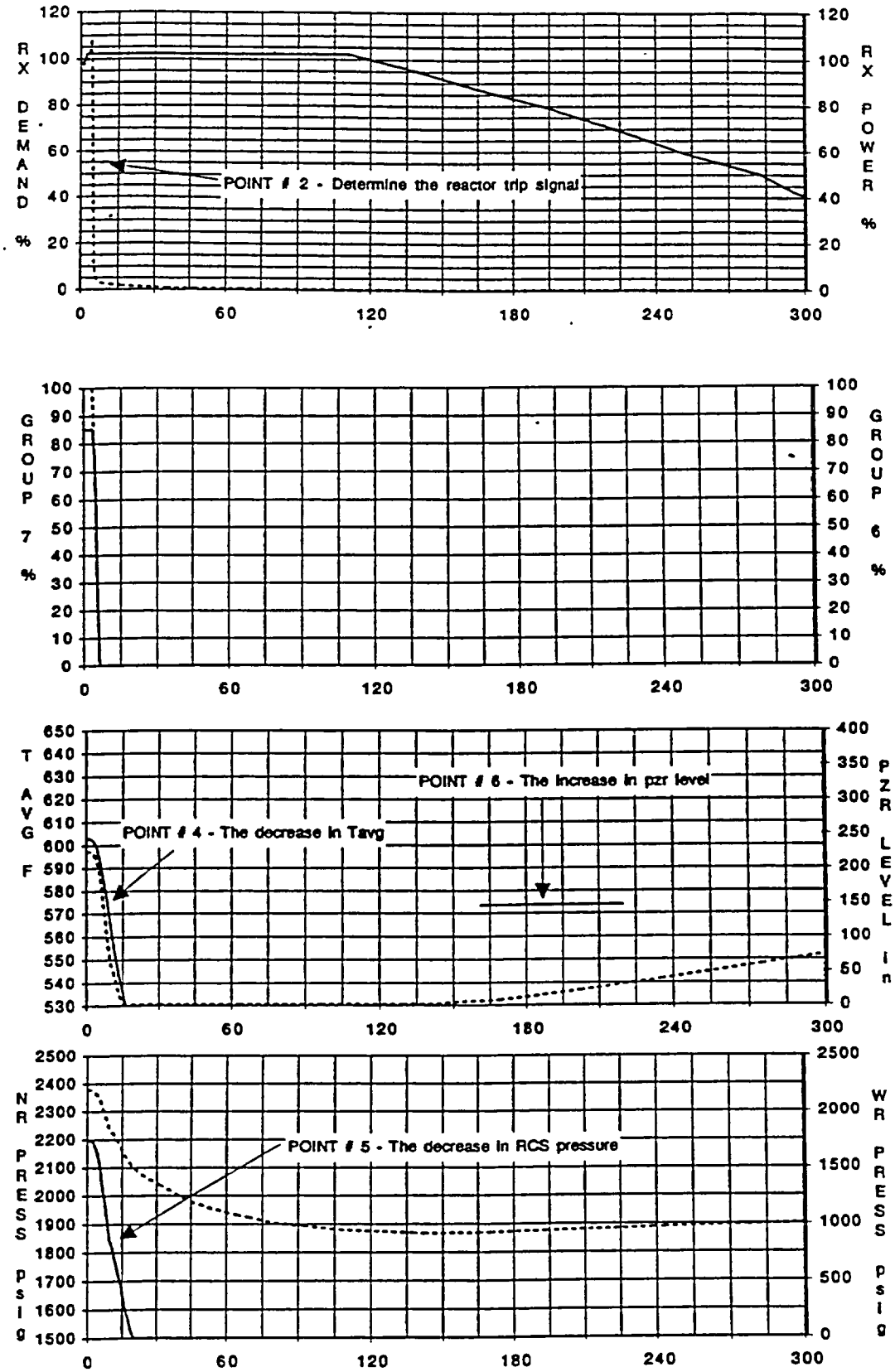

HPI FLOW gpm
900
800
700
600
500
400
300
200
100
0
0
60
120
180
240
300

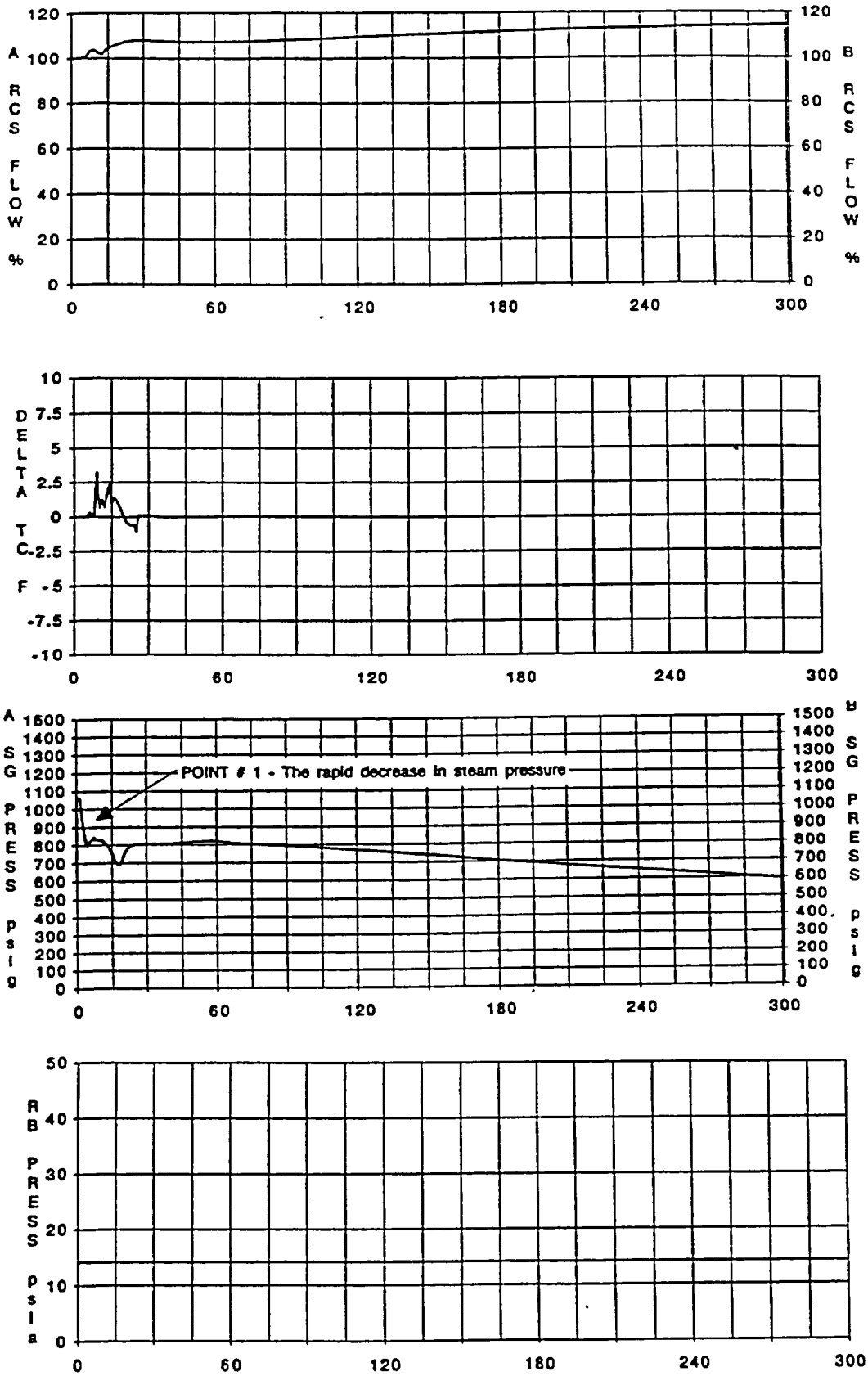

A AFW FLOW gpm
B AFW FLOW gpm
1600
1400
1200
1000
800
600
400
200
0
0
60
120
180
240
300
POINT # 8 - The decrease in "A" AFW flow

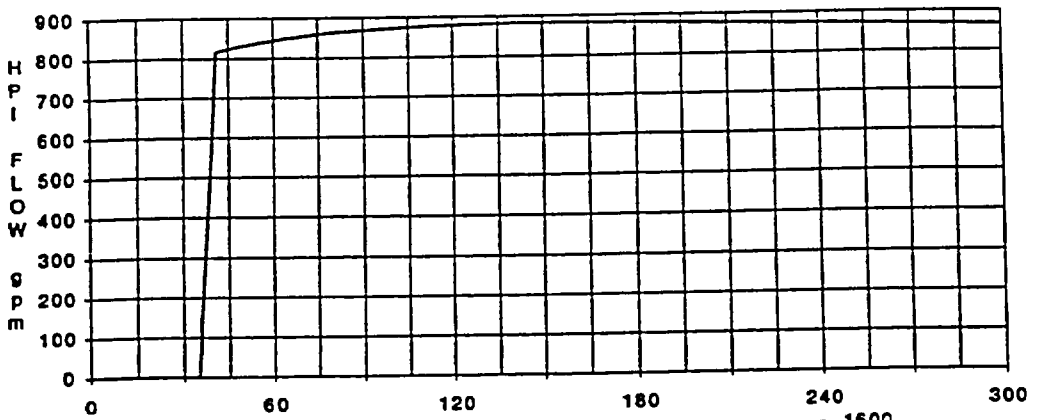
Transient 28

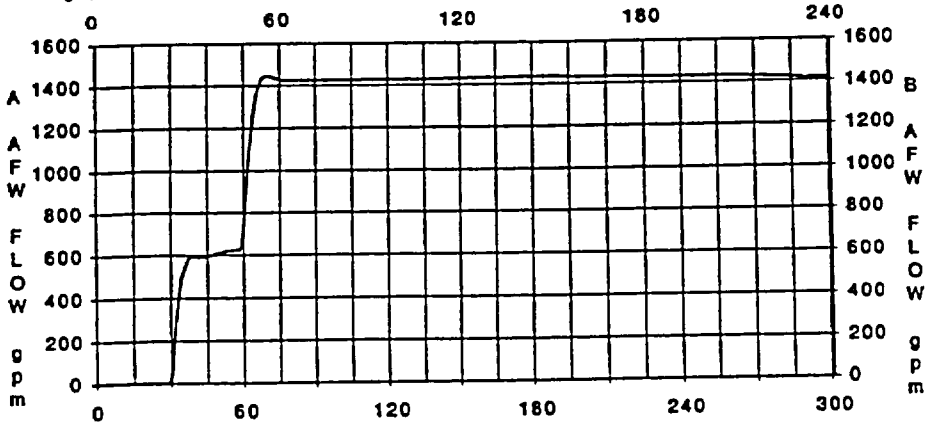
Main Steam Break Downstream of the MSIVs


MWe
HDR PRESS psig
A FW DEMAND %
A FW FLOW %
B FW DEMAND %
B FW FLOW %
A OTSG LEVEL In
B OTSG LEVEL In
POINT # 3 - The increase in OTSG levels
0
60
120
180
240
300


RX DEMAND %
RX POWER %
POINT # 2 - Determine the reactor trip signal
GROUP 7 %
GROUP 6 %
TAVG F
PZR LEVEL in
POINT # 6 - The increase in pzr level
POINT # 4 - The decrease in Tavg
NR PRESS psig
WR PRESS psig
POINT # 5 - The decrease in RCS pressure
0
60
120
180
240
300

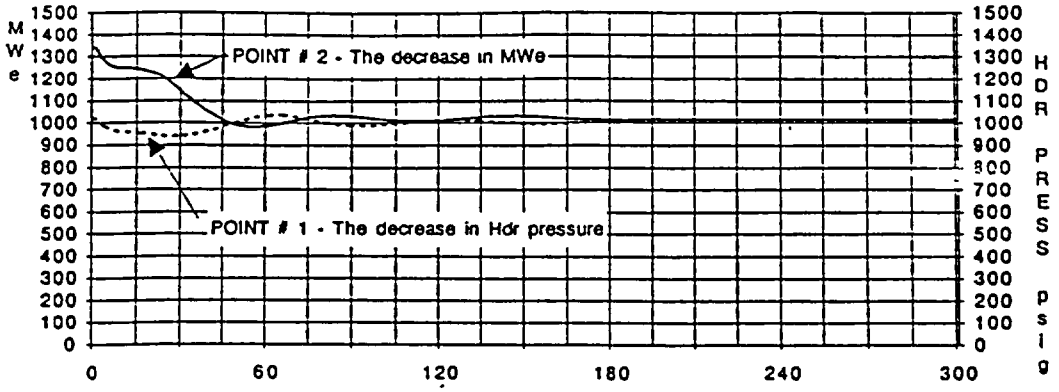

A RCS FLOW %
B RCS FLOW %
DELTA TC F
A SG PRESS psig
B SG PRESS psig
POINT # 1 - The rapid decrease in steam pressure
RB PRESS psia

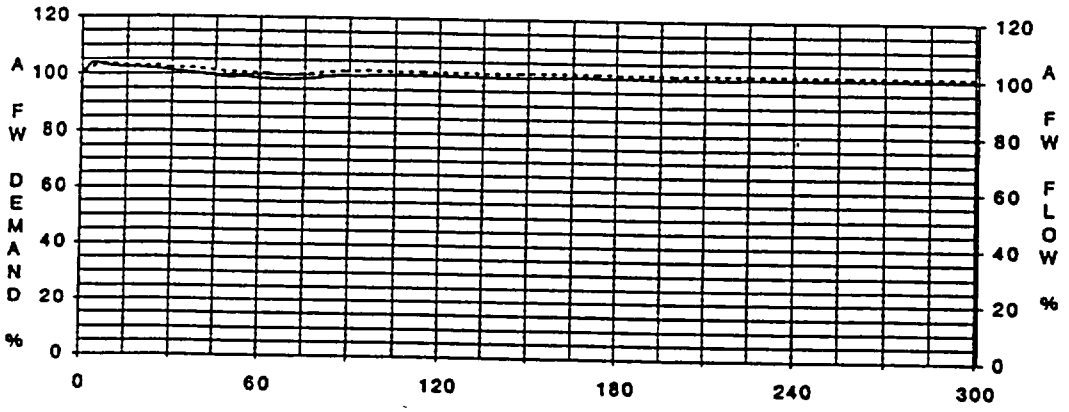

HPI FLOW gpm
900
800
700
600
500
400
300
200
100
0
0
60
120
180
240
300

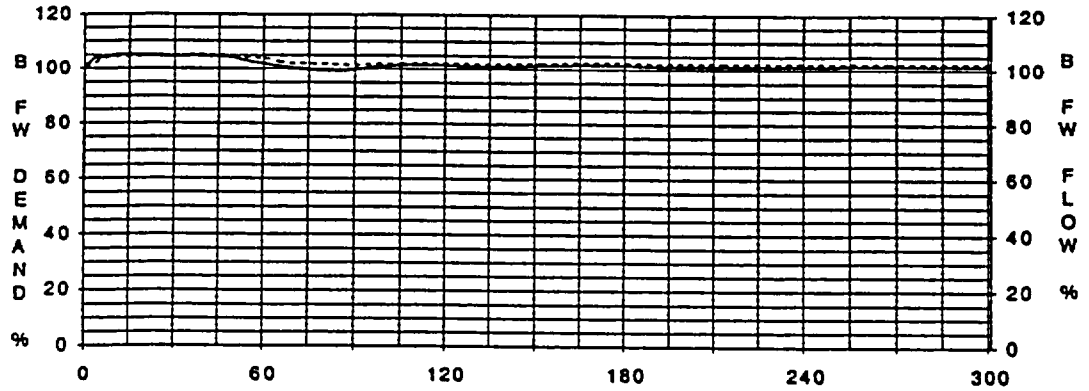

A AFW FLOW gpm
1600
1400
1200
1000
800
600
400
200
0
B AFW FLOW gpm
1600
1400
1200
1000
800
600
400
200
0
0
60
120
180
240
300

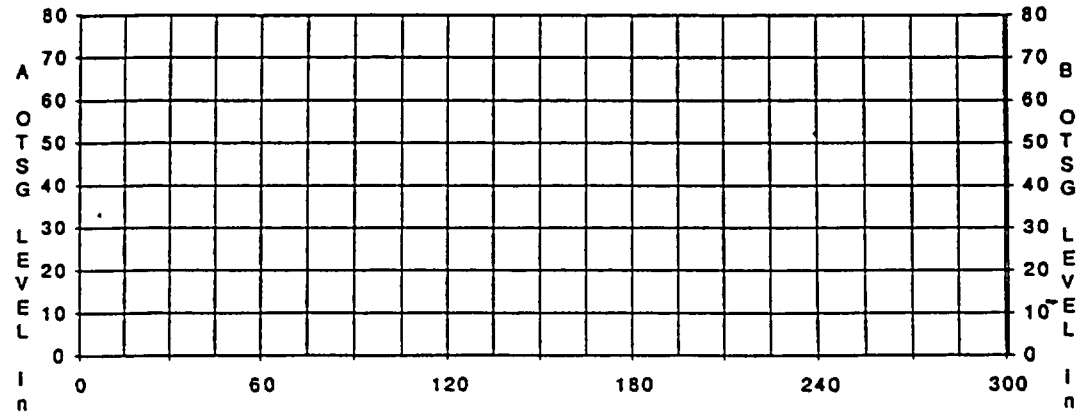
Transient 29

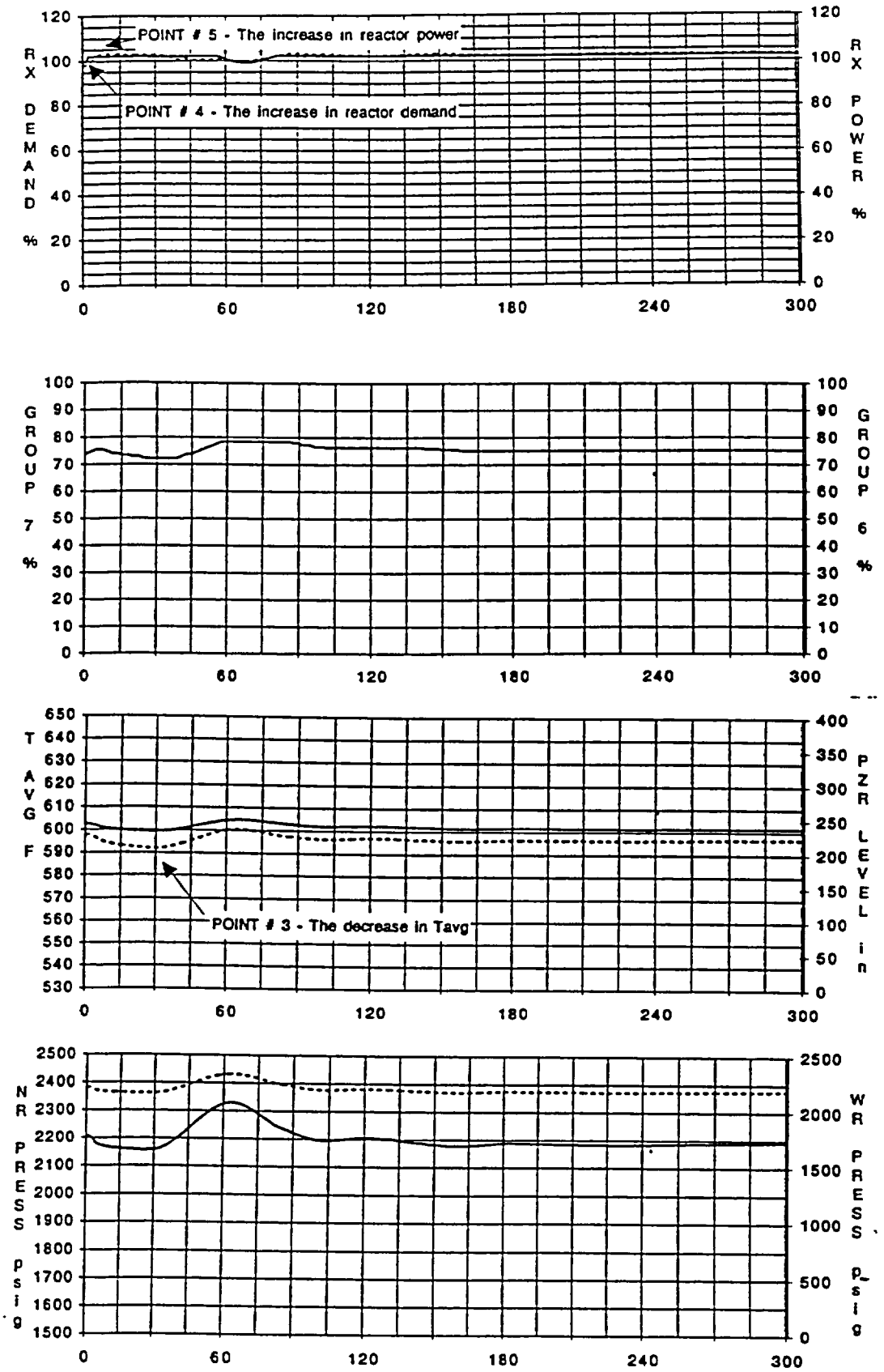
Main Steam Safety Fails Open

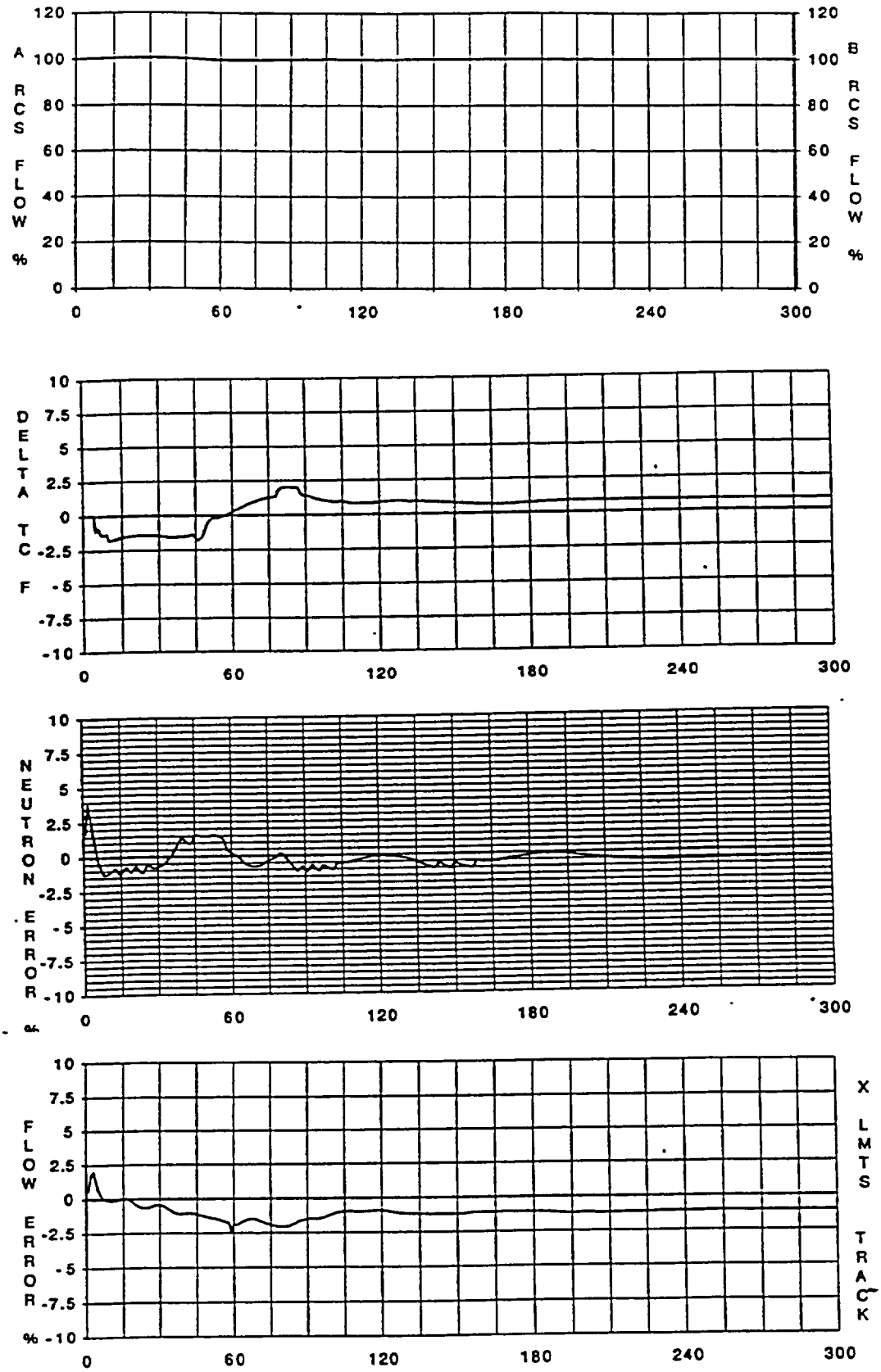

MWe
HDR PRESS psig
POINT # 2 - The decrease in MWe
POINT # 1 - The decrease in Hdr pressure
1500
1400
1300
1200
1100
1000
900
800
700
600
500
400
300
200
100
0
0
60
120
180
240
300


A FW DEMAND %
A FW FLOW %
120
100
80
60
40
20
0
0
60
120
180
240
300


B FW DEMAND %
B FW FLOW %
120
100
80
60
40
20
0
0
60
120
180
240
300

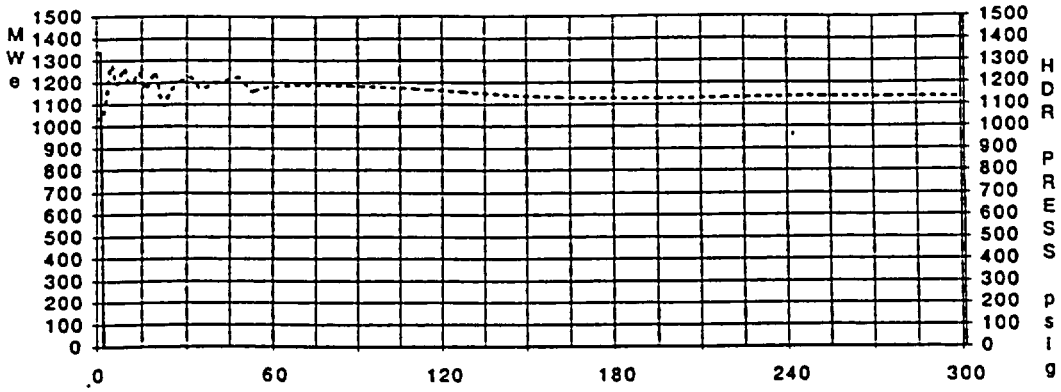

A OTSG LEVEL In
B OTSG LEVEL In
80
70
60
50
40
30
20
10
0
0
60
120
180
240
300

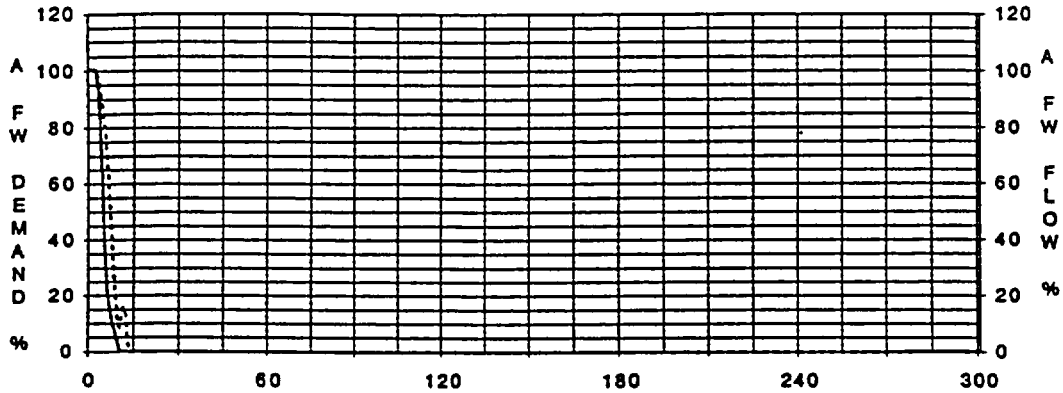

POINT # 5 - The increase in reactor power
POINT # 4 - The increase in reactor demand
RX DEMAND %
RX POWER %
GROUP 7 %
GROUP 6 %
TAVG F
PZR LEVEL in
POINT # 3 - The decrease in Tavg
NR PRESS psig
WR PRESS psig
0
60
120
180
240
300

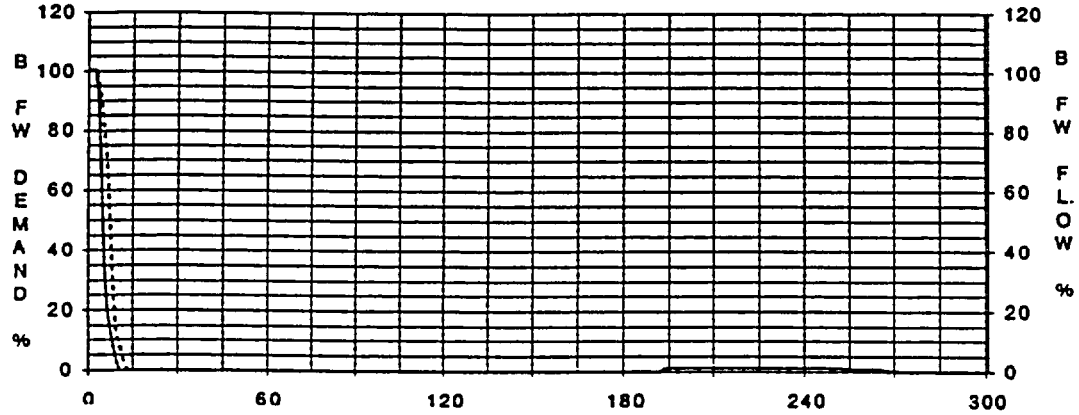

A RCS FLOW %
B RCS FLOW %
DELTA TCF
NEUTRON ERROR %
FLOW ERROR %
XLMTS TRACK
0
60
120
180
240
300
120
100
80
60
40
20
0
10
7.5
5
2.5
0
-2.5
-5
-7.5
-10

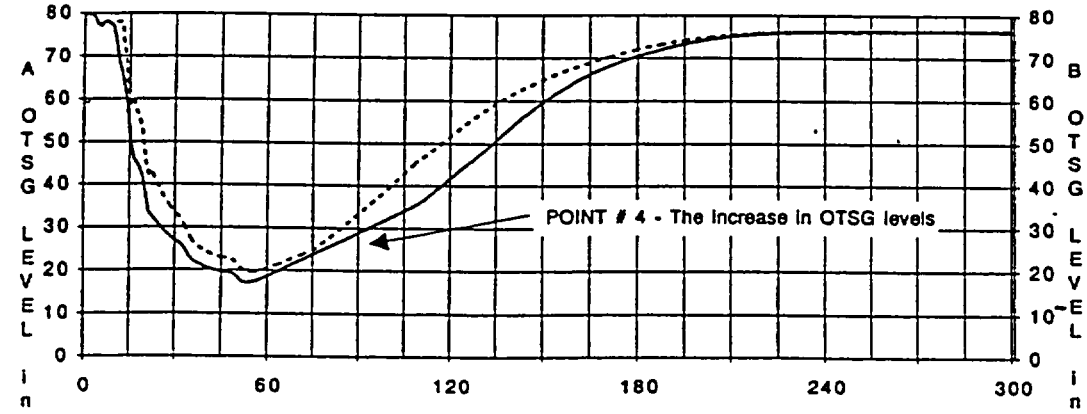
Transient 30

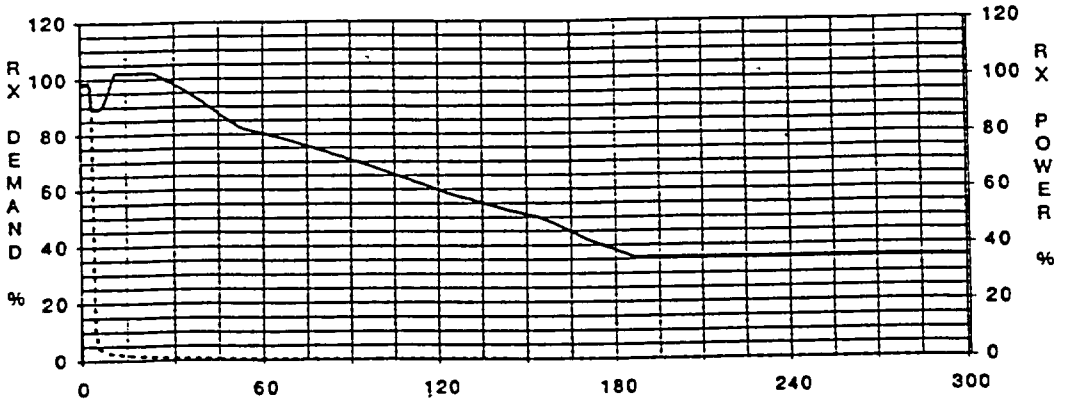
Loss of Offsite Power

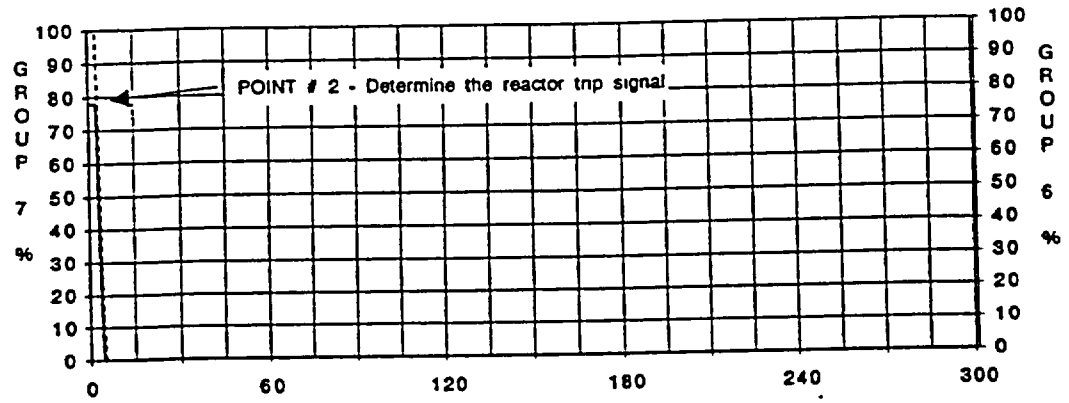

MWe
HDR PRESS psig
1500
1400
1300
1200
1100
1000
900
800
700
600
500
400
300
200
100
0
0
60
120
180
240
300

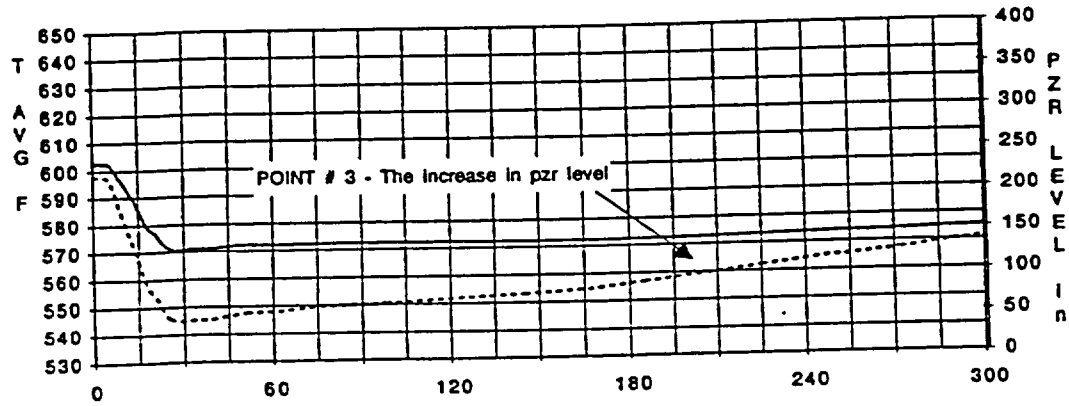

A FW DEMAND %
A FW FLOW %
120
100
80
60
40
20
0
0
60
120
180
240
300

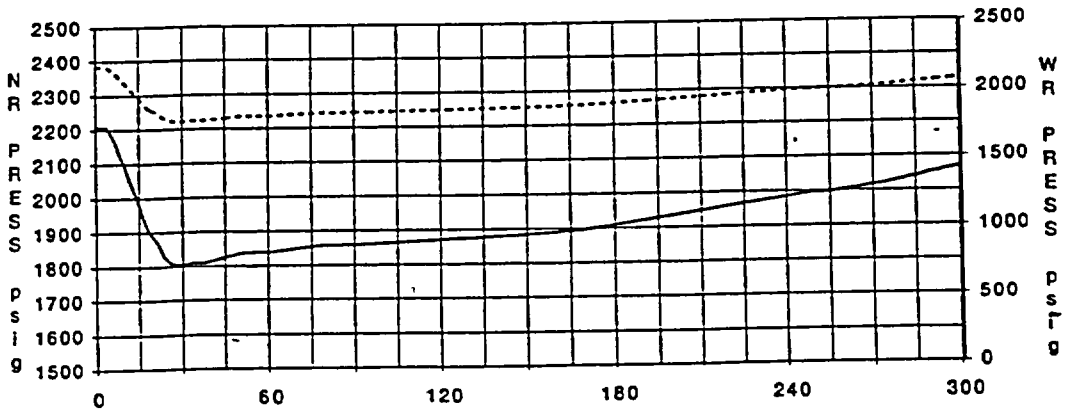

B FW DEMAND %
B FW FL.OW %
120
100
80
60
40
20
0
0
60
120
180
240
300

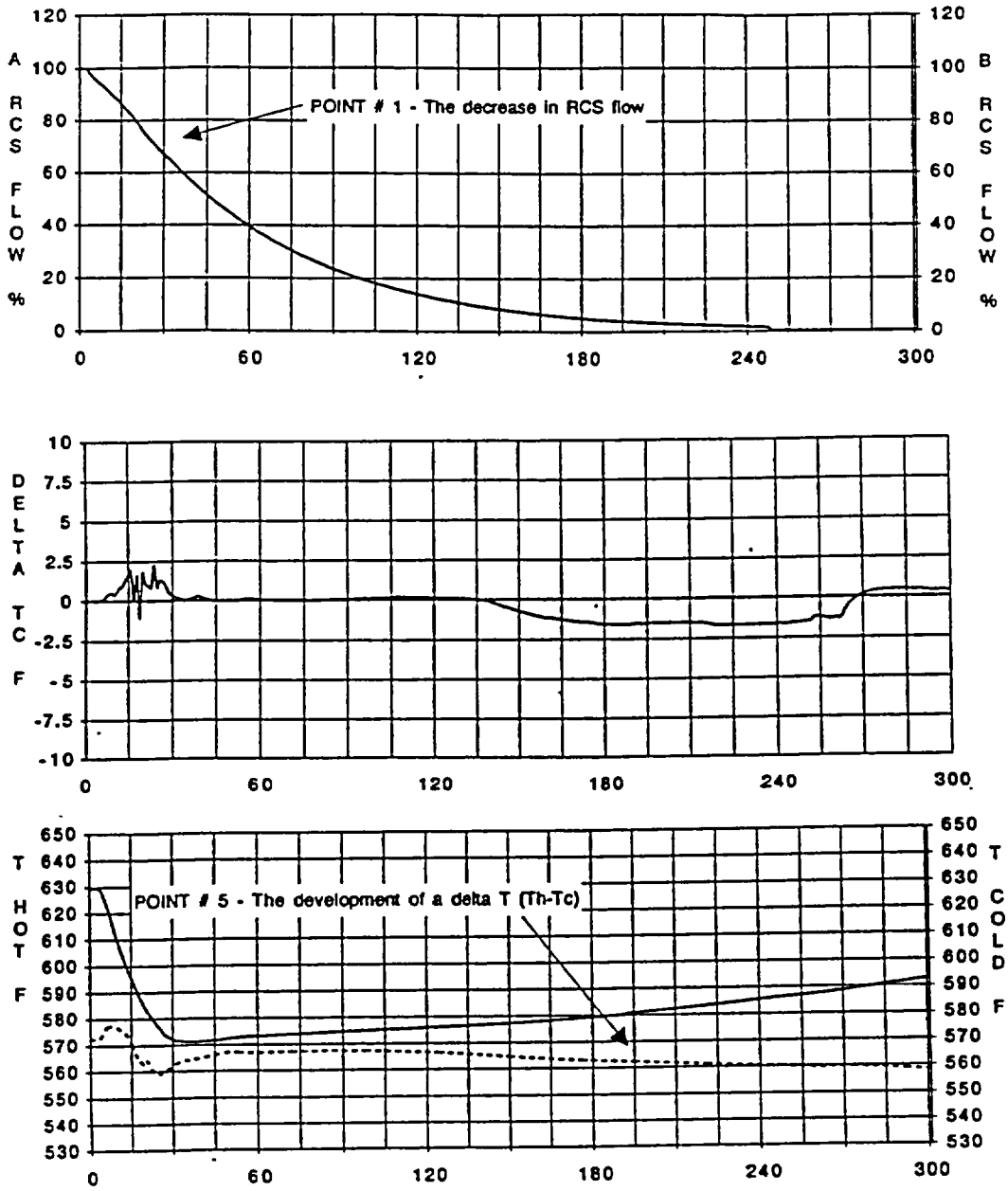

A OTSG LEVEL in
B OTSG LEVEL in
80
70
60
50
40
30
20
10
0
0
60
120
180
240
300
POINT # 4 - The Increase in OTSG levels


RX DEMAND %
RX POWER %
0
20
40
60
80
100
120
0
60
120
180
240
300


GROUP 7 %
GROUP 6 %
POINT # 2 - Determine the reactor trip signal
0
10
20
30
40
50
60
70
80
90
100
0
60
120
180
240
300

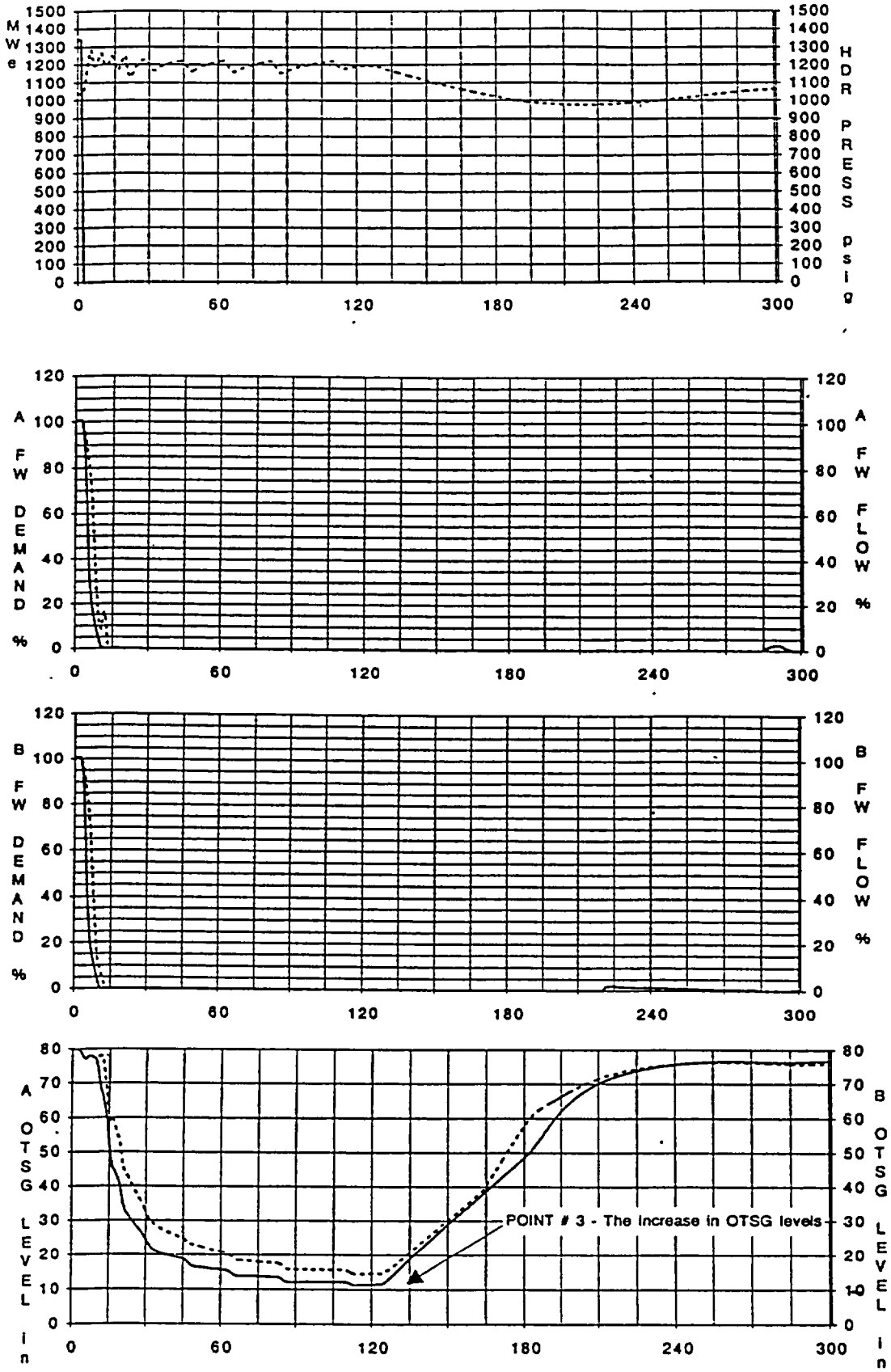

T AVG F
PZR LEVEL in
POINT # 3 - The increase in pzr level
530
540
550
560
570
580
590
600
610
620
630
640
650
0
50
100
150
200
250
300
350
400
0
60
120
180
240
300

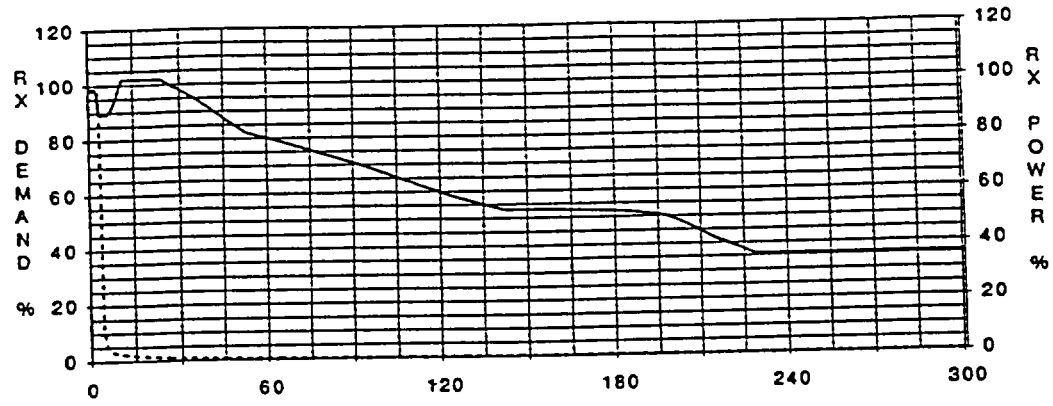

NR PRESS psig
WR PRESS psig
1500
1600
1700
1800
1900
2000
2100
2200
2300
2400
2500
0
500
1000
1500
2000
2500
0
60
120
180
240
300

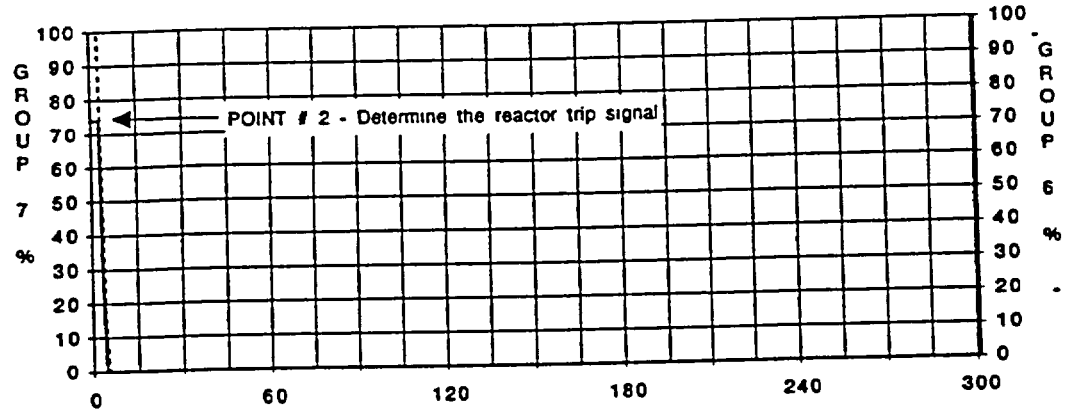

A RCS FLOW %
B RCS FLOW %
POINT # 1 - The decrease in RCS flow
DELTA TC F
T HOT F
T COLD F
POINT # 5 - The development of a delta T (Th-Tc)
0
60
120
180
240
300

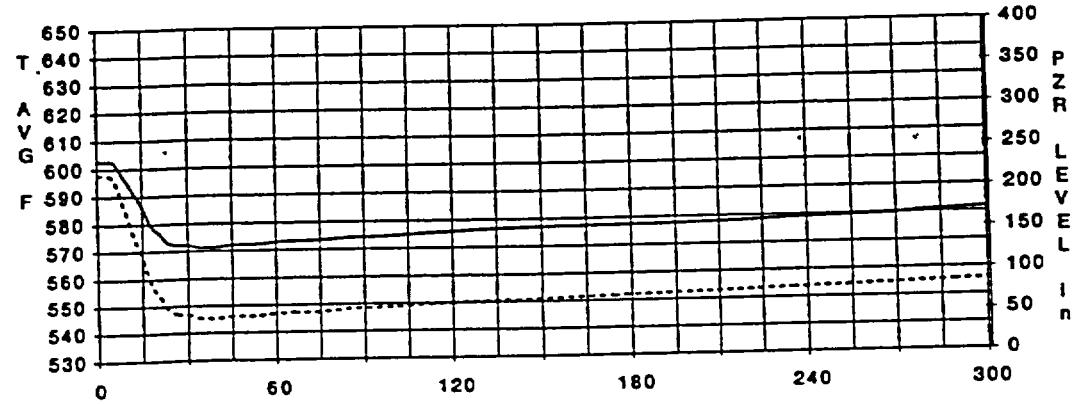
Transient 31

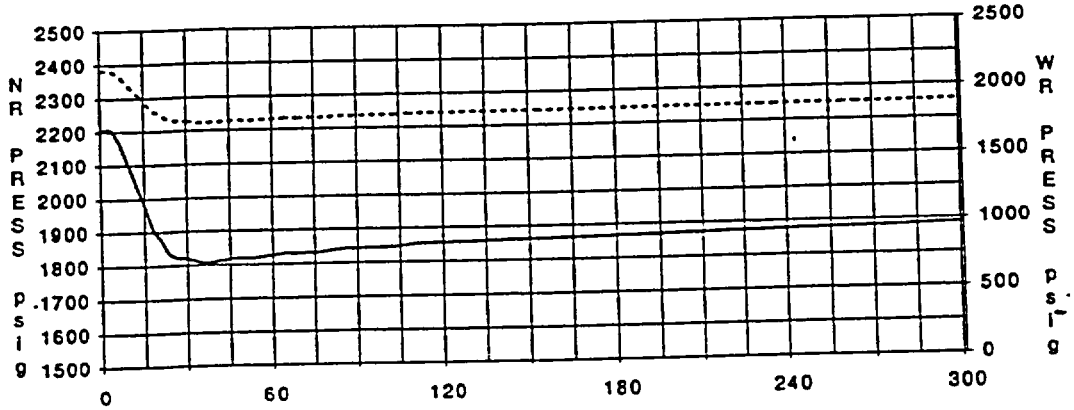
Loss of All AC (Station Blackout)

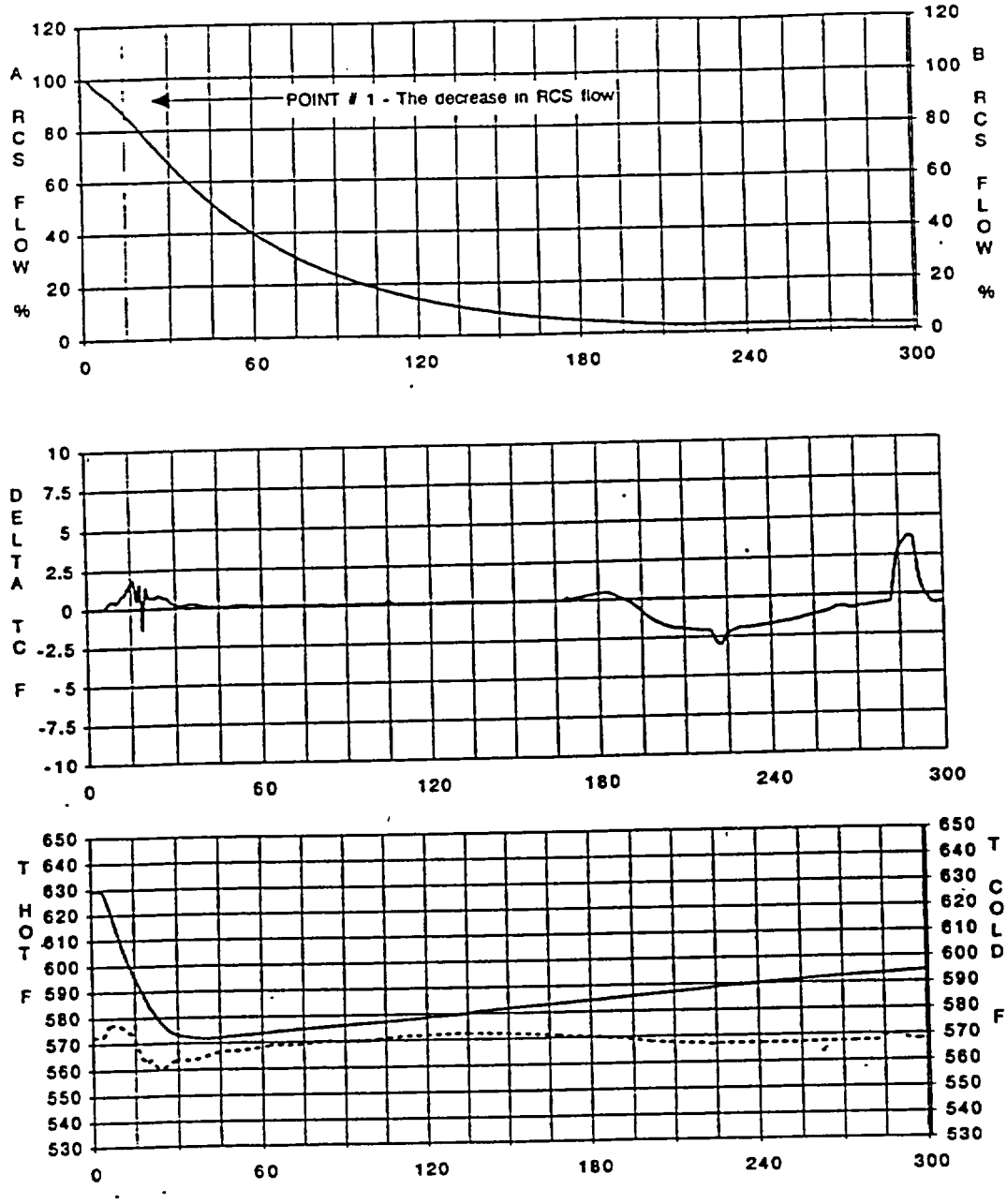

MWe
HDR PRESS psig
A FW DEMAND %
A FW FLOW %
B FW DEMAND %
B FW FLOW %
A OTSG LEVEL in
B OTSG LEVEL in
POINT # 3 - The Increase in OTSG levels


RX DEMAND %
RX POWER %
0
20
40
60
80
100
120
0
60
120
180
240
300


GROUP 7 %
GROUP 6 %
POINT # 2 - Determine the reactor trip signal
0
10
20
30
40
50
60
70
80
90
100
0
60
120
180
240
300

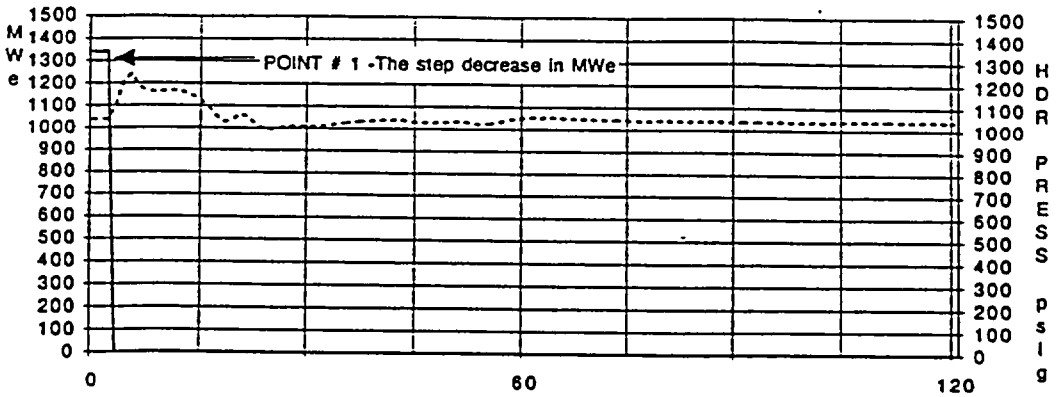

T AVG F
PZR LEVEL in
530
540
550
560
570
580
590
600
610
620
630
640
650
0
50
100
150
200
250
300
350
400
0
60
120
180
240
300

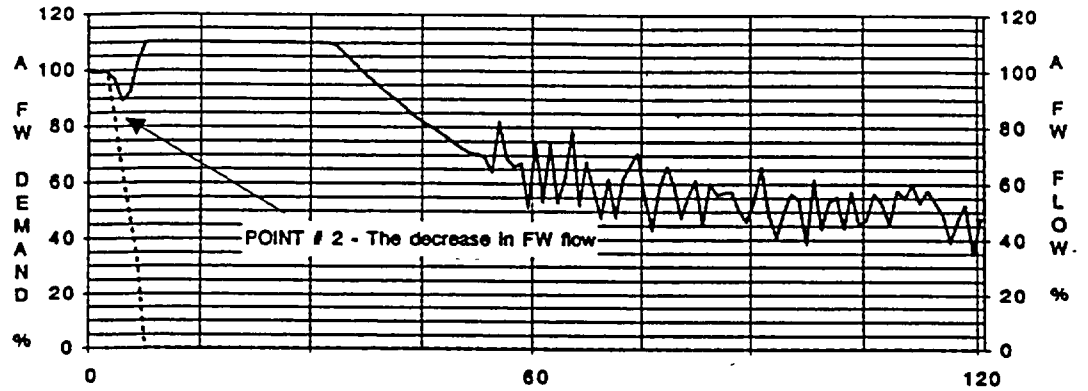

NR PRESS psig
WR PRESS psig
1500
1600
1700
1800
1900
2000
2100
2200
2300
2400
2500
0
500
1000
1500
2000
2500
0
60
120
180
240
300

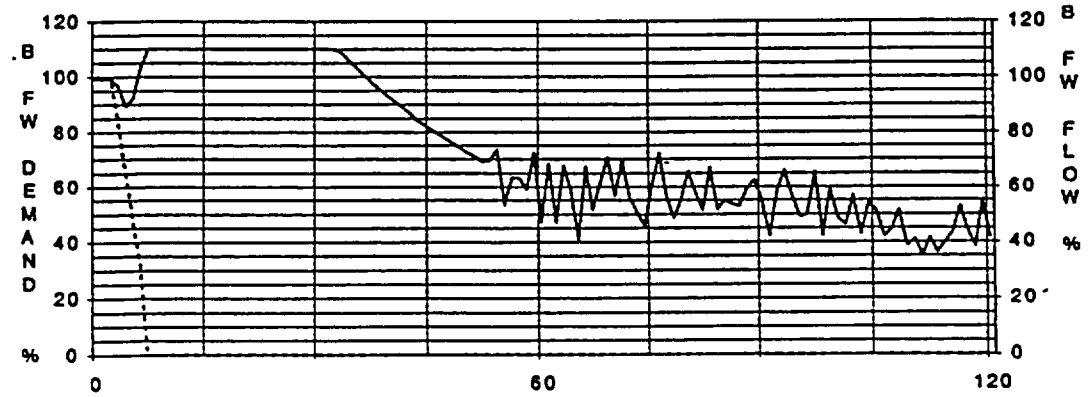

A RCS FLOW %
B RCS FLOW %
POINT # 1 - The decrease in RCS flow
DELTA TC F
T HOT F
T COLD F
0
60
120
180
240
300

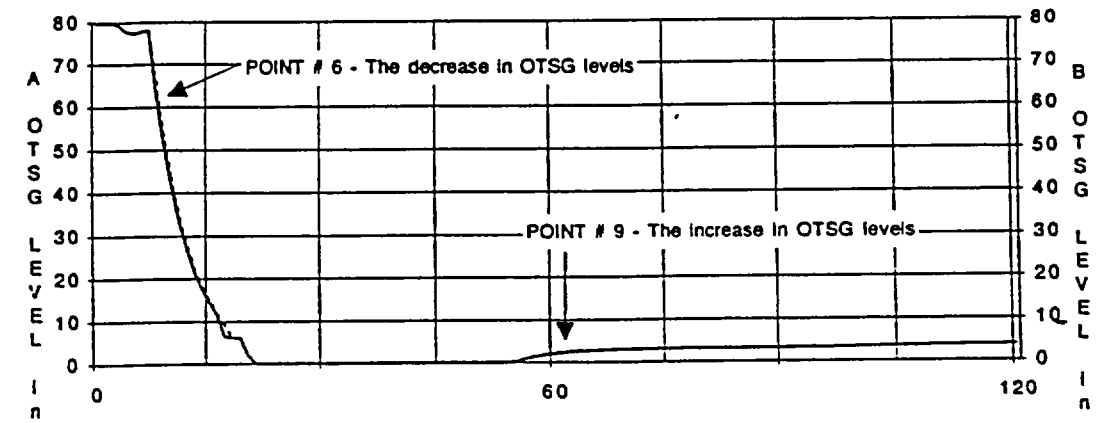
Transient 32

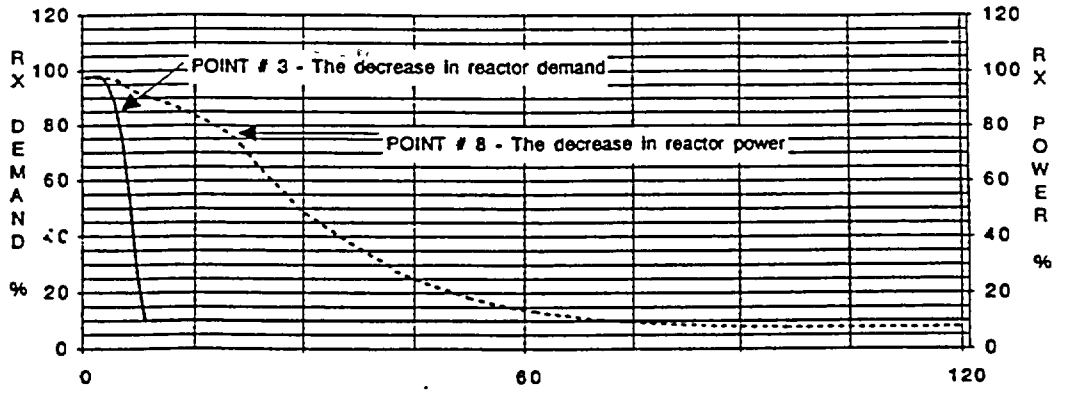
Anticipated Transient Without Scram

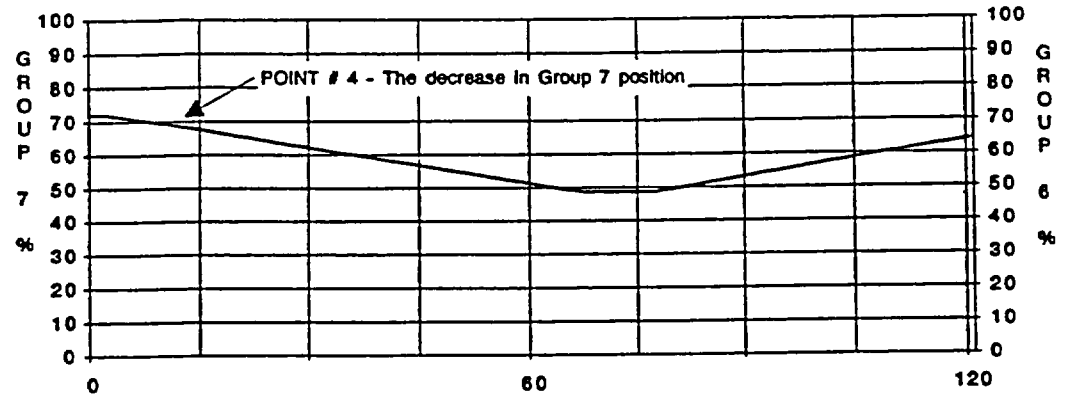

MWe
HDR PRESS psig
POINT # 1 -The step decrease in MWe
0
60
120

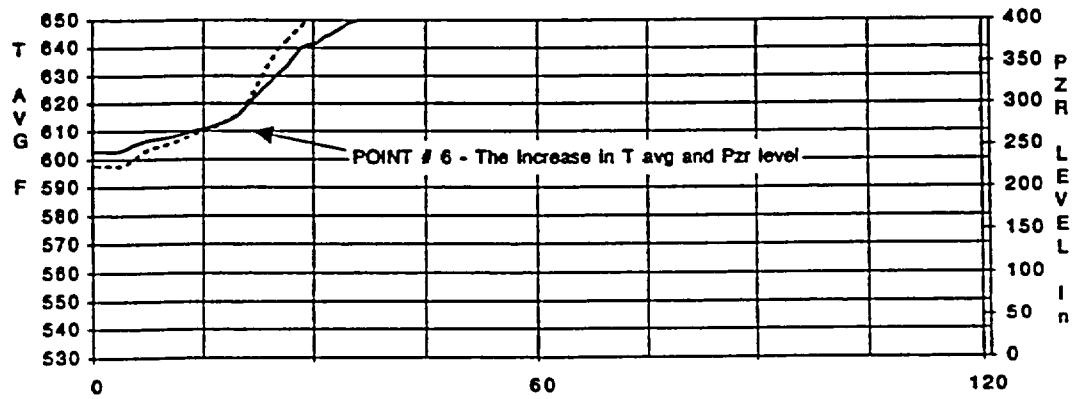

A FW DEMAND %
A FW FLOW %
POINT # 2 - The decrease in FW flow
0
60
120

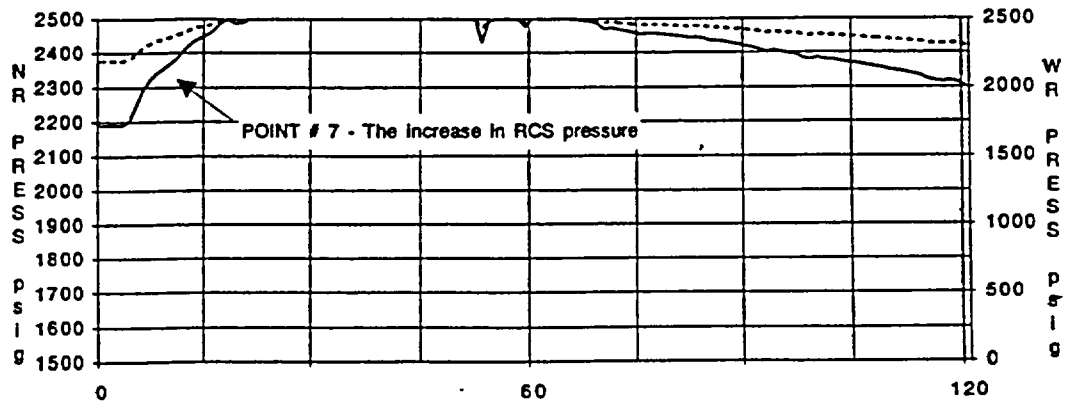

B FW DEMAND %
B FW FLOW %
0
60
120

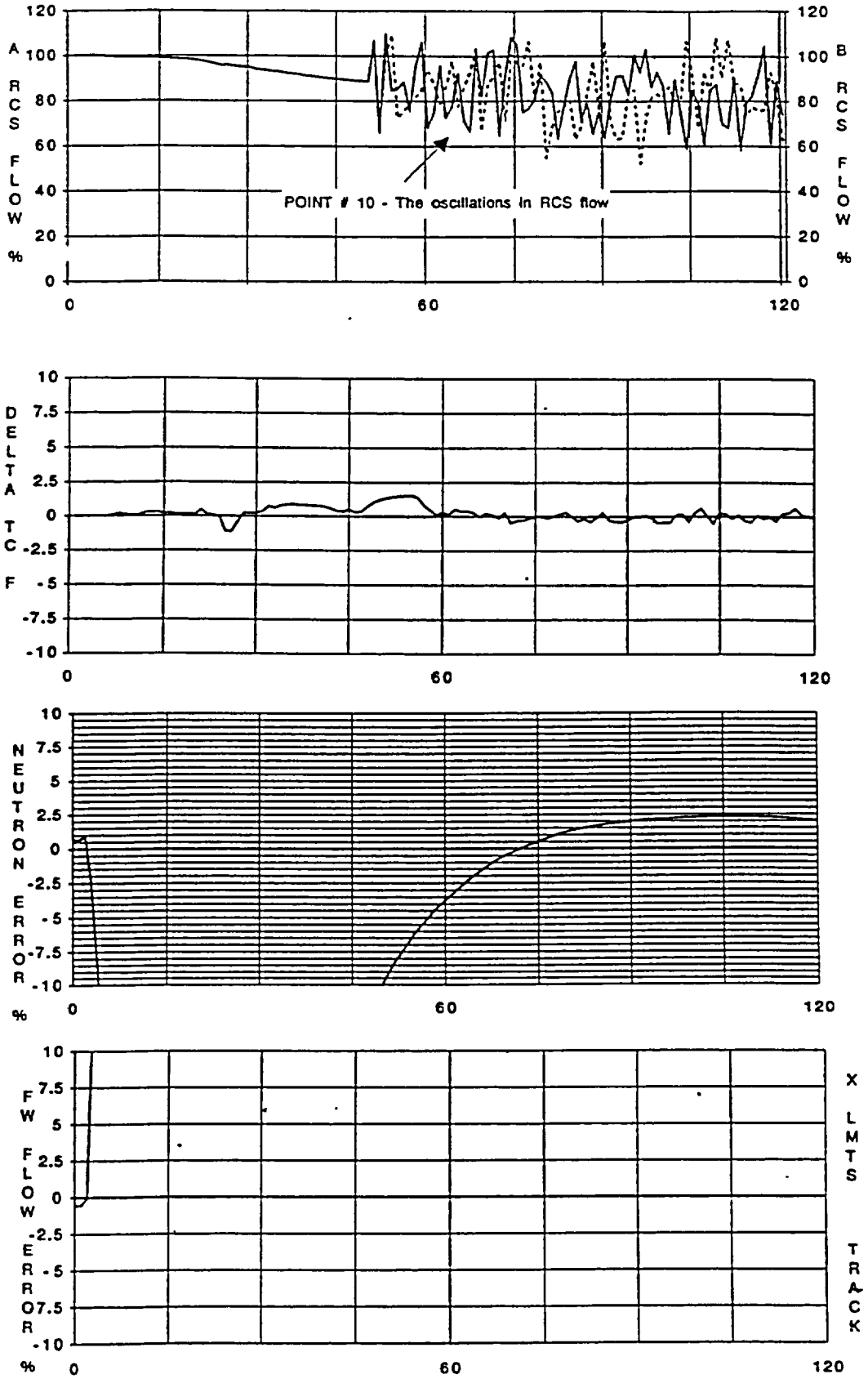

A OTSG LEVEL in
B OTSG LEVEL in
POINT # 6 - The decrease in OTSG levels
POINT # 9 - The increase in OTSG levels
0
60
120

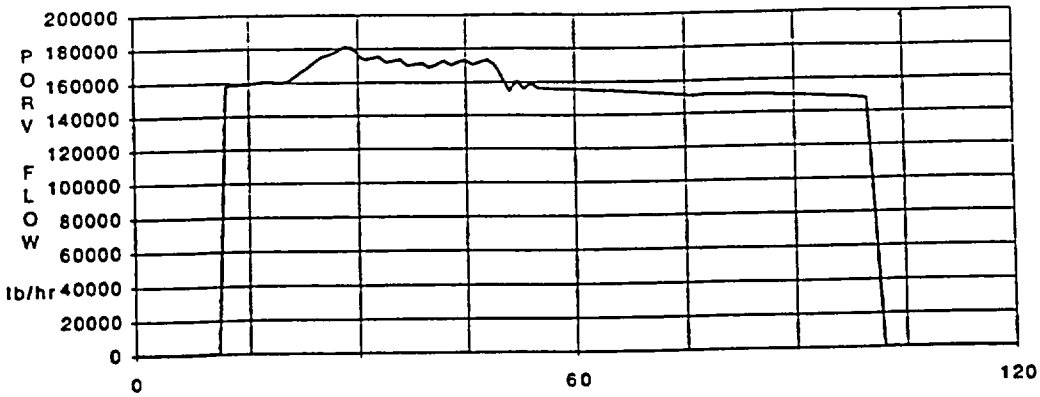

POINT # 3 - The decrease in reactor demand
POINT # 8 - The decrease in reactor power
RX DEMAND %
RX POWER %
120
100
80
60
40
20
0
0
60
120

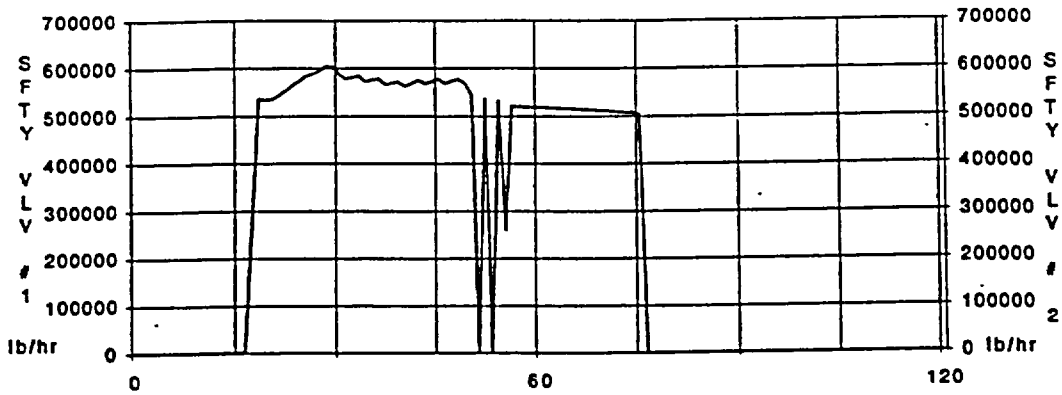

POINT # 4 - The decrease in Group 7 position
GROUP 7 %
GROUP 6 %
100
90
80
70
60
50
40
30
20
10
0
0
60
120


POINT # 6 - The increase in T avg and Pzr level
T AVG F
PZR LEVEL in
650
640
630
620
610
600
590
580
570
560
550
540
530
400
350
300
250
200
150
100
50
0
0
60
120


POINT # 7 - The increase in RCS pressure
NR PRESS psig
WR PRESS psig
2500
2400
2300
2200
2100
2000
1900
1800
1700
1600
1500
2500
2000
1500
1000
500
0
0
60
120

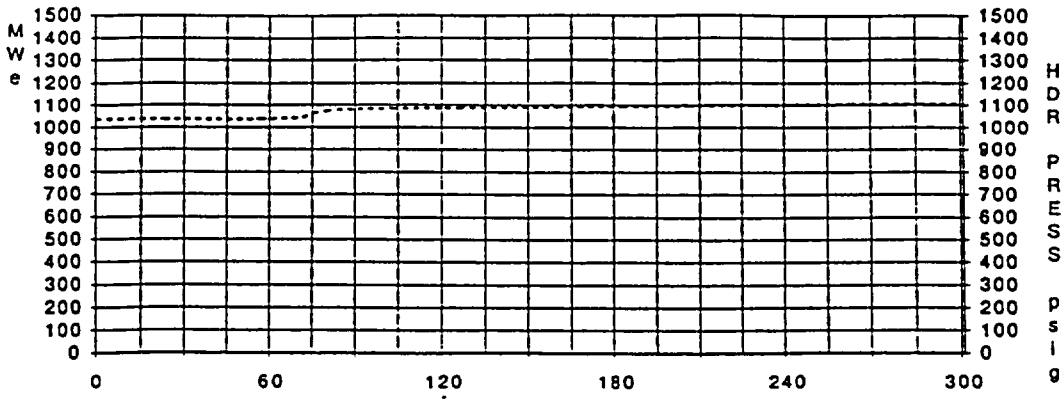

A RCS FLOW %
B RCS FLOW %
POINT # 10 - The oscillations in RCS flow
DELTA TCF
NEUTRON ERROR %
FW FLOW ERROR %
X LMTS TRACK
0
60
120

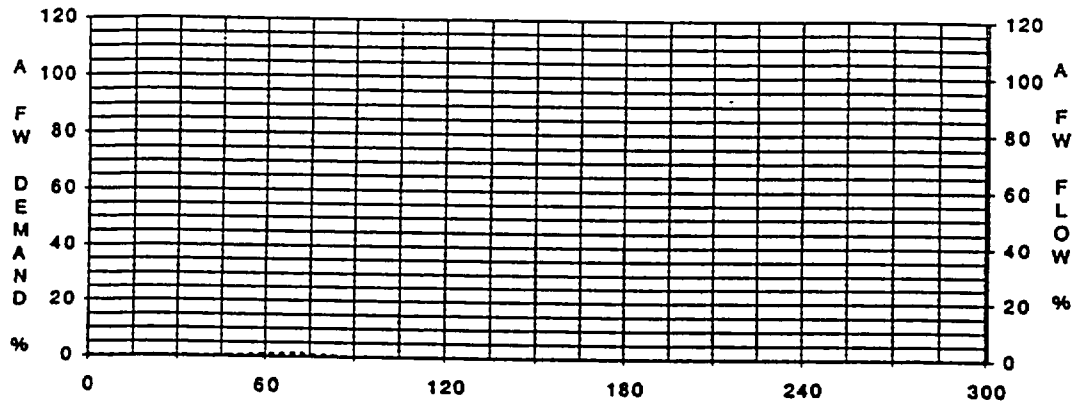

200000
180000
160000
140000
120000
100000
80000
60000
40000
20000
0
PORV FLOW lb/hr
0
60
120

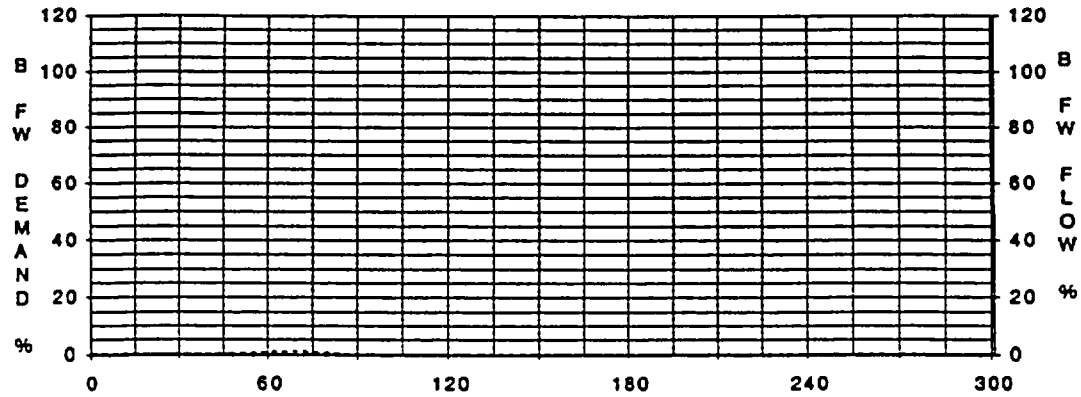

700000
600000
500000
400000
300000
200000
100000
0
SFTY VLV # 1 lb/hr
SFTY VLV # 2 lb/hr
0
60
120

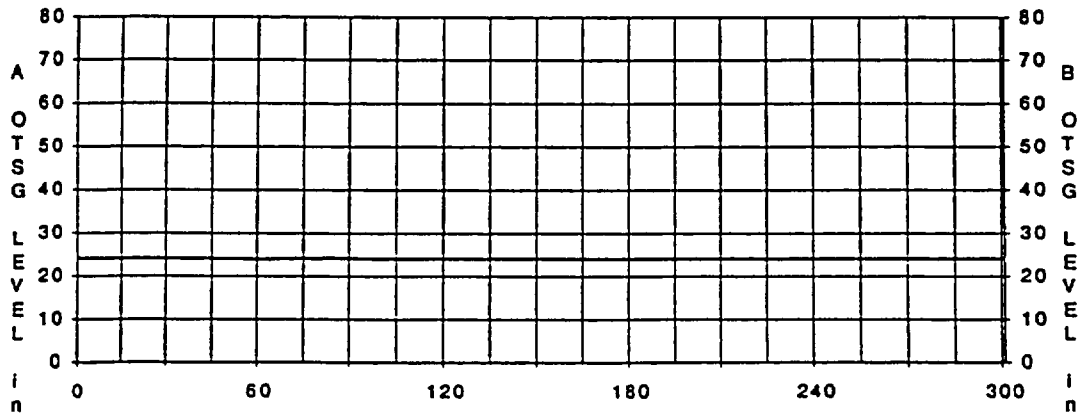
Transient 33

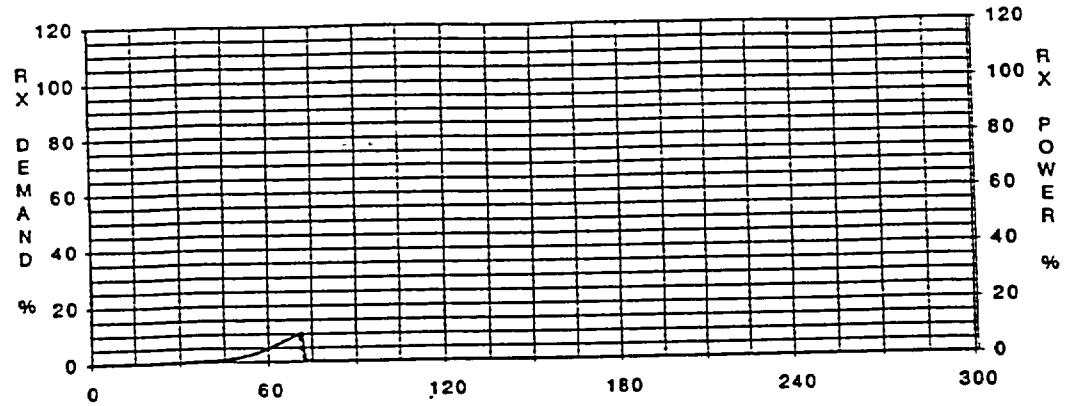
Continuous Rod Withdrawal (BOL)

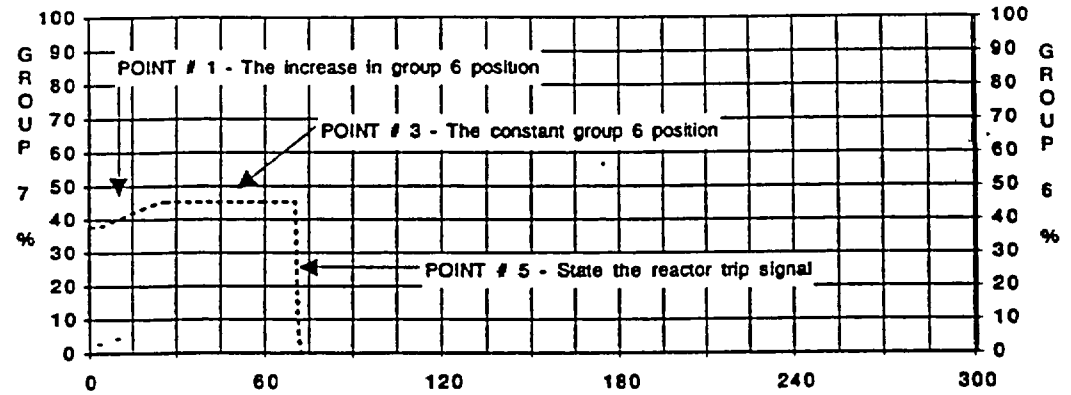

MWe
HDR PRESS psig
1500
1400
1300
1200
1100
1000
900
800
700
600
500
400
300
200
100
0
0
60
120
180
240
300

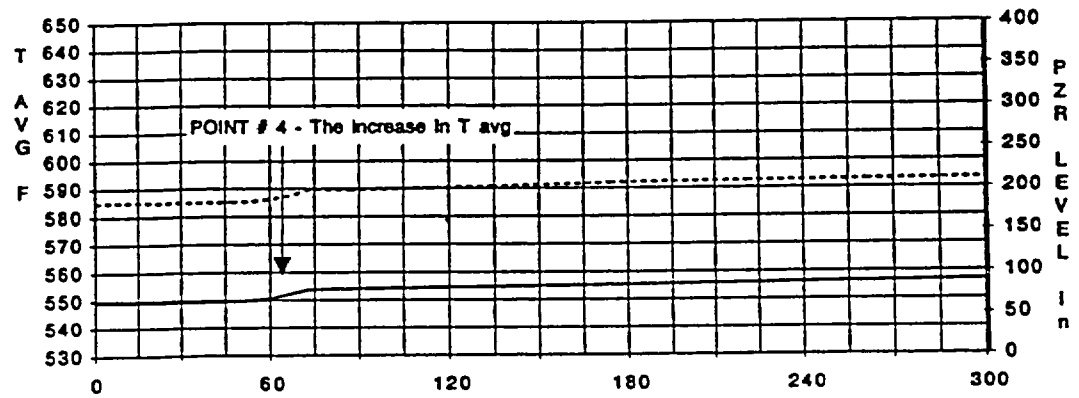

A FW DEMAND %
A FW FLOW %
120
100
80
60
40
20
0
0
60
120
180
240
300

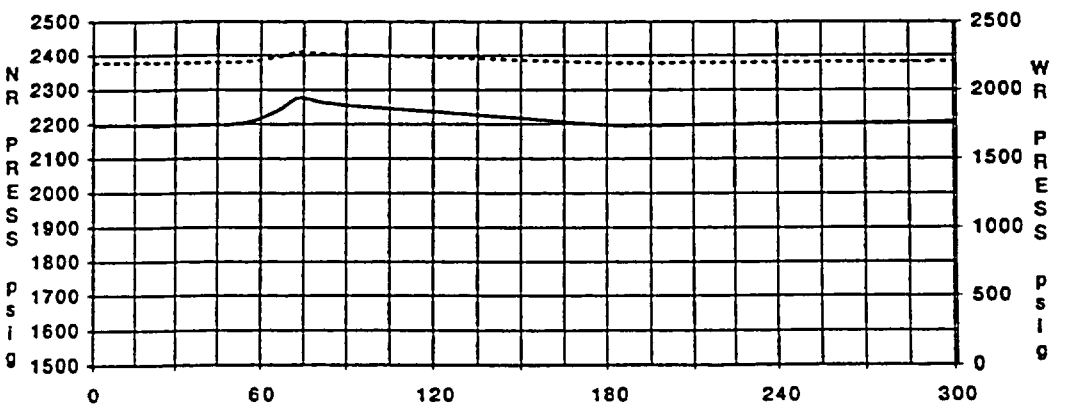

B FW DEMAND %
B FW FLOW %
120
100
80
60
40
20
0
0
60
120
180
240
300

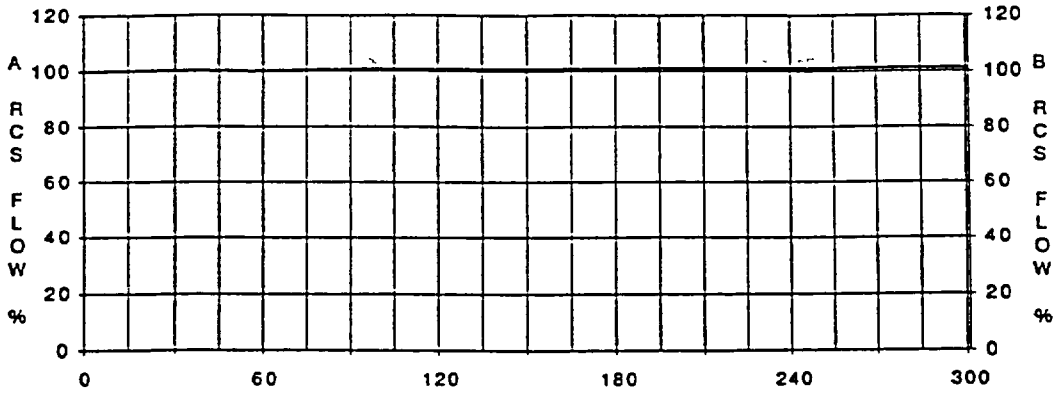

A OTSG LEVEL in
B OTSG LEVEL in
80
70
60
50
40
30
20
10
0
0
60
120
180
240
300

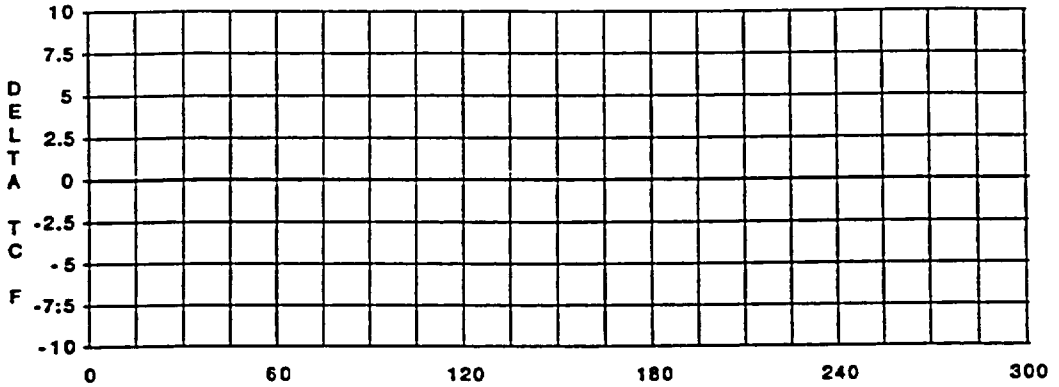

RX DEMAND %
RX POWER %
0
20
40
60
80
100
120
0
60
120
180
240
300

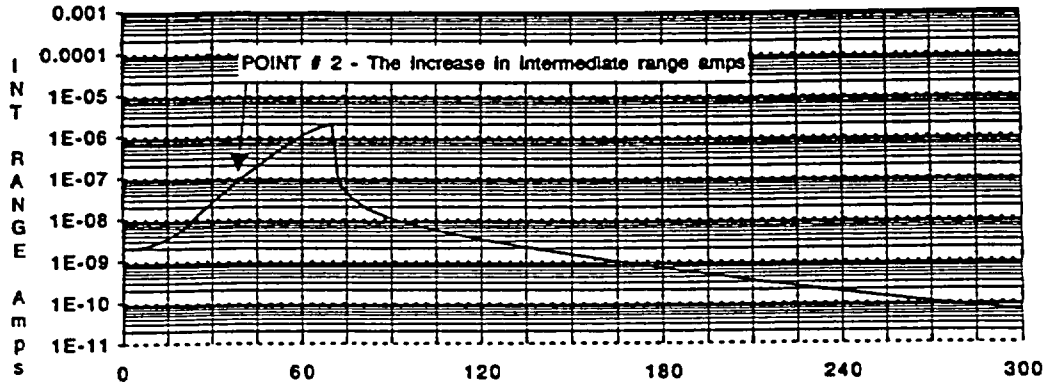

GROUP 7 %
GROUP 6 %
POINT # 1 - The increase in group 6 position
POINT # 3 - The constant group 6 position
POINT # 5 - State the reactor trip signal
0
10
20
30
40
50
60
70
80
90
100
0
60
120
180
240
300

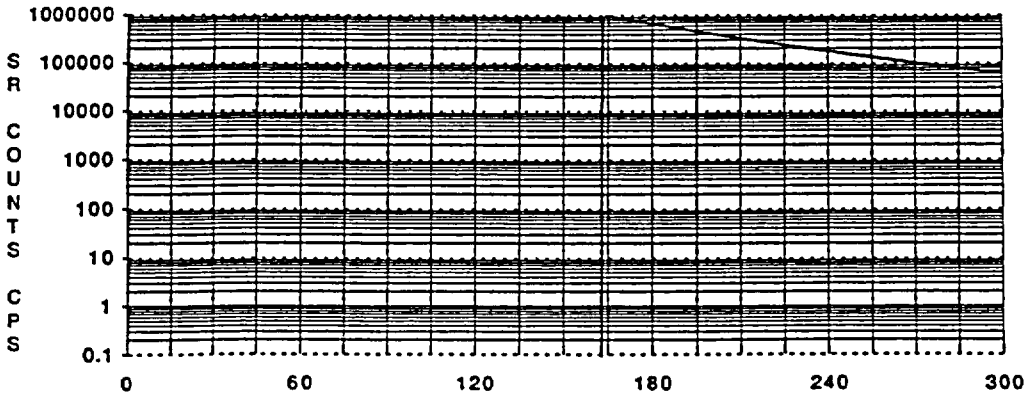

T AVG F
PZR LEVEL in
POINT # 4 - The increase in T avg
530
540
550
560
570
580
590
600
610
620
630
640
650
0
50
100
150
200
250
300
350
400
0
60
120
180
240
300


NR PRESS psig
WR PRESS psig
1500
1600
1700
1800
1900
2000
2100
2200
2300
2400
2500
0
500
1000
1500
2000
2500
0
60
120
180
240
300


A RCS FLOW %
B RCS FLOW %
120
100
80
60
40
20
0
0
60
120
180
240
300

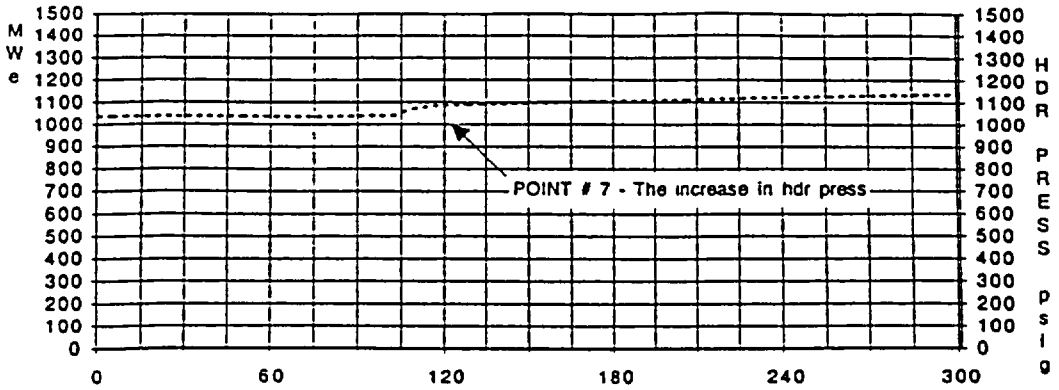

DELTA TCF
10
7.5
5
2.5
0
-2.5
-5
-7.5
-10
0
60
120
180
240
300

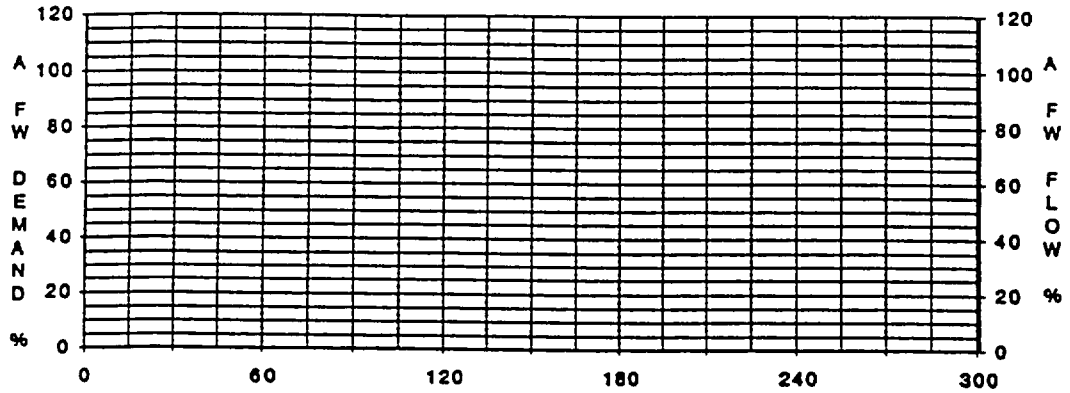

INT RANGE Amps
POINT # 2 - The increase in intermediate range amps
0.001
0.0001
1E-05
1E-06
1E-07
1E-08
1E-09
1E-10
1E-11
0
60
120
180
240
300

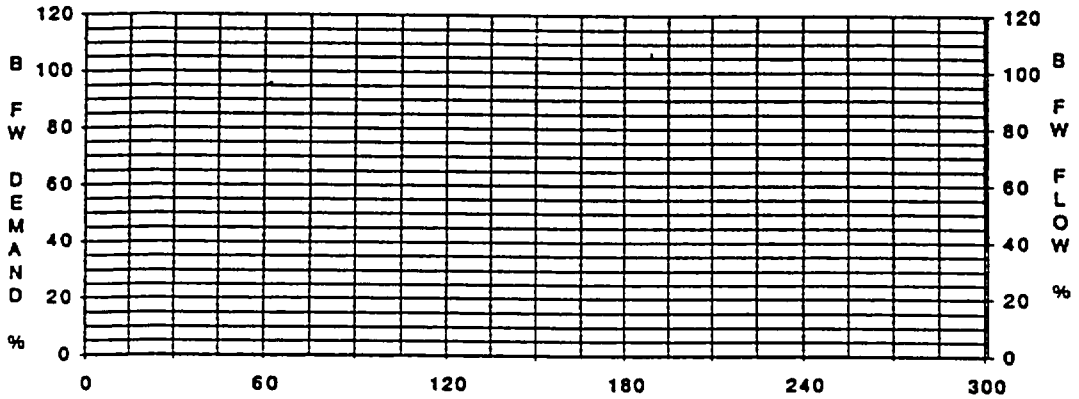

SR COUNTS CPS
1000000
100000
10000
1000
100
10
1
0.1
0
60
120
180
240
300

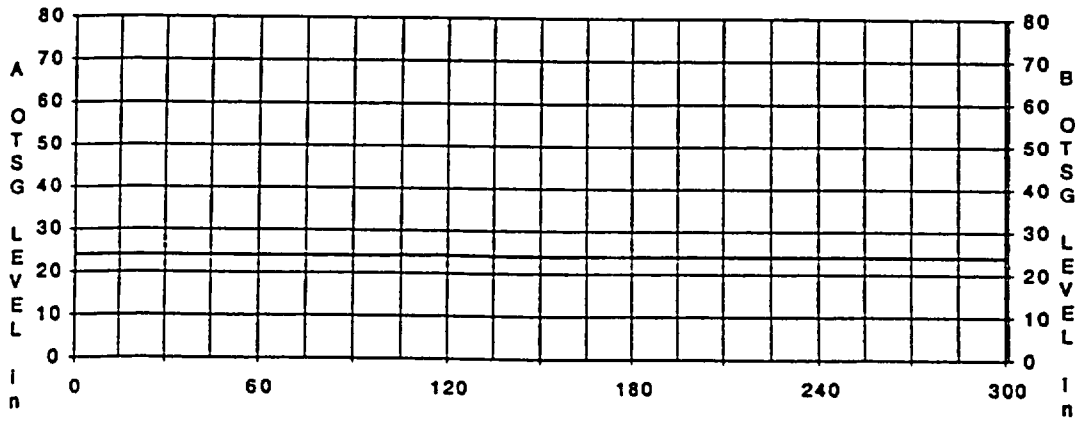
Transient 34

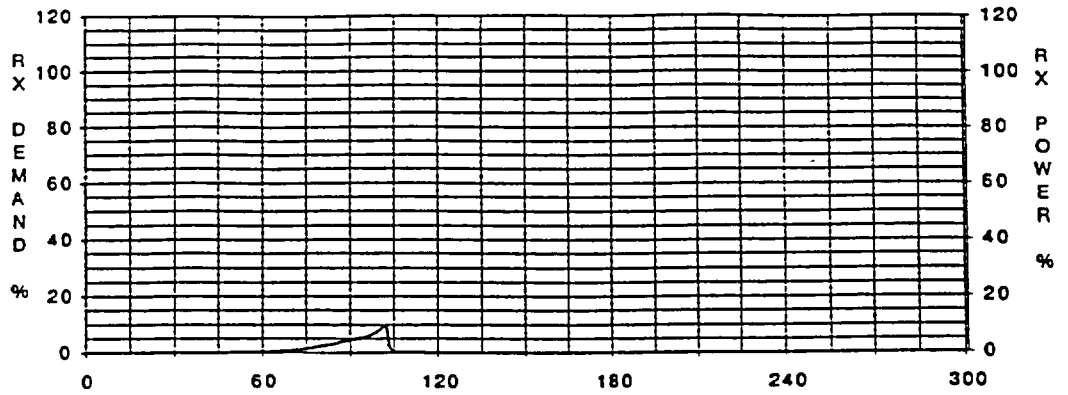
Continuous Rod Withdrawal

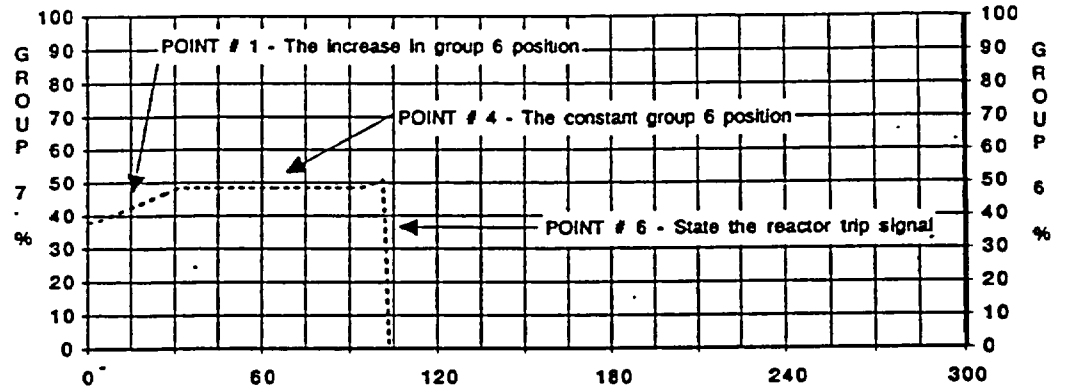

MWe
HDR PRESS psig
POINT # 7 - The increase in hdr press
0
60
120
180
240
300

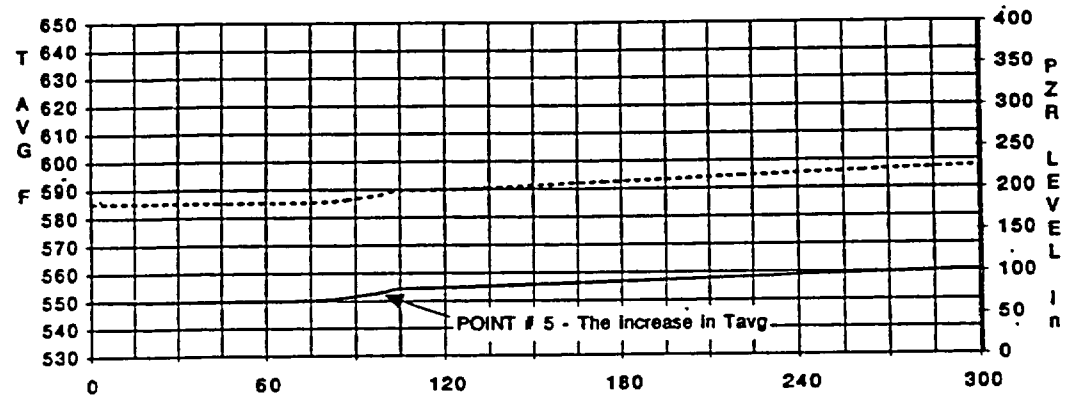

A FW DEMAND %
A FW FLOW %

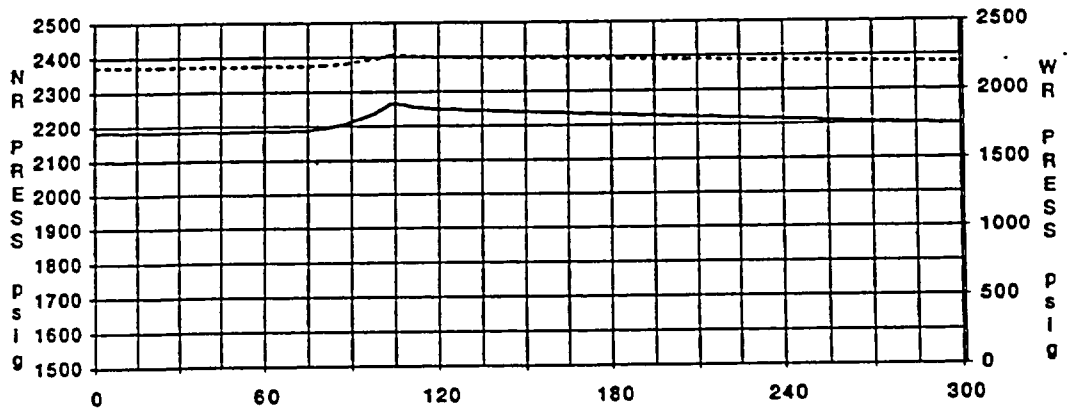

B FW DEMAND %
B FW FLOW %

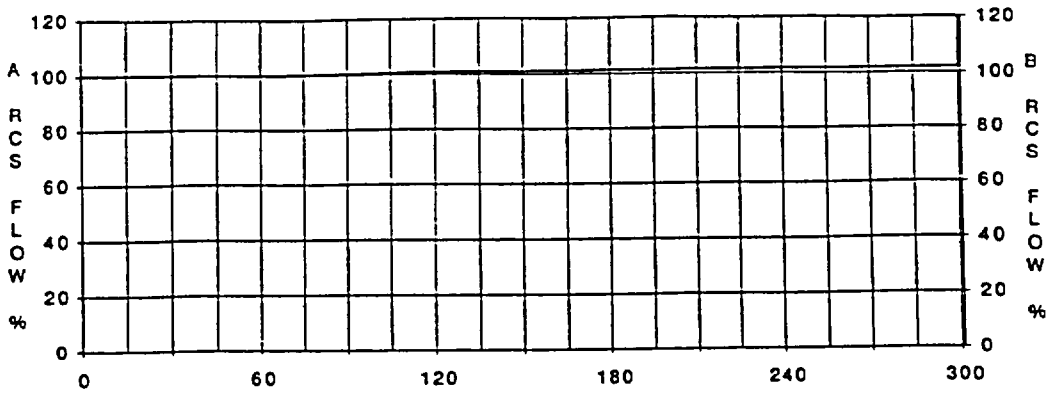

A OTSG LEVEL in
B OTSG LEVEL in

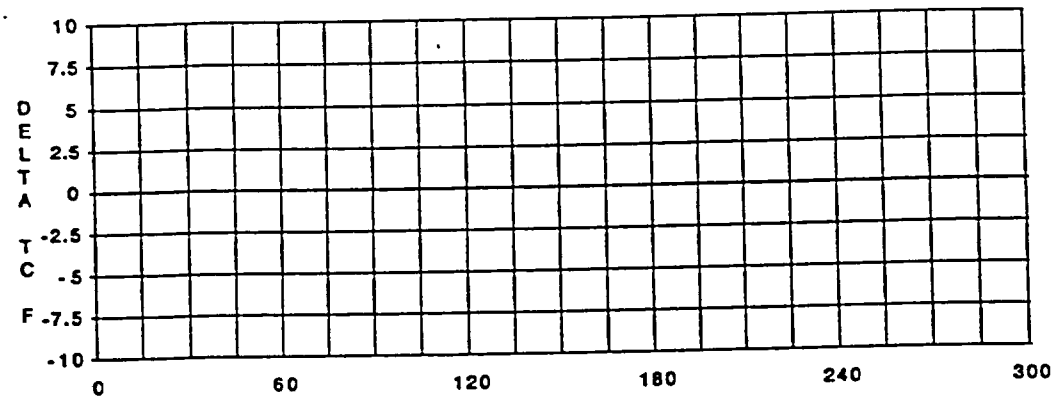

RX DEMAND %
RX POWER %
0
20
40
60
80
100
120
0
60
120
180
240
300

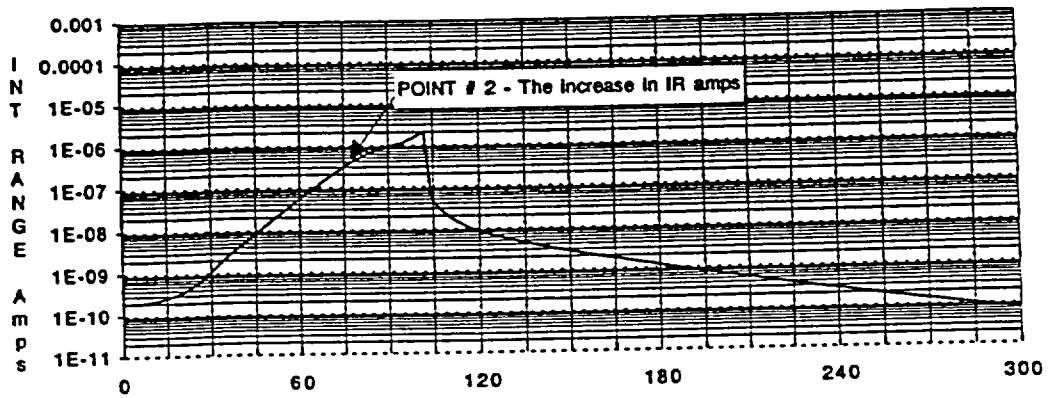

GROUP 7 %
GROUP 6 %
POINT # 1 - The increase in group 6 position
POINT # 4 - The constant group 6 position
POINT # 6 - State the reactor trip signal
0
10
20
30
40
50
60
70
80
90
100
0
60
120
180
240
300

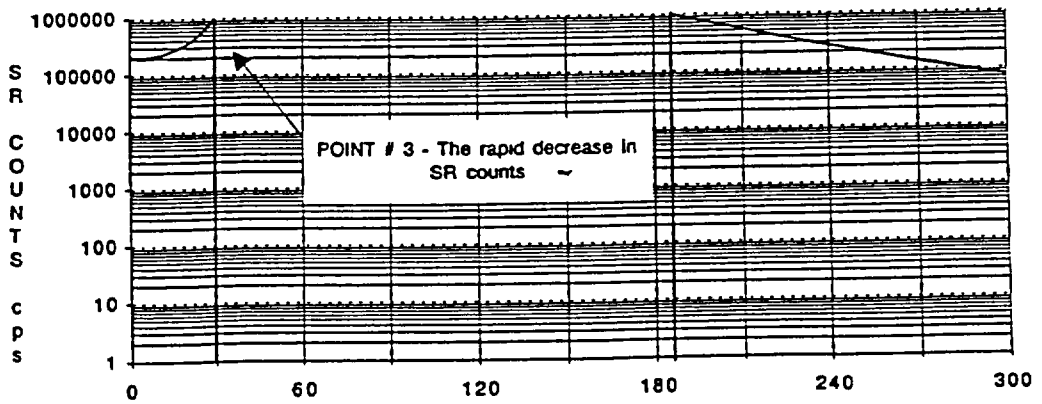

TAVG F
PZR LEVEL in
POINT # 5 - The increase in Tavg
530
540
550
560
570
580
590
600
610
620
630
640
650
0
50
100
150
200
250
300
350
400
0
60
120
180
240
300


NR PRESS psig
WR PRESS psig
1500
1600
1700
1800
1900
2000
2100
2200
2300
2400
2500
0
500
1000
1500
2000
2500
0
60
120
180
240
300


A RCS FLOW %
B RCS FLOW %
120
100
80
60
40
20
0
0
60
120
180
240
300

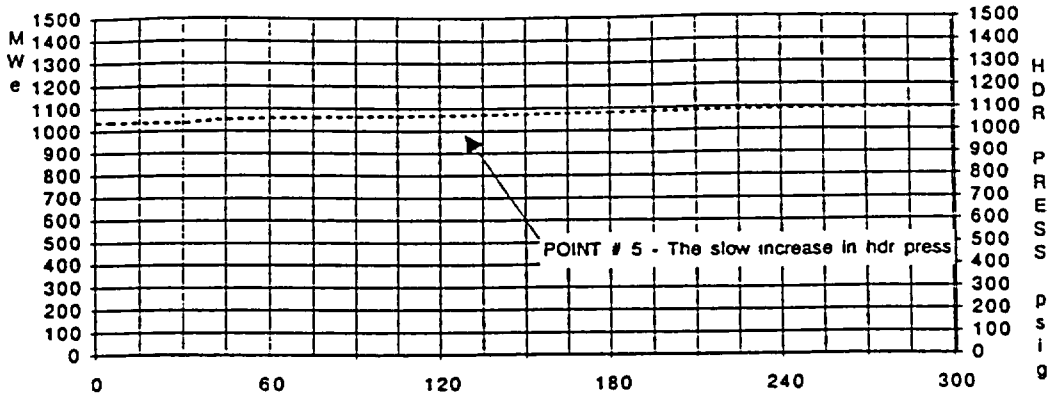

DELTA TC F
10
7.5
5
2.5
0
-2.5
-5
-7.5
-10
0
60
120
180
240
300

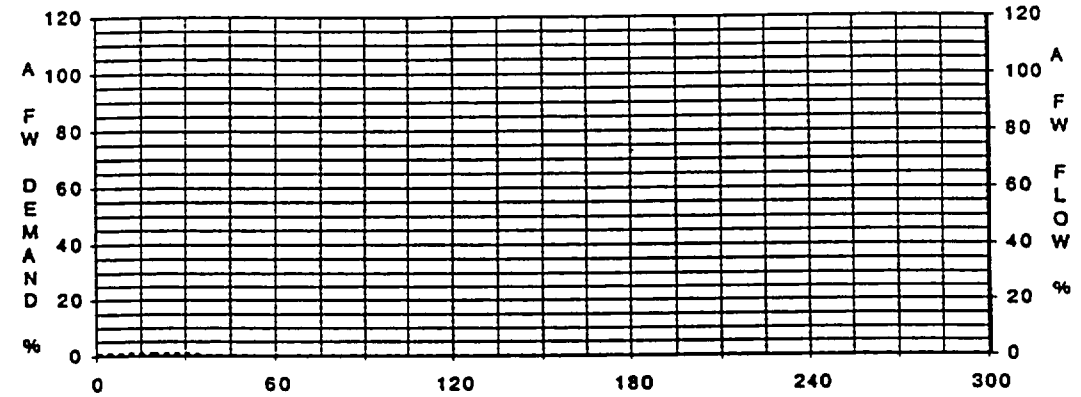

INT RANGE Amps
0.001
0.0001
1E-05
1E-06
1E-07
1E-08
1E-09
1E-10
1E-11
POINT # 2 - The increase in IR amps
0
60
120
180
240
300

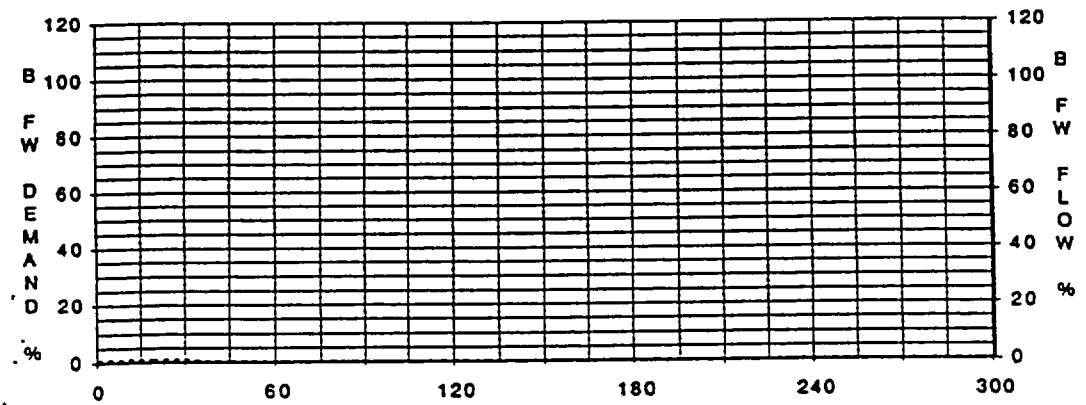

SR COUNTS cps
1000000
100000
10000
1000
100
10
1
POINT # 3 - The rapid decrease in SR counts
0
60
120
180
240
300

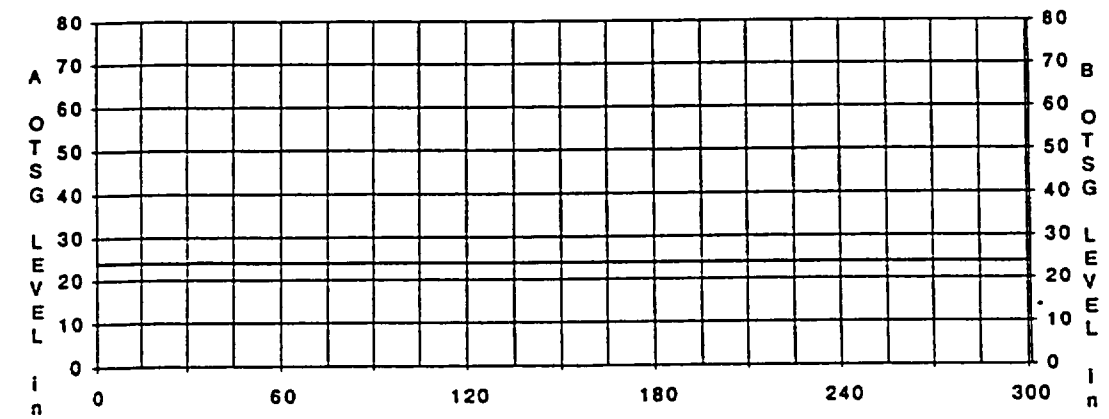
Transient 35

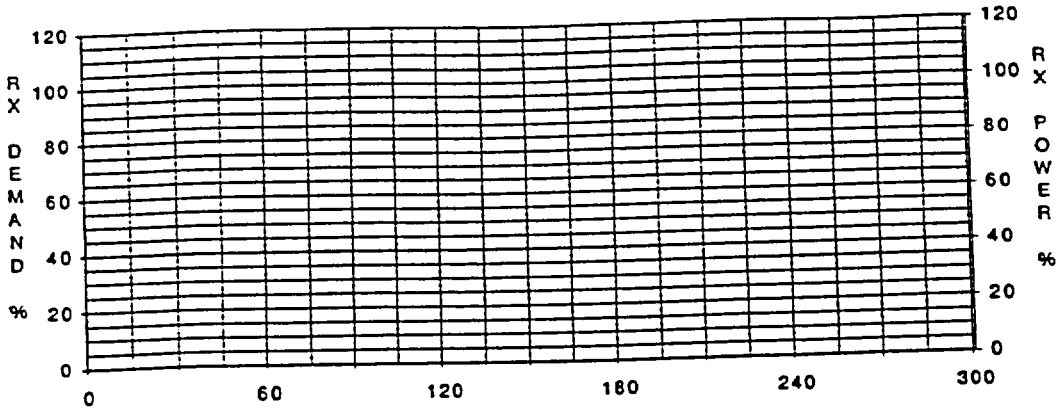
Ejected Rod - 0% Power

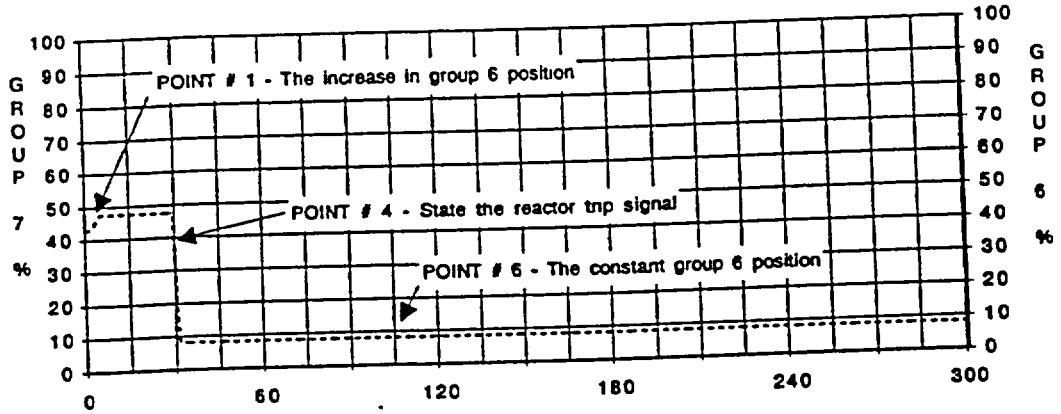

MWe
HDR PRESS psig
POINT # 5 - The slow increase in hdr press
0
60
120
180
240
300

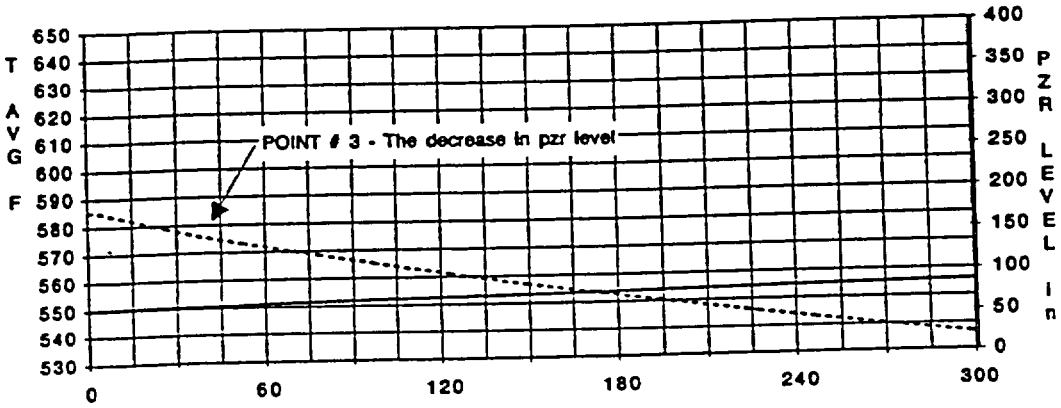

A FW DEMAND %
A FW FLOW %
0
60
120
180
240
300

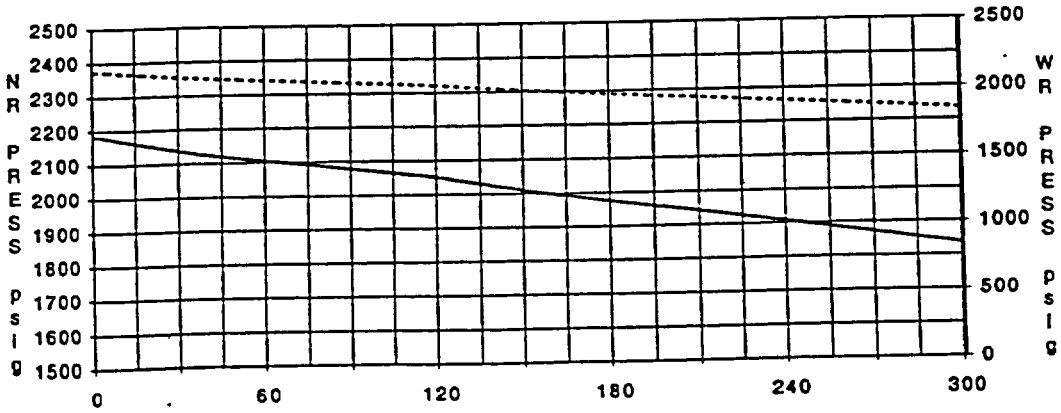

B FW DEMAND %
B FW FLOW %
0
60
120
180
240
300

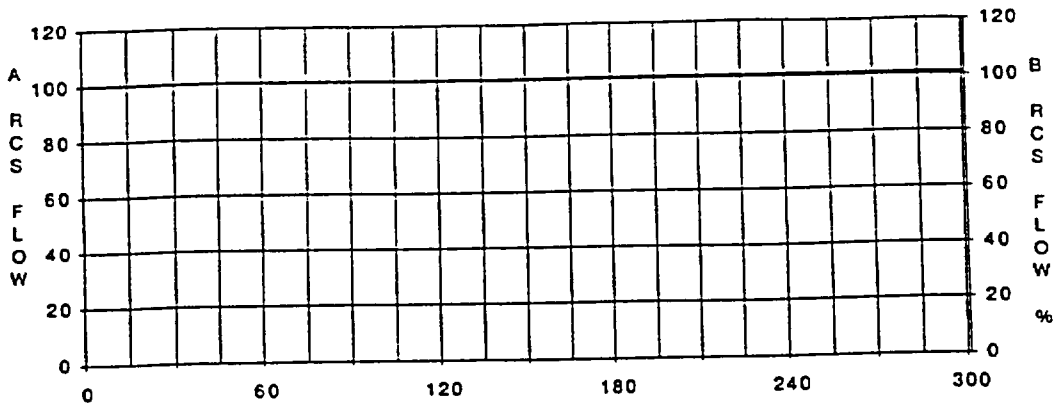

A OTSG LEVEL in
B OTSG LEVEL in
0
60
120
180
240
300

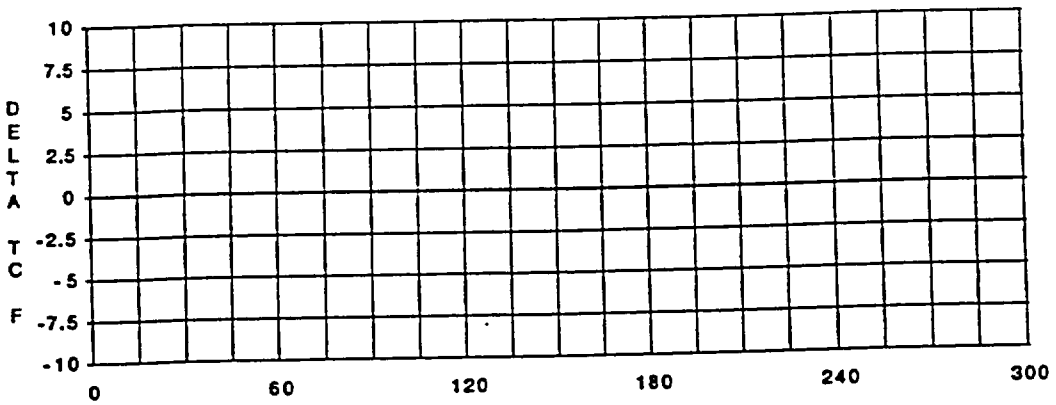

RX DEMAND %
RX POWER %
0
20
40
60
80
100
120
0
60
120
180
240
300

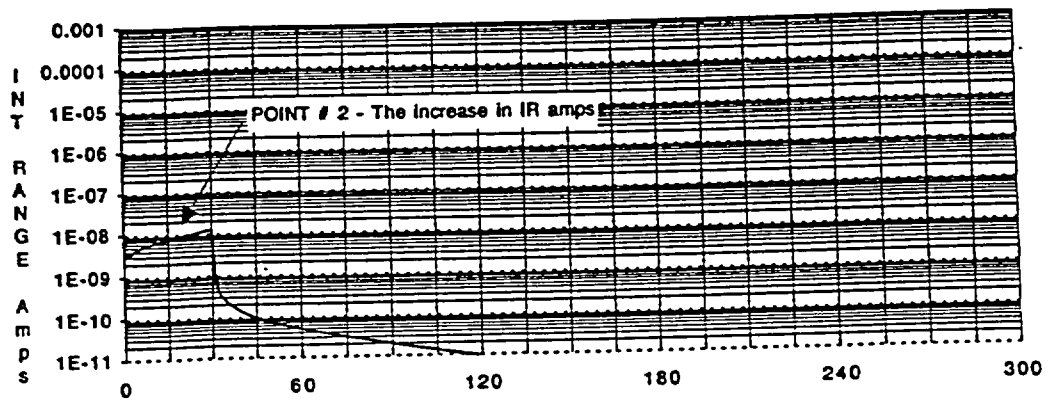

GROUP 7 %
GROUP 6 %
POINT # 1 - The increase in group 6 position
POINT # 4 - State the reactor trip signal
POINT # 6 - The constant group 6 position
0
10
20
30
40
50
60
70
80
90
100
0
60
120
180
240
300

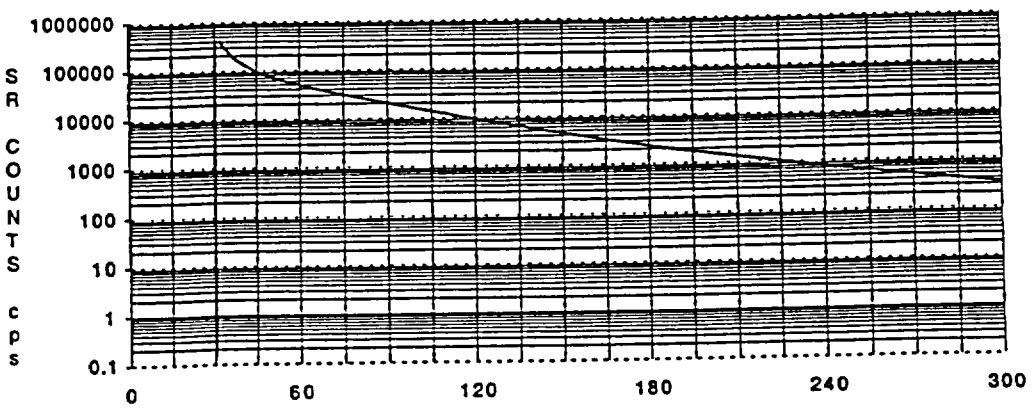

T AVG F
PZR LEVEL in
POINT # 3 - The decrease in pzr level
530
540
550
560
570
580
590
600
610
620
630
640
650
0
50
100
150
200
250
300
350
400
0
60
120
180
240
300


NR PRESS psig
WR PRESS psig
1500
1600
1700
1800
1900
2000
2100
2200
2300
2400
2500
0
500
1000
1500
2000
2500
0
60
120
180
240
300


A RCS FLOW
B RCS FLOW %
120
100
80
60
40
20
0
0
60
120
180
240
300

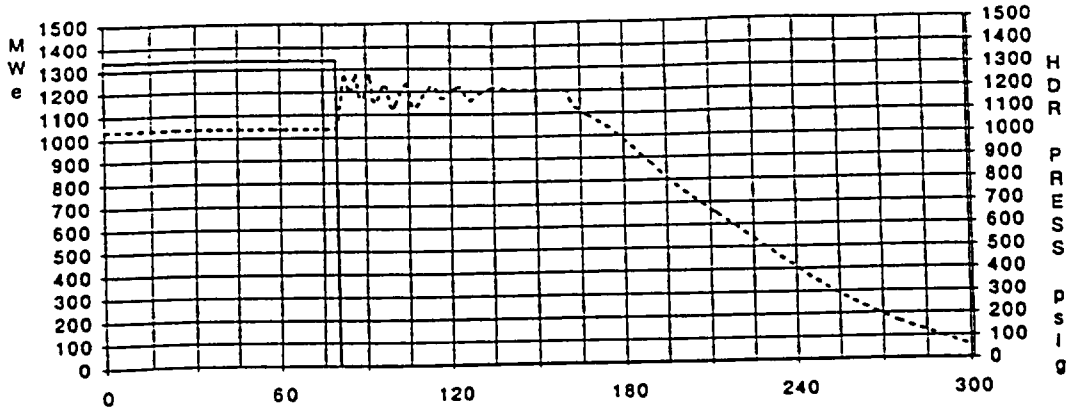

DELTA TCF
10
7.5
5
2.5
0
-2.5
-5
-7.5
-10
0
60
120
180
240
300

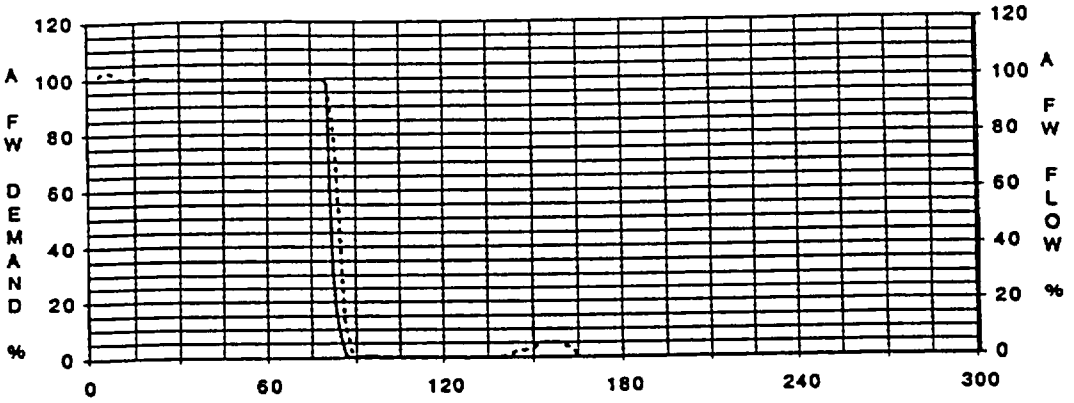

INT RANGE Amps
0.001
0.0001
1E-05
1E-06
1E-07
1E-08
1E-09
1E-10
1E-11
POINT # 2 - The increase in IR amps
0
60
120
180
240
300

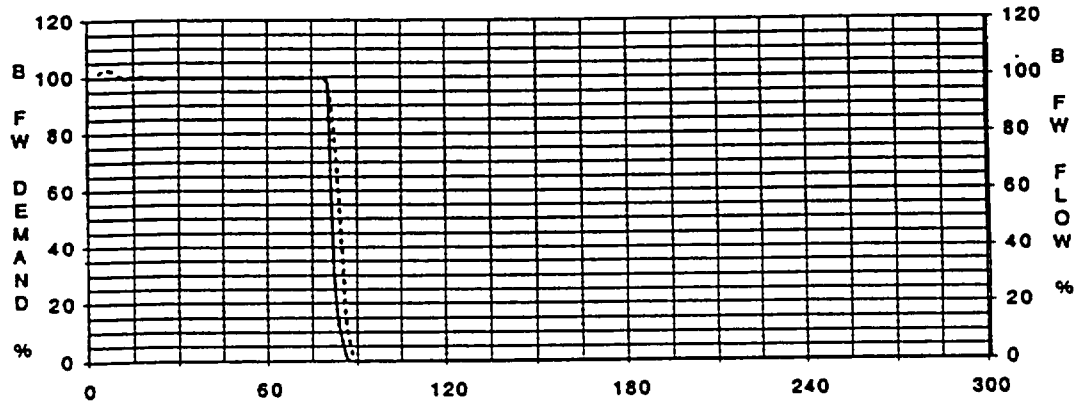

SR COUNTS cps
1000000
100000
10000
1000
100
10
1
0.1
0
60
120
180
240
300

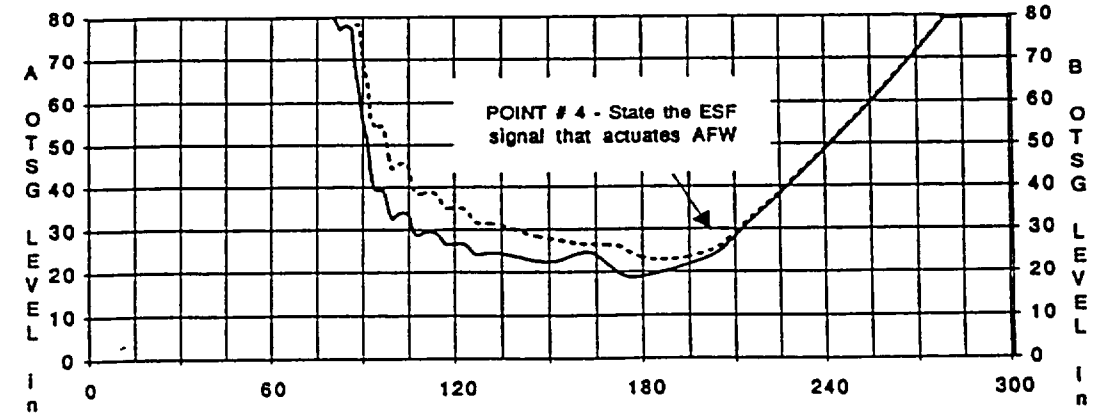
Transient 36

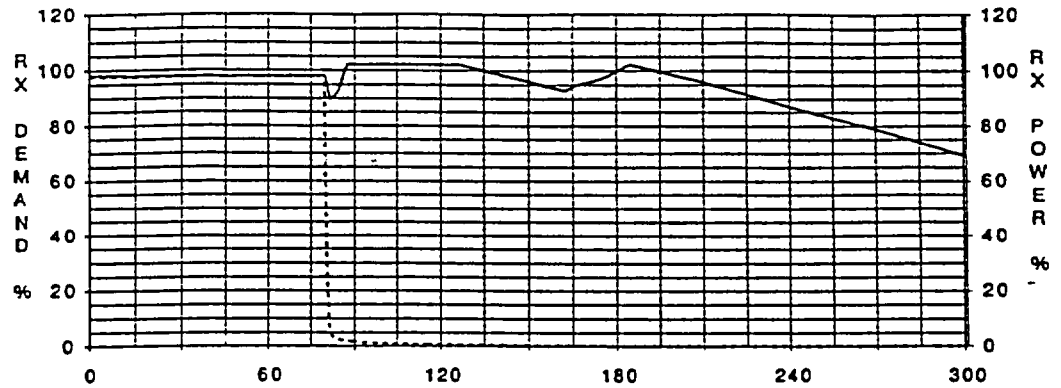
Ejected Rod - 100% Power

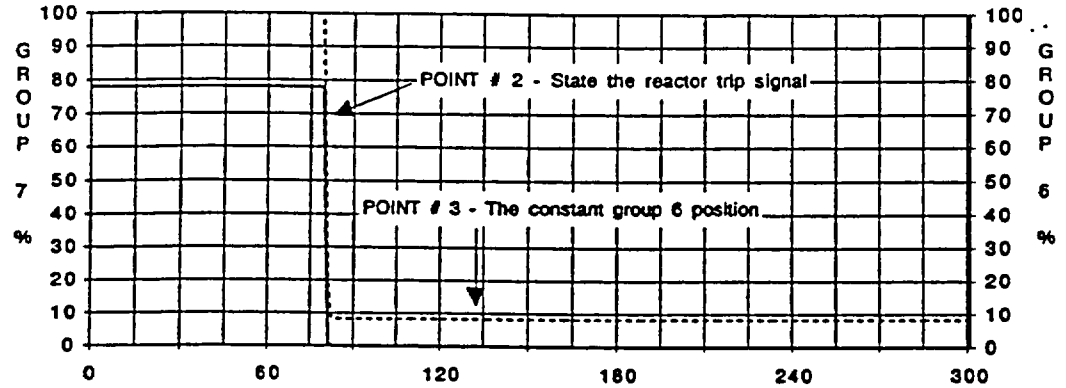

MWe
HDR PRESS psig
0
60
120
180
240
300

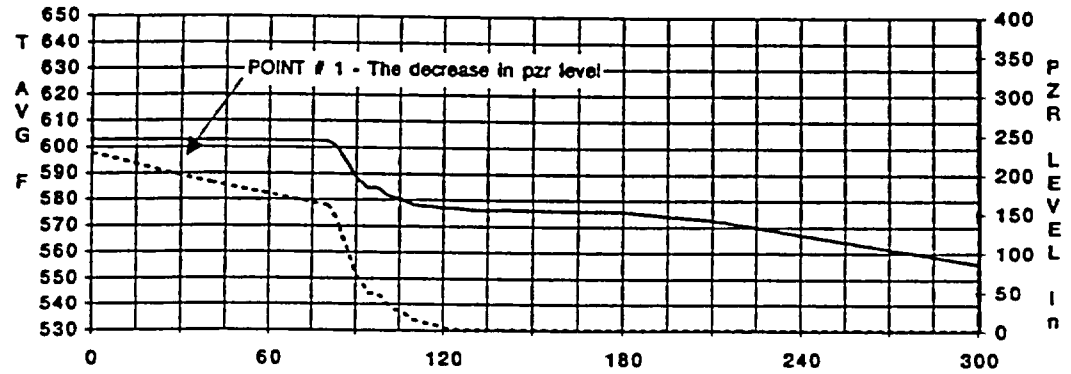

A FW DEMAND %
A FW FLOW %
0
60
120
180
240
300

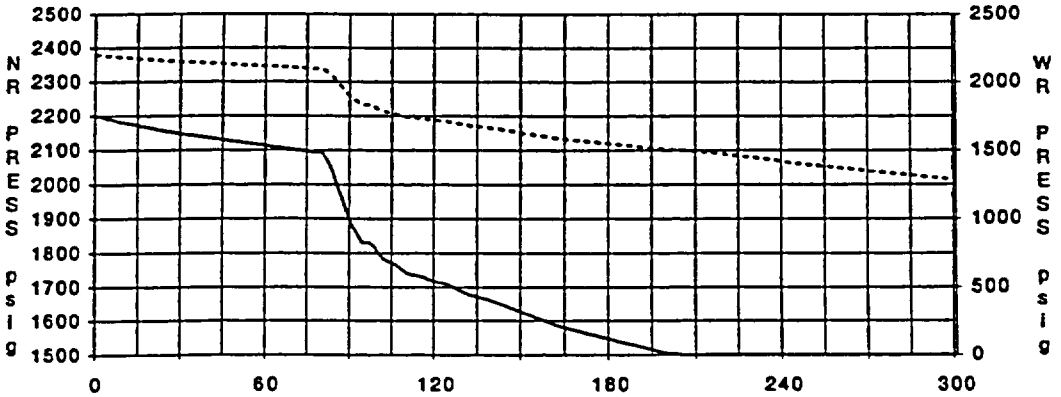

B FW DEMAND %
B FW FLOW %
0
60
120
180
240
300

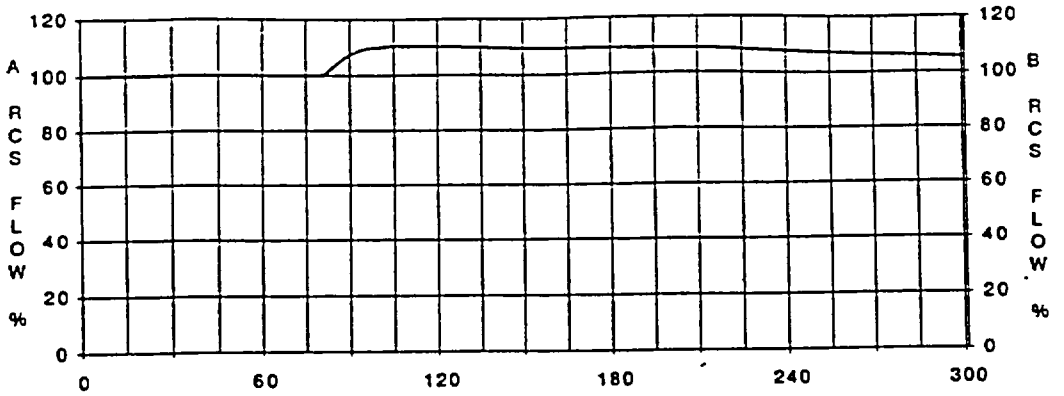

A OTSG LEVEL in
B OTSG LEVEL in
POINT # 4 - State the ESF signal that actuates AFW
0
60
120
180
240
300

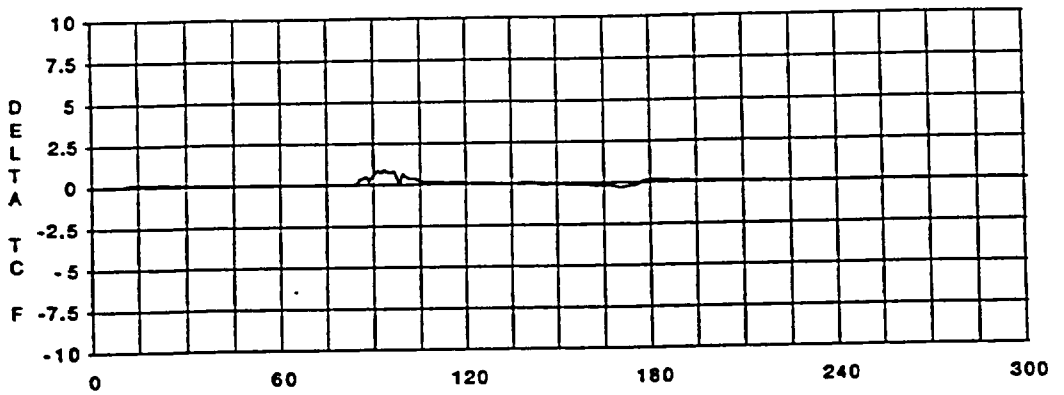

RX DEMAND %
RX POWER %
0
20
40
60
80
100
120
0
60
120
180
240
300

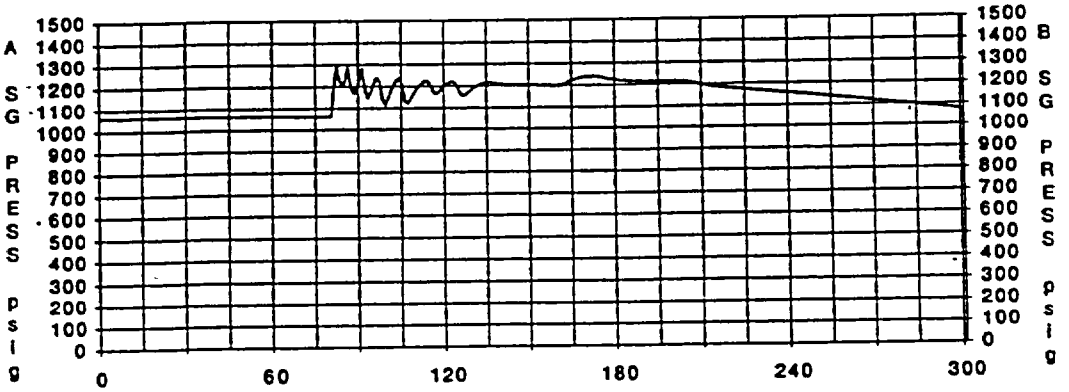

GROUP 7 %
GROUP 6 %
POINT # 2 - State the reactor trip signal
POINT # 3 - The constant group 6 position
0
10
20
30
40
50
60
70
80
90
100
0
60
120
180
240
300

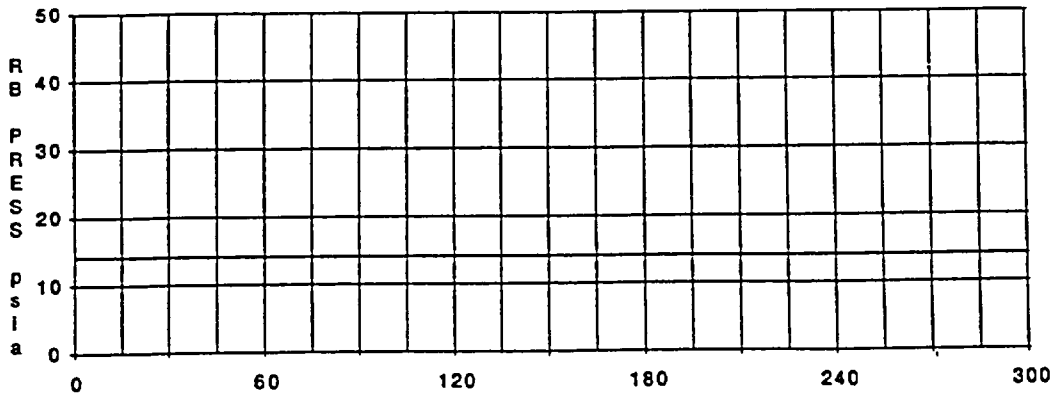

T AVG F
PZR LEVEL in
POINT # 1 - The decrease in pzr level
530
540
550
560
570
580
590
600
610
620
630
640
650
0
50
100
150
200
250
300
350
400
0
60
120
180
240
300


NR PRESS psig
WR PRESS psig
1500
1600
1700
1800
1900
2000
2100
2200
2300
2400
2500
0
500
1000
1500
2000
2500
0
60
120
180
240
300


A RCS FLOW %
B RCS FLOW %
120
100
80
60
40
20
0
0
60
120
180
240
300


DELTA TCF
10
7.5
5
2.5
0
-2.5
-5
-7.5
-10
0
60
120
180
240
300


A SG PRESS psig
B SG PRESS psig
1500
1400
1300
1200
1100
1000
900
800
700
600
500
400
300
200
100
0
0
60
120
180
240
300


RB PRESS psia
50
40
30
20
10
0
0
60
120
180
240
300

BABCOCK AND WILCOX
CROSS TRAINING MANUAL

CHAPTER 20 B&W Plant Differences

TABLE OF CONTENTS

20.0 B&W PLANT DIFFERENCES
20.1 Normal Operating Parameters 20-1
20.2 Reactor Protection and Engineered Safety Features Actuation Systems 20-1
20.2.1 Reactor Protection System 20-1
20.2.2 Engineered Safety Features Actuation System 20-1
20.3 Integrated Control System 20-1
20.3.1 Integrated Master 20-2
20.3.2 Feedwater Subassembly 20-2
20.3.3 Feedwater Flow Control 20-3
20.3.4 ICS and NNI Inputs 20-3
20.4 Reactor Coolant System Differences 20-3
20.4.1 Reactor Vessel Construction (Figure 20-8) 20-3
20.4.2 Fuel Assemblies and Control Rods 20-3
20.4.3 Loop Designs 20-3

LIST OF TABLES

20-1 B&W Plant Comparisons 20-4
20-2 RPS Comparisons 20-5

LIST OF FIGURES

20-1 RCS Temperatures vs. Power
20-2 Header Pressure Control
20-3 177FA FW Demand
20-4 MFW Regulating Valves
20-5 Fuel Assembly
20-6 Lowered Loop NSSS

20.0 B&W PLANT DIFFERENCES

Introduction:

The major differences between the B&W plant described in this manual and the operating B&W plants are: (1) normal operating parameters, (2) the reactor protection system design, and (3) the Integrated Control System (ICS). Minor differences in reactor vessel and fuel assembly construction also exist. These differences, as well as known dissimilarities, between operating B&W units will be described in this chapter.

20.1 Normal Operating Parameters

Babcock and Wilcox designates generations of plants by the number of fuel assemblies contained in the core. All of the operating B&W plants in the U.S. are 177 fuel assembly (177FA) units. Table 20-1 lists the major characteristics of each B&W unit.

The temperature program for a typical 177FA unit starts at 532°F at zero power, and temperature is escalated to 579°F as power changes from 0 to 15%. Of course, a constant T_{avg} is maintained at 579°F by the ICS from 15% to 100%. Figure 20-1 graphs hot leg, cold leg, and average temperature versus power. Since steam pressure is a function of no-load T_{avg}, normal operating steam pressure is 885 psig (900 psia). Normal reactor coolant pressure is 2155 psig and is sensed on one of the two hot legs. Pressurizer pressure is not measured in the 177FA units.

20.2 Reactor Protection and Engineered Safety Features Actuation Systems

20.2.1 Reactor Protection System

Chapter 10.1 describes the functioning of the 177FA Reactor Protection System (RPS). The 205FA RPS operates almost identically. However, there are differences in trip functions. The 205FA RPS contains a microprocessor that generates DNBR, offset, and power to ΔT trips. The offset trip prevents local heat generation from exceeding the centerline fuel limit, while the power to ΔT trip provides startup protection. High and low pressurizer level trips are provided in the 205FA RPS as diverse trips to the high and low RCS pressure trips. In addition, the high RB pressure trip is eliminated in the 205FA RPS design. 205FA RPS differences are discussed here to provide the student with a list of the trips that affect the transients in chapter 19 of this manual. Table 20-2 compares the trip functions of the two RPS designs.

20.2.2 Engineered Safety Features Actuation System

The Engineered Safety Features Actuation Systems (ESFAS) at the operating plants are essentially identical to the ESFAS described in chapter 10.2 with the exception of Davis-Besse and Crystal River. The Davis-Besse ESFAS system has a 2-out-of-4 actuation logic. According to the Updated Safety Analysis Report (USAR), the Crystal River ESFAS is a 2-out-of-3 system, but does not contain unit control modules. Override switches for the entire system are installed. Also, high and low pressure injection are bypassed separately; high pressure at 1700 psig, and low pressure at 900 psig.

20.3 Integrated Control System (ICS)

The major differences in the ICS can be found in the integrated master and the feedwater assemblies. Typically, the integrated master controls fewer turbine bypass and steam dump valves while the feedwater demand subassembly has the addition of high OTSG level limits. In addition, two methods of feedwater flow control exist.

20.3.1 Integrated Master

Since the control of the turbine, the "kicker" signal, the calibrating integral, and the characterization of the feedwater and reactor demand signals are identical to those discussed in chapter 9, only the control of the turbine bypass and atmospheric dump valves will be discussed in this section. As shown on Figure 20-2, the header pressure setpoint (885 psig) is combined with one of three bias values. If the reactor is not tripped and the turbine is tripped, then a zero bias is added to the header pressure error. If neither the reactor nor turbine is tripped, then 50 psig is added to the turbine header pressure error. Finally, if the reactor is tripped, 125 psig is added to the header pressure error.

The zero bias value is selected by the ICS during plant startups and maintains a header pressure of 885 psig. When the unit load demand is greater than 15%, the turbine bypass valves are closed, and there is less than a 10-psig header pressure error, the 50-psig bias value will be selected. This bias value will cause turbine bypass or atmospheric dump valve actuation at 935 psig and provides steam header relief. The 125 bias value is selected when the reactor trips. This selection biases the header pressure setpoint up to 1010 psig and minimizes the reduction in T_{avg} and the associated outsurge from the pressurizer. Normal post-trip T_{avg} is approximately 550°F.

The bias value selection relay (T_1) determines the value of bias that will be added to the header pressure setpoint. The header pressure setpoint, and the appropriate bias value is compared with actual header pressure in difference amplifiers. Each difference amplifier also receives an input from its associated header pressure transmitter. (The steam headers at some units are not cross-connected unless the turbine throttle and control valves are open.) The error signal from the comparison of header pressure setpoint plus bias and actual header pressure is routed to a high select unit via hand-automatic stations that allow the operator to manually control the steam valves.

In the high select unit, the higher of either turbine header pressure error or OTSG pressure error is selected for valve control. OTSG pressure error is determined by comparing a fixed setpoint of 1025 psig with actual OTSG pressure. The 1025 psig setpoint sets an upper limit of steam pressure for transients that do not actuate the steam line safety valves.

The outputs of the high select units are routed to the turbine bypass valves or the atmospheric dump valves by relays T_2 and T_3. If the condenser is able to accept steam from the turbine bypass valves, as determined by condenser vacuum and circulating water flow, then relay T_3 will allow the turbine bypass valves to open. If condenser vacuum or circulating water flow is lost, then relay T_3 will transfer a zero demand to the bypass valves, while relay T_2 will transfer the valve demand signal to the atmospheric dumps.

20.3.2 Feedwater Subassembly

The level in the OTSG must not be allowed to cover the steam aspirating ports; therefore, a circuit must be added to limit level in the steam generators. The circuit is located downstream of the feedwater error amplifier and consists of an input from the operating range level detector, an error amplifier that compares actual operate range level with a fixed setpoint, and a low select amplifier. This circuit is shown in Figure 20-3. If actual operating range level is equal to or greater than the fixed setpoint, the low select amplifier will select the output of the high level limits circuit for the control of feedwater flow. In other words, feedwater flow will be limited to a value that

maintains OTSG level at a value less than or equal to the high level limits setpoint.

20.3.3 Feedwater Flow Control

Most of the operating B&W units control feedwater flow by the methods described in chapter 9; however, Arkansas Nuclear One - Unit 1 and Crystal River Unit 3 control feedwater flow differently. These units have three feedwater flow control valves instead of two, and a cross-connect valve that is closed above 50% feedwater demand is also installed (Figure 20-4). The startup feedwater regulating valve is used to control feedwater flow when the steam generator is on low level limits and with low flow demands. When the startup valve reaches 80%, the low load block valve opens, and the low load control valve controls feedwater flow from approximately 15% to 50% power. At 50% power, the main feedwater block valve opens and the cross-connect valve is closed. From 50% to 100% load, feedwater flow is controlled by varying main feedwater pump speed.

20.3.4 ICS and NNI Inputs

Since pressurizer pressure is not measured on the 177FA plants, RCS pressure from two of the RPS channels is supplied to the NNI system, via buffer amplifiers, for the control of pressure. Also, RCS flow is supplied to the ICS and NNI systems from the RPS for indication and control functions.

Power range inputs also differ from unit to unit. Some units have a separate power range detector that supplies the reactor power input to the ICS, while other units use some sort of averaging and high select units.

20.4 Reactor Coolant System Differences

20.4.1 Reactor Vessel Construction

There are two differences (besides dimensional variations) between the 205FA reactor vessel and the 177FA reactor vessel. First, the 177FA vessel contains a thermal shield that is installed to minimize the effects of neutron and gamma reactions in the reactor vessel. Secondly, the core support assembly is constructed in two pieces. The first part of the core support assembly is called the core support barrel and consists of the flow distributor, the lower grid, and a right circular cylinder that houses the baffle plates and fuel assemblies. The thermal shield is attached to the core support barrel. The second part of the core support assembly is called the core support shield and is bolted to the top of the core support barrel. After initial assembly, the core support barrel and the core support shield may be thought of as one component.

20.4.2 Fuel Assemblies and Control Rods

The 177FA fuel assembly has a 15 X 15 array of fuel pins and guide tubes instead of a 17 X 17 array. The fuel assembly spring design utilizes a single spring instead of the four springs that are installed on the 205FA design (see Figure 20-5). In addition, the control rods have 16 fingers instead of 24 fingers.

20.4.3 Loop Designs

All of the 177FA plants with the exception of Davis-Besse are of the lowered loop design (Figure 20-6). Davis-Besse has a raised loop design similar to the design discussed in the technology manual. The design was changed to a raised loop design to improve the natural circulation response of the plant.

TABLE 20-1 B&W PLANT COMPARISONS

Plant	Docket #	Mw(t)	Mw(e)	Turbine	Cond/UHS
ANO-1	50-313	2568	836	Westinghouse	River/Pond
Crystal River-3	50-302	2544	821	Westinghouse	Sea/Sea
Davis-Besse	50-346	2772	860	General Electric	Tower/Lake
Oconee-1	50-269	2568	846	General Electric	Lake/Lake
Oconee-2	50-270	2568	846	General Electric	Lake/Lake
Oconee-3	50-287	2568	846	General Electric	Lake/Lake
TMI-1	50-289	2568	808	General Electric	Tower/River
WNP-1 (TTC Sim.)	50-460	3760	1266	Westinghouse	Tower/Pond

Plant	RCPs	Diesel (kw)	AFW Motor/Turbine	EFIC	Bypass/Dump (% steam flow)
ANO-1	Byron-Jackson	2750	1/1	Yes	15.8/6.2
Crystal River-3	Byron-Jackson	2750	1/1	Yes	15/7.5
Davis-Besse	Byron-Jackson	2600	0/2	No	25/15
Oconee-1	Westinghouse	Hydro	2/1	No	25/0
Oconee-2	Bingham	Hydro	2/1	No	25/0
Oconee-3	Bingham	Hydro	2/1	No	25/0
TMI-1	Westinghouse	2600	2/1	No	22.5/6.4
WNP-1 (TTC Sim.)	Bingham	7060	2/1	No	18.5/78.4

General Notes:

1. Davis-Besse has a separate high pressure injection system. The discharge of the HPI pumps is approximately 1600 psig.
2. ANO-1 and Crystal River-3 control feedwater flow above 50% power by varying MFP speed.
3. The Oconee units do not have main steam isolation valves. The piping from the OTSGs to the turbine is Seismic Category I.

TABLE 20-2 RPS COMPARISONS

177FA RPS Trips	205FA RPS Trips
High Reactor Power	High Reactor Power
High RCS Pressure	High RCS Pressure
Low RCS Pressure	Low RCS Pressure
Variable Low RCS Pressure	*DNBR
Flux/Imbalance/Flow	*Offset, Power to Flow
Power to Pumps	*DNBR, *RCP Status
High Outlet Temperature	High Outlet Temperature
High Reactor Building Pressure	No Equivalent Trip
Anticipatory Loss of MFW Pumps Turbine Trip	To Be Determined
	High Pressurizer Level Low Pressurizer Level *Power to ΔT

* These trips are calculated by a safety-related digital computer.

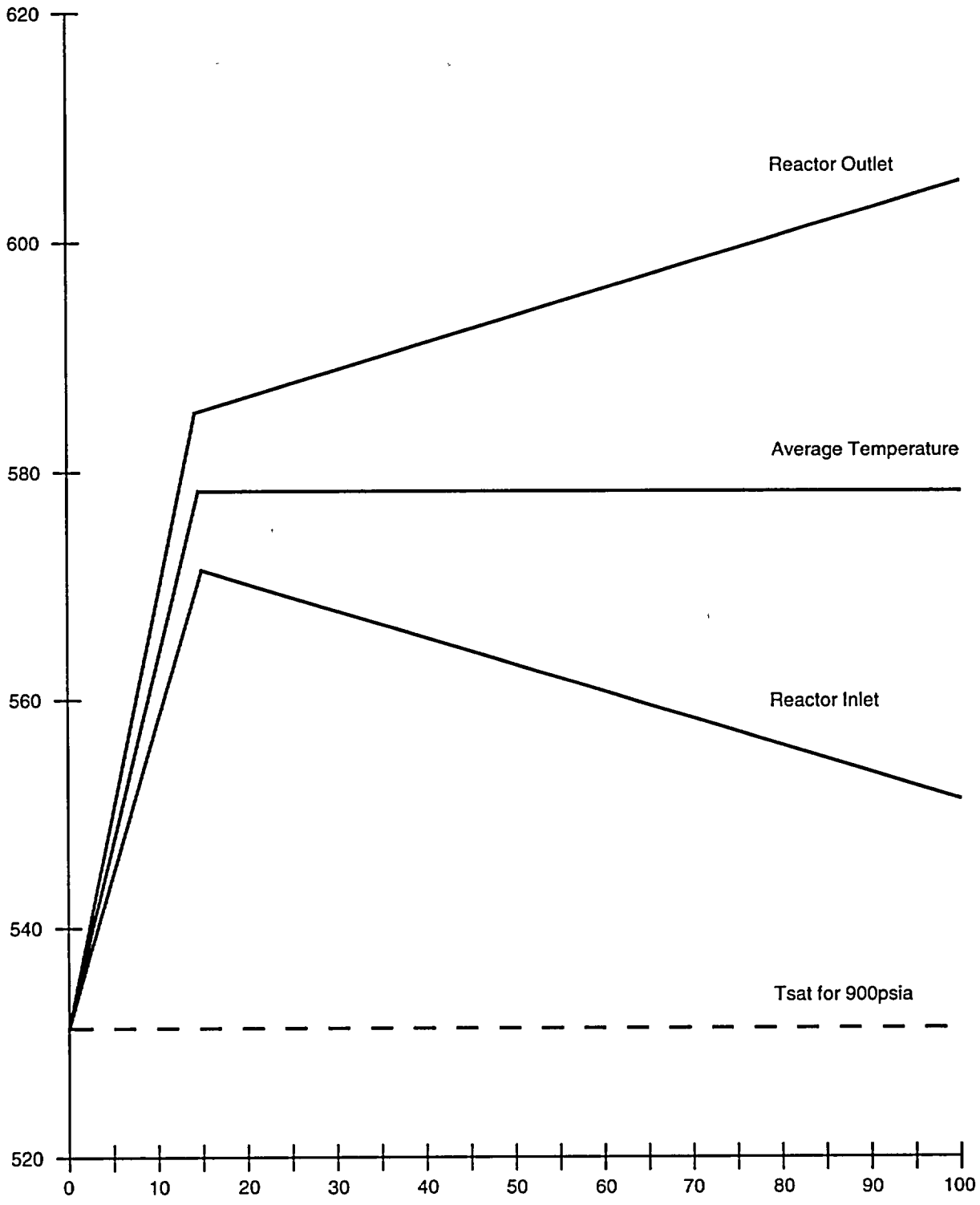


Figure 20-1 RCS Temperatures vs. Power

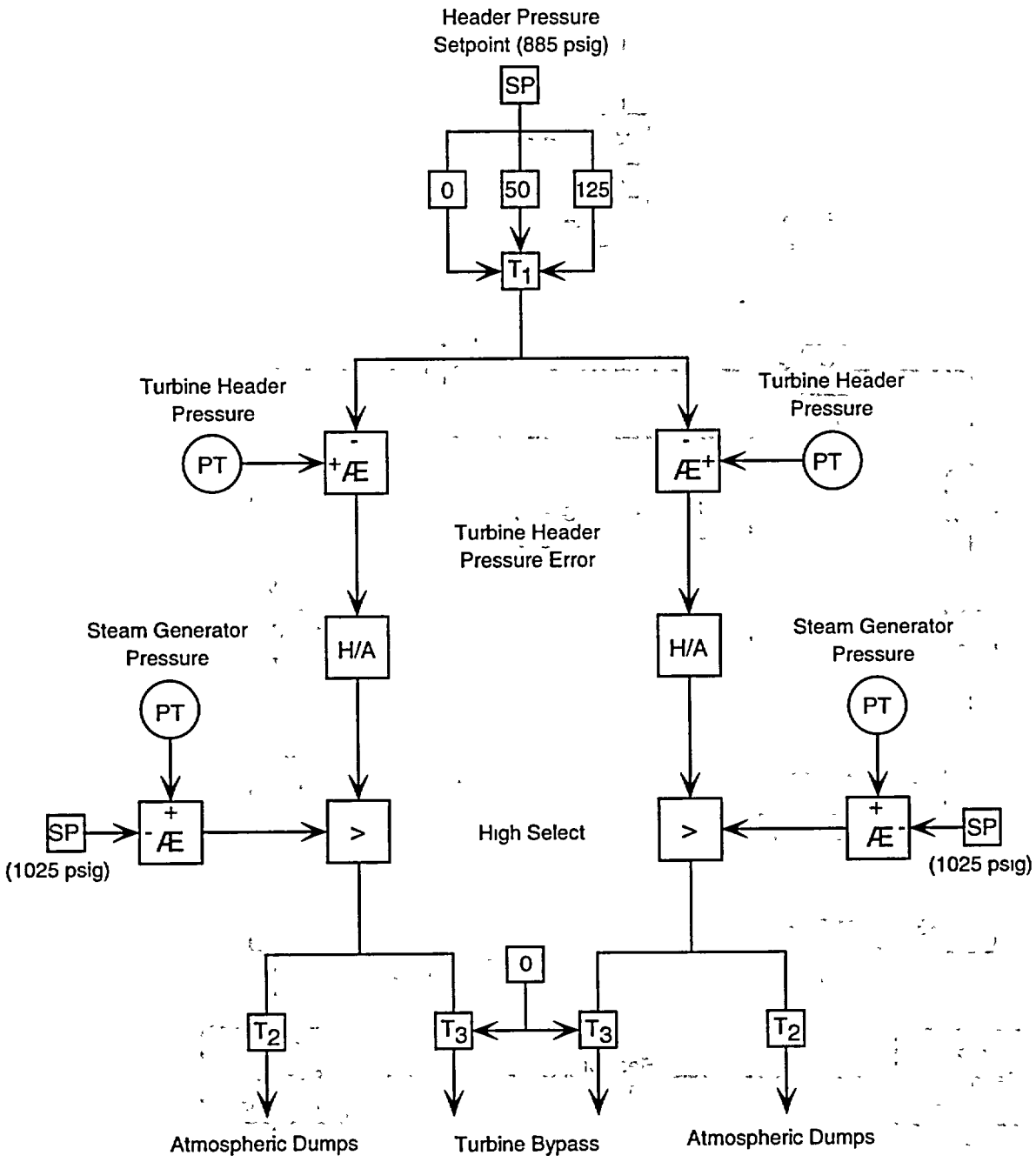


Figure 20-2 Header Pressure Control

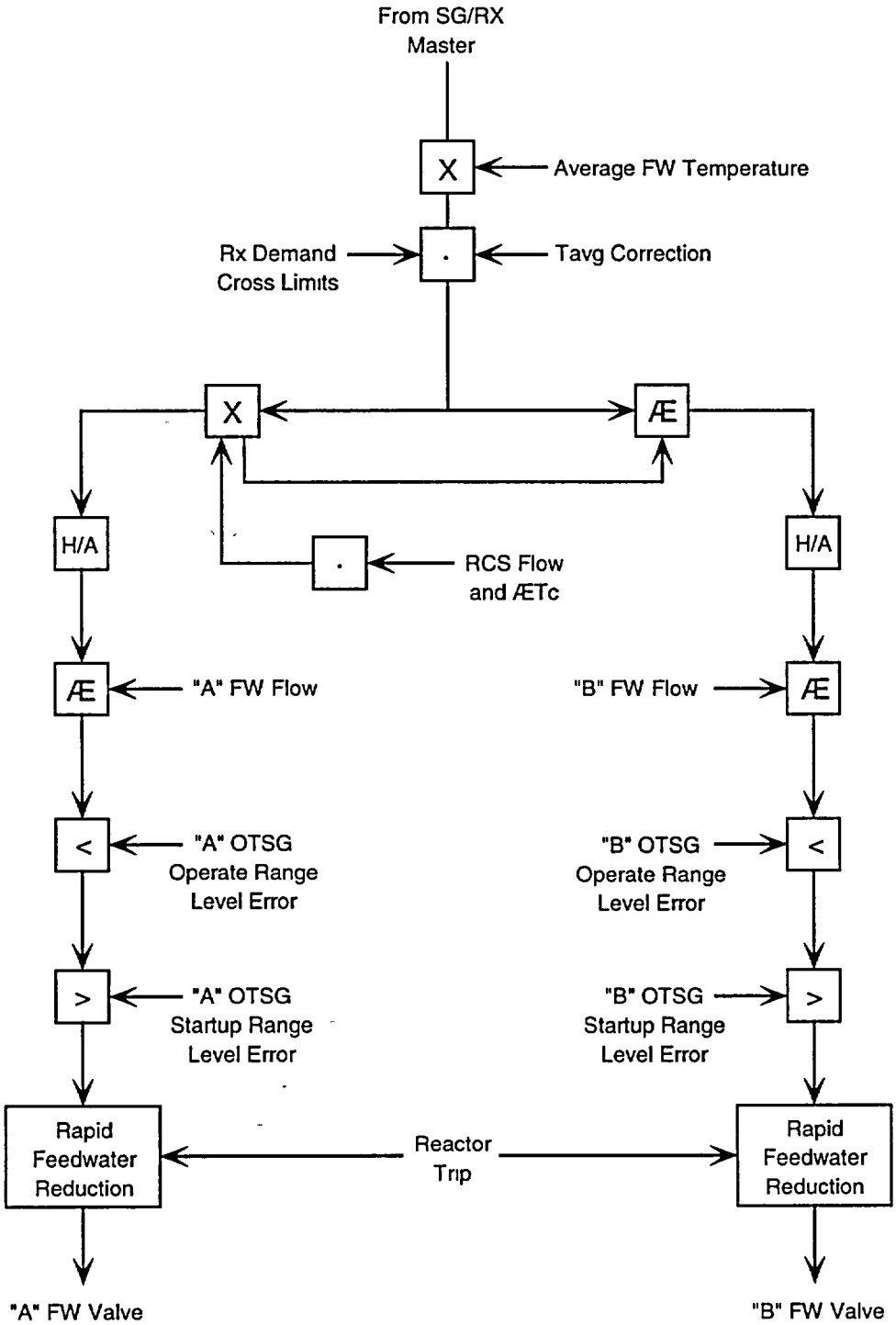


Figure 20-3 177FA FW Demand

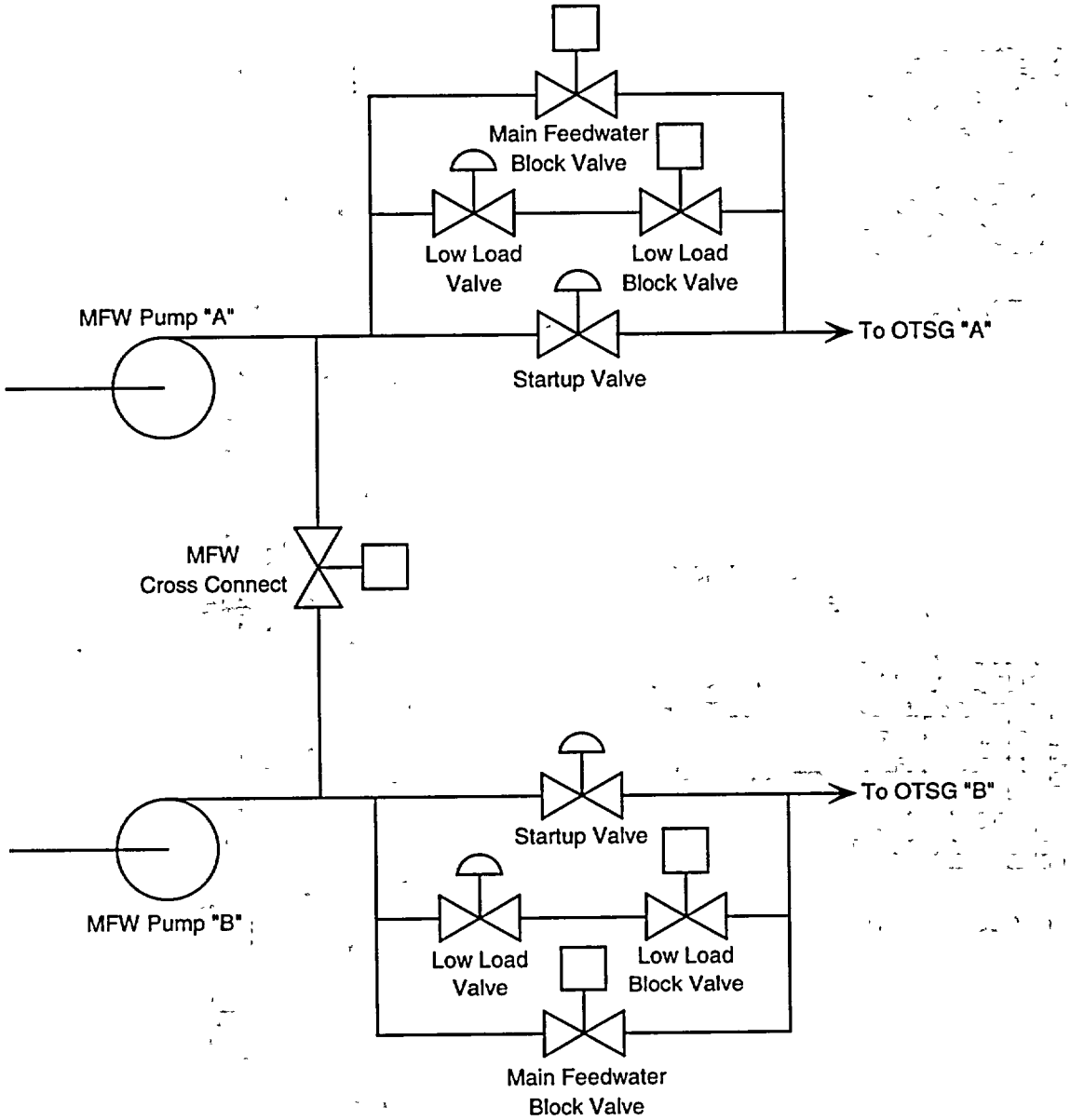


Figure 20-4 MFW Regulating Valves

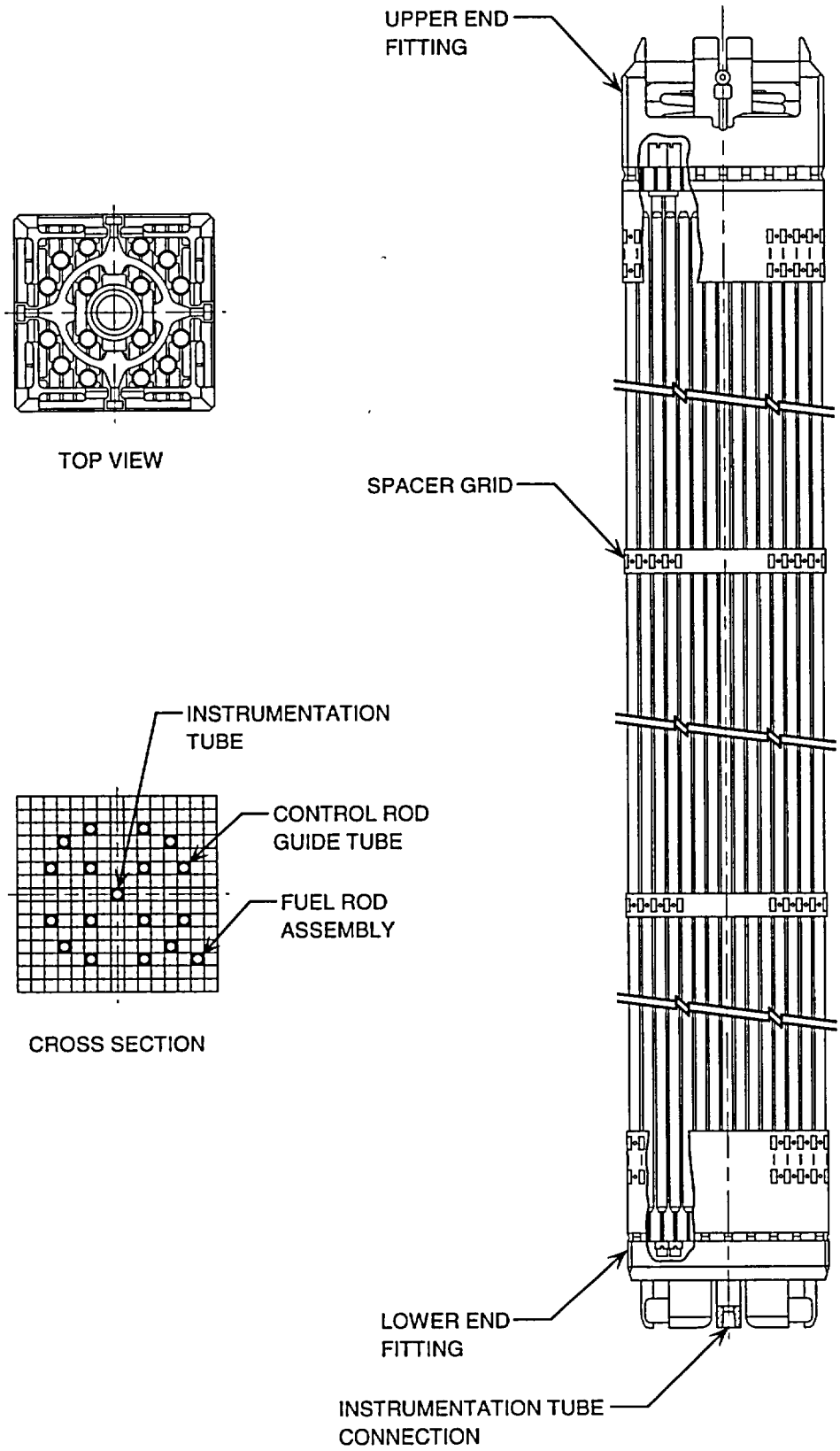


Figure 20-5 Fuel Assembly

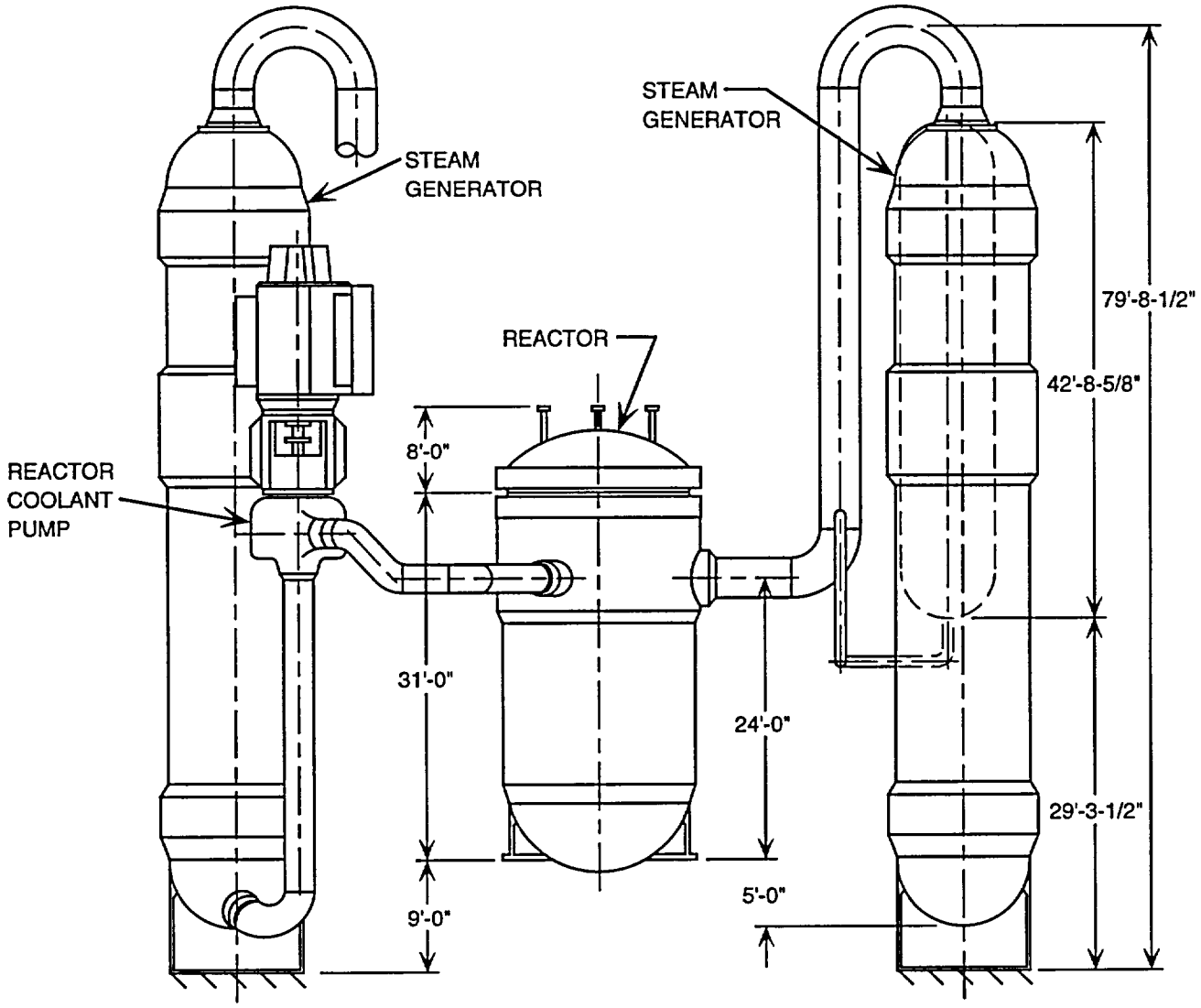


Figure 20-6 Lowered Loop NSSS

BABCOCK AND WILCOX
CROSS TRAINING MANUAL

CHAPTER 21 Advanced Control System

TABLE OF CONTENTS

21.0 ADVANCED CONTROL SYSTEM
21.1 Introduction 21-1
21.2 ACS Subsystems 21-1
21.2.1 Core Thermal Power Demand Subsystem 21-1
21.2.1.1 Operator Set Mode 21-1
21.2.1.2 Automatic Load Limiting 21-2
21.2.1.3 Tracking 21-2
21.2.2 Integrated Master 21-3
21.2.2.1 Calibrating Integrals 21-3
21.2.2.2 Cross Terms 21-3
21.2.2.3 Turbine Control 21-3
21.2.2.4 Turbine Bypass Valves (TBVs) 21-4
21.2.2.5 Turbine Loading and Unloading 21-4
21.2.3 Feedwater Control Subsystem 21-4
21.2.3.1 Feedwater Demand Corrections 21-5
21.2.3.2 Feedwater Loop Demands 21-5
21.2.3.3 Feedwater Pump Speed Control 21-7
21.2.4 Reactor Control Subsystem 21-7
21.2.4.1 Reactor Demand Corrections 21-7
21.2.4.2 Rod Control 21-8
21.3 Integrated Operations 21-8
21.3.1 Normal Power Increase 21-8
21.3.2 Loss of One Reactor Coolant Pump 21-9
21.3.3 Load Rejection 21-9
21.3.4 Reactor Trip 21-10
21.3.5 Turbine Master in Manual 21-11
21.3.6 Seam Generator Master in Manual 21-11
21.3.7 Reactor Master in Manual 21-12

LIST OF TABLES

21-1 Automatic Load Limits and Rate Limits 21-13
21-2 Control Variables and Calibrating Integrals 21-13
21-3 Turbine Loading Status Flag 21-14
21-4 Transfer Functions 21-14

LIST OF FIGURES

21-1 ACS Block Diagram
21-2 Advanced Control System
21-3 Feedwater Valve Operations
21-4 Load Control Panel

21.0 ADVANCED CONTROL SYSTEM

Learning Objectives:

1. Explain the function of the following Advanced Control System (ACS) subsystems:

   a. Core Thermal Power Demand (CTPD)
   b. Integrated Master
   c. Feedwater Control
   d. Reactor Control

2. Define the following terms:

   a. Tracking
   b. Runback
   c. Cross Limits

3. With the use of a block diagram of the ACS, discuss the following:

   a. Normal power increase and decrease
   b. Runbacks
   c. Cross Limits
   d. Placing ACS hand/auto stations in manual
   e. Load rejection
   f. Turbine trip
   g. Reactor trip

21.1 Introduction

The purpose of the ACS is to match reactor thermal power with core thermal power demand while maintaining a balance between heat production in the primary system and heat removal in the secondary system.. As shown in Figure 21-1, the ACS accomplishes this purpose through four subsystems: Core Thermal Power Demand (CTPD), Integrated Master, Feedwater Control, and Reactor Control. Figure 21-2 is a more detailed diagram of the ACS.

The CTPD subsystem functions as a setpoint generator for the ACS. The Integrated Master receives the core thermal power setpoint from the CTPD subsystem and produces signals for feedwater control, reactor control, and steam valves. The ACS will maintain constant T_{ave}, steam pressure at setpoint, core thermal power within limits, and feedwater flow to obtain balanced steam conditions in the two steam generators.

Section 21.2 is a description and explanation of the operations of each of the four subsystems in the ACS. Section 21.3 is an explanation of the integrated operations of the ACS.

21.2 ACS Subsystems

21.2.1 Core Thermal Power Demand Subsystem

The Core Thermal Power Demand subsystem (CTPD) is the primary interface between the operator and the ACS. In the operator set mode, the operator can communicate to the CTPD a core thermal power setpoint and a rate. The CTPD also recognizes various limiting conditions of the plant or the ACS and will initiate either automatic load limiting or tracking.

21.2.1.1 Operator Set Mode

The Load Control Panel (LCP), shown in Figure 21-4, has the controls and indications that the operator uses to communicate with the CTPD. The operator uses the "increase" and "decrease" pushbuttons to establish a setpoint that will appear in the lower window of the LCP. The lower window is called "CTPD SET," and it displays the target setpoint. The setpoint that is being processed by the rest of the ACS appears in the upper window which is called "CTP DEMAND." The rate at which the setpoint in the upper window changes is determined by the "RATE SET" thumbwheels on the LCP. The operator can select any value from 0.0 to 9.9%. The scale, either % per minute or % per hour, is selected by a pushbutton. There is a

"HOLD" pushbutton that will prevent changes from occurring while the operator is entering a setpoint and rate, or stop a power change in progress.

21.2.1.2 Automatic Load Limiting

There are several plant conditions which will cause the CTPD to automatically limit the maximum target setpoint. The lowest limit corresponding to a load limiting condition in effect will be selected by transfer function T1 (Fig. 212), and the operator input will be blocked. The automatic load limits are:

1. Loss of one reactor coolant pump (74%) sensed by breaker open contacts.
2. Loss of one feedwater pump (65%) sensed by low hydraulic oil in the turbine.
3. Asymmetric rod (55%) sensed by rod control logic using absolute position indication.
4. Loss of reactor coolant flow (variable limit) sensed by the sum of the loop flow signals.
5. Both generator output breakers open (20%).
6. Maximum runback (15%), when selected.

The rates for automatic load limiting are listed in Table 21-1.

21.2.1.3 Tracking

Tracking is a mode of operation that will be in effect when an ACS subsystem is unable to function normally, and the ACS may not be able to maintain the primary to secondary heat balance or automatic maneuvering. During tracking, the target setpoint is determined by a tracking parameter which enters the CTPD by transfer function T2 (Fig. 21-2), while manual control and automatic load limiting are blocked.

An ACS subsystem is unable function normally when it has become independent due to manual operation, maneuvering rate limitation, or by final component failure. The conditions that cause tracking are:

1. Reactor or feedwater cross limits.
2. Steam generator master in manual with at least one SG <u>not</u> on low level limit.
3. Blocking <u>both</u> feedwater loop flow control paths with at least one SG <u>not</u> on low level limit. (A control path is blocked by low level limit or manual operation of the loop master demand station or the controlling feedwater valve control station.)

NOTE: When both SGs are on low level limits, tracking will not occur due to a feedwater condition.

4. Reactor demand or rod control system in manual.
5. Turbine master in manual.

The tracking parameter that becomes the target setpoint in the CTPD is determined by the status of the ACS subsystems. When the feedwater subsystem is independent, total feedwater flow will be the target setpoint, and appears in the CTPD SET window. That signal will be fed to the CTP DEMAND window at 20% per minute. When the turbine control is independent, generated megawatts will be the tracking parameter. When reactor control is independent, NI flux will be the tracking parameter. If more than one subsystem is independent, the highest priority tracking parameter will be the target setpoint. Feedwater flow has the highest priority, followed by generated megawatts and the NI flux. An exception to this occurs when feedwater and reactor are both independent, in which case the CTP Best signal will be the tracking

parameter.

21.2.2 Integrated Master

The integrated master (Fig. 21-2) sends demand signals to reactor control and feedwater control subsystems, and controls the turbine and turbine bypass valves, and coordinates their actions in order to maintain heat balance. There are four control variables that are measured and used by the ACS subsystems to control the plant. The control variables, in priority order, are T_{ave}, Turbine Header Pressure (THP), Core Thermal Power (CTP), and ΔT_C. T_{ave} and THP are indications of the heat transfer from the primary to the secondary system. Controlling CTP is the primary objective of the ACS. ΔT_C is related to the balance of heat transfer through the two SGs.

21.2.2.1 Calibrating Integrals

In the process of maintaining a balance between plant systems, an imbalance may develop because of changing efficiency, miscalibration, instrumentation drift, or SG heat transfer capability. Calibrating integrals are used in the ACS to compensate for imbalances and allow stable operation of the plant. The control variables are assigned to the integrals based on the mode of operation. The four calibrating integrals are the reactor integral, the feedwater integral, the EHC Load Reference Motor (LRM), and the ΔT_C integral in the feedwater subsystem.

During normal operation, T_{ave} is controlled by reactor control using the reactor integral, THP is controlled by the LRM, CTP is assigned to the feedwater integral, and ΔT_C is controlled by the ΔT_C integral. When reactor control is in manual, the feedwater integral takes over T_{ave} control, THP is still controlled by the LRM, the plant operator controls CTP, and ΔT_C is still controlled by the ΔT_C integral. Control variable assignments to integrals for other modes of operation are shown in Table 21-2.

21.2.2.2 Cross Terms

THP Error

The difference between the turbine header pressure and setpoint is applied to the reactor and feedwater demands to compensate for differences in response times of control systems and processes. The signal is limited to a maximum of ±50 psi.

T_{ave} Error

A proportional T_{ave} error is applied to feedwater demand to maintain proper heat balance. There is a deadband of ±1.2°F which must be exceeded before any modification is applied. In the event that feedwater has T_{ave} control, the deadband is removed and T_{ave} error is also applied to the feedwater integral.

RCS Pressure

When RCS pressure is above the setpoint, a correction term is applied to reactor demand, feedwater demand, and turbine demand. This is done to restore heat balance and prevent a high RCS pressure reactor trip. Reactor demand would be decreased during correction while feedwater and turbine demands would be increased.

21.2.2.3 Turbine Control

The turbine is normally controlled by the error between turbine header pressure and an operator selectable setpoint (THP error) and the error between CTP demand and feedback from the pulser through a compensator. The error is also increased when RCS pressure is higher than setpoint. The error signal drives a pulser that continues to demand turbine valve movement until the error is zero.

When there is a ±50 psi THP error for 5 seconds, the turbine master will switch to manual operation. However, this action is blocked during

the following conditions when a large THP error is anticipated:

1. Crosslimits
2. Loss of RCP
3. Loss of FWP
4. Asymmetric rod
5. Loss of RCS flow
6. Both generator breakers open
7. Maximum runback

When the governor valve control system (EHC) assumes control due to a stator coolant runback or a power load unbalance, the transfer of the turbine control station to manual cannot be blocked.

21.2.2.4 Turbine Bypass Valves (TBVs)

There are three basic functions of the TBVs. When power is low and the turbine can not control THP (Turbine Load Status Flag is FALSE), or if the turbine is tripped, the bypass valves will control at the THP setpoint. When the turbine is operating and the turbine can control THP (Turbine Load Status Flag is TRUE), there is a 50 psi bias added to the THP setpoint, and the bypass valves act as pressure relief. When the reactor trips, a 125 psi bias is added to control at a higher pressure and limit the RCS cooldown. There is also an independent high pressure relief if the SG pressure exceeds 1035 psig.

The position demand for the TBVs is based on turbine header pressure (THP) when the turbine master is in automatic, and steam generator outlet pressure when the turbine master is in manual. The selection is made by the transfer function T7. In both cases, the setpoint is established by the operator at the turbine master. The transfer function T9 will block operation from the control room unless the condenser vacuum is greater than 7″. When vacuum is less than 7″, the TBVs can be controlled only at the auxiliary shutdown panel (ASDP).

21.2.2.5 Turbine Loading and Unloading

During a normal plant startup, the reactor will be producing 10% to 20% power before the turbine is rolled and loaded. The TBVs will be dumping steam to the condenser and maintaining steam pressure at 885 psig. After the turbine is rolled to synchronous speed, it will be loaded with about 30 Mw. The operator can manually load the turbine by using the turbine master or the increase pushbutton on the EHC panel. As load increases, TBVs will close to maintain steam pressure at setpoint.

Another method of loading the turbine would be to use the turbine load pushbutton on the load control panel. When the CTPD is greater than 10%, pressing the load pushbutton will initiate automatic turbine loading. When the TBV demand is zero, the 50 psi bias is applied to the bypass valves. This happens because the Turbine Load Status Flag is set (TRUE). Pushing the unload pushbutton when CTPD is less than 20% resets the flag (FALSE). Table 21-3 shows the conditions that determine the status of the flag and its effects.

21.2.3 Feedwater Control Subsystem

The feedwater control subsystem normally maintains feedwater flow equal to feedwater demand while maintaining the correct balance of flow to the two steam generators. The subsystem must also maintain SG levels above minimum requirements and correct for heat balance problems during abnormal conditions.

When SG level is above 25″ in the startup range, feedwater demand is compared to actual feedwater flow to position feedwater regulating valves. When startup level is 25″ or less, valve position will be positioned according to a level error. Transfer functions T4 and T5 will select flow control or level control for SG A and B, respectively.

The integrated master converts the CTP demand into a feedwater demand signal, which is sent to the feedwater subsystem. This demand signal is corrected by a number of signals to produce the total feedwater demand.

21.2.3.1 Feedwater Demand Corrections

Feedwater Calibrating Integral

The input to the calibrating integral is either the CTP error, the T_{ave} error, or the THP error. During normal operation with all the ACS stations in automatic, CTP error is selected. When the reactor is in manual control, T_{ave} error will be selected. When T_{ave} is controlled by the reactor and the turbine is in manual control, THP error is selected (as long as the turbine load status flag is true).

THP Error

The THP error, up to a maximum of ±50 psi, is added to feedwater demand at all times.

T_{ave} Error

Any time the T_{ave} error exceeds ±1.2°F, the excess is added to feedwater demand. When the feedwater control subsystem takes control of T_{ave}, the deadband is removed.

RCS Pressure Error

When RCS pressure exceeds 2250 psig, then feedwater demand is increased proportionally.

CTP Error

The difference between the CTP demand and a the CTB Best calculation is added proportionally to the feedwater demand.

Neutron Error (Reactor Cross Limit)

Whenever the difference between reactor demand and neutron flux exceeds ±5%, the excess is added to feedwater demand. When reactor power is higher that demand by more than 5% (negative neutron error), feedwater demand is increased. A positive neutron error exceeding 5% will decrease feedwater demand. Reactor cross limit will also cause tracking with NI flux as the tracking parameter.

Feedwater Temperature

Feedwater demand is modified by a function of feedwater temperature when that temperature differs from what is expected for the load. When feedwater temperature is too low, feedwater demand is decreased to maintain the heat balance from primary to secondary. When temperature is too high, demand is increased.

21.2.3.2 Feedwater Loop Demands

Both steam generators receive coolant flow from the reactor at the same temperature (T_H). The temperature of the coolant returning to the reactor (T_C) from each steam generator is determined by the amount of heat transfer in that generator. It is desirable to maintain the outlet temperatures of the two steam generators nearly equal for core power distribution concerns. The only controlled variable available to control ΔT_C is feedwater flow. Feedwater flow to the two steam generators is rationed to maintain ΔT_C at zero while maintaining total flow equal to the demand.

As an example, consider that SG A has some fouling. Its heat transfer characteristics are lower than SG B. As a result, loop T_C is higher in loop A than in loop B. If the feedwater flow to SG A is increased, T_C in loop A will be decreased. At the same time, feedwater flow to SG B will be decreased so that the total flow matches the total demand.

Loop Demand Multiplier

The rationing of feedwater flow is done by controlling the multiplier used to modify the signal from total feedwater demand to loop A demand. The difference between the total demand and the loop A demand becomes the loop B demand. Normally, the multiplier is 0.5 so that half the total demand goes to SG A and half to SG B. When the multiplier is increased, more demand goes to SG A and less to SG B.

The multiplier is modified by two signals, RCS loop flow error and ΔT_C control. The RCS loop flow error anticipates a large change in ΔT_C and changes the multiplier immediately regardless of any other condition. When a reactor coolant pump is lost, flow in that loop and the associated SG decrease, with a corresponding decrease in energy input to the SG. T_C in that loop will decrease. Decreasing feedwater flow to that SG will restore T_C.

The second signal comes from ΔT_C control, which uses proportional and integral action to correct the measured temperature difference between the loops. The proportional part of the circuit is for the large T_C errors and the integral part is correction for small changes that occur over long periods. ΔT_C control is blocked when the control station is in manual, either loop master or feedwater valve is in manual, or either steam generator is on level control.

Loop Feedwater Flow Error

The actual loop feedwater flow is subtracted from the loop demand. This flow error is the signal that normally drives the valve controllers to position the feedwater valves. When conditions are abnormal, the flow error will be replaced by one of several level errors.

Level Errors

The first of these level errors is based on a high level limit. The level in the operating range should be maintained below the aspirating ports. The actual operating level is subtracted from a setpoint in the 92% to 96% range. This level error is normally higher than the flow error, so the low select unit selects the flow error. As the operating range level approaches the high level setpoint, the level error decreases and may become lower than the flow error.

The second level error is the startup level subtracted from a setpoint of 25″. This level error will be selected whenever the SG startup level is less than 25″ or on low level limits (LLL). LLL is in effect on a SG if startup range level is less than 25″ and loop T_{ave} is less than setpoint. The selection of "level control" is made by T4 on SG A and by T5 on SG B. Once initiated, LLL will be in effect until both initiating conditions clear. The SGs will normally be on LLL below 15% power. LLL will result in the following actions:

1. Both SGs on LLL releases T_{ave} from control priority, and T_{ave} error will not increase reactor demand when T_{ave} is below setpoint.

2. Both SGs on LLL limits CTPD rate to 1%/minute.

3. Both SGs on LLL prevents feedwater tracking.

4. Either SG on LLL blocks ΔT_C correction.

The third level error will be selected by the T6 transfer functions when all reactor coolant pump breakers are open. This will cause the startup valves to raise the SG levels to enhance natural circulation. The operating range level is subtracted from a setpoint of 50%, or 95% with a degraded containment (RB pressure > 3 psig).

Valve Sequencing (Figure 21-3)

Sequencing of the startup and main feedwater regulating valves is accomplished by composite valve demand controllers and sequence bias signals. Startup valves control feedwater flow up to about 15% power and the main valves control up to 100%. The main valves have a capacity about 10 times that of the startup valves. During a startup, each startup valve opens as power increases. At about 90% open, the main feedwater block valve opens automatically. At about 98% open, sequencing bias is applied that will ramp the startup valve to 10% and open the main valve enough to compensate. From this point on, the valves will open together, the startup valve opening 10 times faster, until the startup valve is fully open. On a power decrease, the main valve begins to close first. At some point, the startup valve begins to close at the same time. When it reaches about 9%, the sequencing bias is removed and the startup valves ramp to 90% and the main valve closes. As the startup valve reaches 50%, the main block valve closes.

21.2.3.3 Feedwater Pump Speed Control

Feedwater pump speed is adjusted to maintain a constant differential pressure across the feewater valves. This will result in better control of feedwater by the feedwater valves over a wide range of flow.

The signal to control the speed of the pumps comes from the sum of the loop demands and the differential pressure error. The loop demands are the outputs of the loop masters. Valve ΔP is measured in each loop by two sensors and the higher of the two is selected. The differential pressure error is the difference between the lower of the selected loop ΔPs and a 35 psid setpoint.

21.2.4 Reactor Control Subsystem

The reactor control subsystem maintains the output of the reactor (neutron flux). It also matches heat production to heat removal capability by controlling T_{ave}. When the reactor master is in automatic, the reactor demand is CTP demand modified by a calibrating integral, T_{ave} error, THP error, and RCS pressure error.

21.2.4.1 Reactor Demand Corrections

Reactor Calibrating Integral

T_{ave} error is normally assigned to the reactor calibrating integral. If a steady state T_{ave} error exists, it indicates an error in the reactor demand signal that may be due to changes in SG or plant efficiency or neutron flux calibration errors. The integral will compensate and correct the demand.

When both SGs are on LLL, CTP error is assigned to the reactor integral unless the steam system is also unavailable, in which case THP error is assigned to the reactor integral.

T_{ave} Error

T_{ave} is subtracted from an operator adjusted setpoint and the resulting error is added to the reactor demand to control T_{ave} at a constant 579°F from 15% to 100% power. This will minimize volume changes in the RCS and compensate for changes in heat transfer capability in the SGs. Normally, unit T_{ave} is used, but if the flow falls below 95% in one loop, T_{ave} will be taken from the other loop. When both SGs are on LLL, T_{ave} error can only decrease reactor demand if T_{ave} increases above setpoint, but will not allow an increase in reactor demand.

THP Error

The THP error, up to a maximum of ±20 psi, is added to reactor demand at all times.

RCS Pressure Error

When RCS pressure exceeds 2250 psig, then reactor demand is decreased proportionally.

Reactor Demand Limit

There is a 101% limit on reactor demand to prevent a reactor trip on high power. Actual reactor power may increase above 101% due to a decrease in T_{ave}.

Feedwater Cross Limit

Feedwater cross limit, which initiates tracking, occurs when feedwater demand is more than 5% greater than the total feedwater flow. Feedwater flow will be the tracking parameter. Note that feedwater flow greater than demand does not cause feedwater cross limit.

21.2.4.2 Rod Control

Actual neutron flux from excore nuclear instruments is subtracted from reactor demand to produce neutron error. The neutron error will control rod movement. When power is above 10%, there is a 1% deadband and a 0.1% hysteresis. When the neutron error is greater than 1%, rods withdraw until the error is less than 0.9%. An error less than -1% inserts rods until the error is greater than -0.9%. When power is less than 10%, the deadband and hysteresis decrease proportionally.

21.3 Integrated Operations

21.3.1 Normal Power Increase

Initial Conditions: The plant is stable at 30% power with all ACS stations in automatic. A load increase to 80% at a rate of 5% per minute is desired.

1. CTPD Actions

   a. The operator must input the desired load and the loading rate.

   b. In Figure 21-2, the new load demand (CTPD Set) passes through T1 and T2 and is applied to the rate unit. At the end of 1 minute, the output from the rate unit (CTP Demand) will have increased from 30% to35%.

   c. The demand signal is sent to the integrated master.

2. Integrated Master Actions

   a. The demand signal is sent to reactor control, feedwater control, and turbine control.

   b. The signal sent to turbine control anticipates a change in THP. THP will tend to increase due to the reactor power increase and the increase in feedwater flow. The THP error will ensure that the proper steam pressure is maintained in the secondary system by increasing turbine demand. If the RCS pressure exceeds 2250 psig, turbine demand will be increased.

3. Feedwater Control Actions

   a. The CTP demand is applied to a function generator, the output of which is a total feedwater demand. This demand may be modified by the CTP error, the THP error, the feedwater integral, or the T_{ave} error if the deadband is exceeded.

   b. After this modified signal passes through the SG master, it may be further modified by the feedwater temperature or reactor cross limit. If the RCS pressure exceeds 2250 psig, feedwater demand will be increased.

   c. The modified total demand is rationed to

the two loop feedwater masters. Loop feedwater demand is compared with actual loop feedwater flow, and the resultant error signal opens the feedwater regulating valves.

d. The opening of the feedwater regulating valves decreases the valve ΔP. This signal and the increase in feedwater demand, increases main feedwater pump speed.

4. Reactor Control Actions

a. The CTP demand is sent to reactor demand and may be modified by the T_{ave} error, the THP error, and the reactor integral. If the RCS pressure exceeds 2250 psig, reactor demand will be decreased.

b. The modified reactor demand will be compared to NI flux and the resulting neutron error will be sent to the rod control system and rods will be withdrawn.

The actions described above will continue until the unit load is at 80%.

21.3.2 Loss of One Reactor Coolant Pump

Initial conditions: Unit load is at 90% power. Group 7 control rods are 90% withdrawn.

1. CTPD Actions

a. When the reactor coolant pump is lost, an automatic load limit of 74% overrides the operator input.

b. CTP demand will decrease at the rate of 25% per minute.

2. Integrated Master Actions

a. The decreasing CTP demand closes the turbine valves.

b. The closing of the turbine valves increases actual steam pressure. If steam pressure exceeds its setpoint by 50 psi, the turbine bypass valves will open to relieve excess energy.

c. The CTP demand, reduced by the THP error, is sent to reactor demand and feedwater demand subsystems.

3. Reactor Control Actions

a. The reduced demand signal from the integrated master will cause inward regulating rod motion. T_{ave} error will also decrease this demand signal. If RCS pressure exceeds 2250 psig, demand will be reduced.

b. Because the regulating rods are almost fully withdrawn, a large amount of rod motion occurs with little change in reactor power. This will result in reactor cross limit. Cross limits cause tracking. Since the tracking parameter is inserted downstream of the automatic load limit, the NI flux will be tracked at 20% per minute until the reactor cross limit clears.

4. Feedwater Control Actions

a. The reactor cross limits will limit the decrease in feedwater demand.

b. The actions of the RC flow and ΔT_C circuits will proportion feedwater between the OTSGs.

The actions listed above will continue until the load is stabilized at 75%.

21.3.3 Load Rejection

Initial conditions: Unit load is at 100% power. Group 7 rods are 90% withdrawn.

1. CTPD Actions

   When the main generator output breakers are opened, an automatic load limit of 20% becomes the new CTPD SET. The output of the CTPD wil decrease at 20% per minute.

2. Integrated Master Actions

   a. The turbine load status flag is FALSE because the integrated master cannot control the turbine. However, the closing of the turbine valves by the turbine EHC (in its effort to prevent turbine overspeed) results in a large increase in steam header pressure.

   b. A large header pressure error will be developed and cause actuation of the turbine bypass valves. The opening of these valves removes the excess energy from the RCS. The turbine load status flag being FALSE causes the bias to be zero, so the TBVs control at 885 psig.

   c. The turbine control system will cause the turbine master to go to manual, but no tracking will occur because the turbine load status flag is FALSE.

3. Feedwater Control Actions

   Feedwater flow will respond to the 20% per minute reduction in feedwater demand caused by the load limit. The reduction in feedwater flow will be tempered by cross limits from the reactor demand subassembly.

4. Reactor Control Actions

   a. The reduction in the reactor demand causes insertion of the regulating rods. Due to the low rod worth at the upper end of group 7, the actual decrease in reactor power will not keep up with the decrease in reactor demand. This will result in reactor cross limit to the feedwater demand subassembly.

   b. The reduction in reactor power continues until the reactor demand signal reaches 20%.

5. Final Conditions

   a. Reactor power is being maintained at 20%.

   b. Turbine bypass valves are dissipating the reactor's energy to the condenser while controlling header pressure at 885 psig.

21.3.4 Reactor Trip

When the reactor trips, the turbine is also tripped. Therefore, the reactor demand subsystem cannot control the regulating rods and the integrated master cannot control the turbine.

1. CTPD Actions

   The reactor trip automatically limits load to 0% and the output of the CTPD decreases at 600% per minute.

2. Integrated Master Actions

   The reactor trip biases will be selected for control of the TBVs. As the energy of the RCS is removed by the TBVs, the pressure error will decrease. At the end of the transient, header pressure will be 1010 psig, with the excess energy being dissipated to the condenser by the TBVs.

3. Feedwater Control Actions

   a. When the reactor trip signal is received, a negative loop demand will be input to the valve controllers, causing a rapid reduction in feedwater flow, until level control takes over at 25" (T4 and T5). Feedwater valve

and pump controllers will be put in automatic so that feedwater would decrease even if the SG master were in manual.

b. The main feedwater block valves will close

c. The speed of the main feedwater pumps will decrease to a preset value.

4. Final Conditions

a. Turbine bypass valves will be dissipating reactor decay heat and reactor coolant pump heat to the condenser by controlling header pressure at 1010 psig.

b. Steam generators will be on LLL.

21.3.5 Turbine Master in Manual

Initial conditions: The unit is at 50% load, and the turbine master is put in manual. The operator initiates a load decrease to 40%.

1. CTPD Actions

a. When manual control of the turbine is selected, the ACS goes into tracking. Actual generated Mwe becomes the tracking parameter, which is transferred to the feedwater and reactor control subsystems.

b. As turbine load is reduced, the demand to the feedwater and reactor control subsystems will also be reduced.

2. Integrated Master Actions

a. Since the turbine is being controlled by the operator, the signal from the ACS is not sensed by the turbine.

b. Should header pressure exceed the 885 psig setpoint plus the "normal" bias values (50 psi), TBVs will open.

3. Feedwater Control Actions

a. As the demand signal from the CTPD decreases, the feedwater demand signal decreases.

b. Feedwater flow is reduced as the regulating valve positions and feed pump speeds are reduced by the decreased feedwater demand signal. Feedwater flow will equal 40% at the end of the transient.

4. Reactor Control Actions

a. As reactor demand is decreased, reactor power will be reduced by regulating rod insertion.

b. Reactor power will stablilize at 40% at the end of the transient.

21.3.6 Seam Generator Master in Manual

Initial Conditions: The unit is at 75% power, and the operator places the steam generator master in manual and increases its output to a demanded value of 80%.

1. CTPD Actions

The ACS goes into tracking and will follow feedwater flow. The output or CTPD will follow at a rate of 20% per minute.

2. Integrated Master Actions

The turbine receives the output of the CTPD, which will increase turbine demand. Also, as the turbine header pressure increases because of the actions of the reactor and feedwater control subsystems, the turbine control valves open to return header pressure to the 885 psig setpoint. As the turbine control valves open, load will increase to 80%.

3. Feedwater Control Actions

   Feedwater flow increases to both SGs as it follows the demand from the SG master.

4. Reactor Control Actions

   Reactor control receives the output of the CTPD, which increases reactor demand.

21.3.7 Reactor Master in Manual

Initial Conditions: The unit is at 100% power, and the operator places the reactor demand hand/automatic station to hand and decreases its output to 90%. Placing the station in hand causes tracking and also transfers T_{ave} control to the feedwater demand subassembly.

1. Reactor Demand Actions

   As the regulating rods insert due to a decrease in the reactor demand setpoint, reactor power decreases. With less energy being deposited into the SGs, header pressure decreases.

2. Integrated Master Actions

   a. The turbine control valves close to restore header pressure to 885 psig.

   b. As the control valves close, turbine load decreases to 90%.

3. CTPD Actions

   The output of the CTPD tracks the decrease in reactor power from NIs.

4. Feedwater Demand Actions

   a. Feedwater flow decreases to 90% following the decrease in the CTPD output.

   b. If a T_{ave} error exists, feedwater demand will be altered to return T_{ave} to 579°F.

TABLE 21-1 AUTOMATIC LOAD LIMITS AND RATE LIMITS

Condition	Limit	Rate
Loss of an RCP	74% power	25%/minute
Loss of a MFP	65% power	25%/minute
Asymmetric Rod	55% power	1%/minute minimum (may be increased by operator)
RCS Flow	Variable (based on total RCS flow)	20%/minute
Both Generator Breakers Open	20% power	20%/minute
Maximum Runback	15% power	20%/minute
Reactor Trip	0%	600%/minute)
Tracking	None	20%/minute
Both SGs on LLL	None	1%/minute (may be decreased by operator)

TABLE 21-2 Control Variables and Calibrating Integrals

Operating Mode	Control Variable	Calibrating Integral
Normal	T_{ave} THP CTP ΔT_C	Reactor LRM Feedwater ΔT_C
Reactor in manual	T_{ave} THP CTP ΔT_C	Feedwater LRM Control Room Operator ΔT_C
Feedwater in manual	T_{ave} THP CTP ΔT_C	Reactor LRM Control Room Operator Control Room Operator
Turbine in manual	T_{ave} THP CTP ΔT_C	Reactor Feedwater Control Room Operator ΔT_C
Low level limits in both SGs	T_{ave} THP CTP ΔT_C	Uncontrolled LRM and TBVs Reactor Uncontrolled

TABLE 21-3 Turbine Loading Status Flag

CTP Demand	Flag	When Applicable	Effects
< 10%	FALSE	Always	Turbine cannot take THP control
10% to 20%	FALSE	UNLOAD depressed	No Mwe tracking TBVs control at setpoint
10% to 20%	TRUE	LOAD depressed and TBVs zero demand	Turbine can take THP control Mwe tracking
> 20%	TRUE	Always (except load rejection)	TBVs control at setpoint plus 50#

TABLE 21-4 Transfer Functions

T1 - Automatic Load Limits

Overrides the operator's input with a load limit based on the plant condition. Refer to Table 21-1 for a list of plant conditions and corresponding load limits and rates.

T2 - Tracking

Inputs a tracking parameter when the ACS is in tracking. Note that tracking overrides automatic load limits. The tracking rate is 20% per minute.

T3 - Feedwater Control Correction

Selects the parameter to be input to the feedwater integral.

a. Normal - CTP error
b. Reactor in manual - T_{ave} error
c. Reactor in automatic, turbine in manual (turbine load status flag TRUE) - THP error

T4, T5 - Low Level Limit, SG A and B

Transfers feedwater control from flow control to level control.

a. Startup level < 25″
b. Low Level Limits (Startup level < 25″ and T_{ave} < setpoint)

T6 - Startup Feedwater Valves Control

Transfers control of startup feedwater valves to level control on a loss of all RCPs. Level setpoint is 50% on the operating range (95% if the reactor building pressure > 3 psig).

T7 - Turbine Bypass Valve Control Error Selector

When the turbine master is in automatic, THP error controls TBVs. When the turbine master is in manual, SG pressure error controls TBVs.

TABLE 21-4 Transfer Functions (continued)

T8 - Turbine Bypass Valve Bias

Selects the proper bias for TBV setpoint based on plant conditions.

a. Reactor trip - setpoint + 125 psi
b. Turbine load status flag TRUE - setpoint + 50 psi
c. Turbine trip or turbine load status flag FALSE - no bias

T9 - Condenser Vacuum Interlock

Prevents automatic operation of TBVs when condenser vacuum < 7″.

T10 - Reactor Control Correction

Selects the parameter to be input to the reactor integral.

a. Normal - T_{ave} error
b. Both SGs on LLL - CTP error
c. Steam system unavailable while both SGs on LLL - THP error

T11 - Maneuvering Rate

Determines the rate of change of CTP demand for automatic load limiting, tracking, and other plant conditions.

a. Automatic load limiting - as shown in Table 21-1
b. Tracking - 20% per minute
c. LLL - 1% per minute
d. Reactor trip - 600% per minute

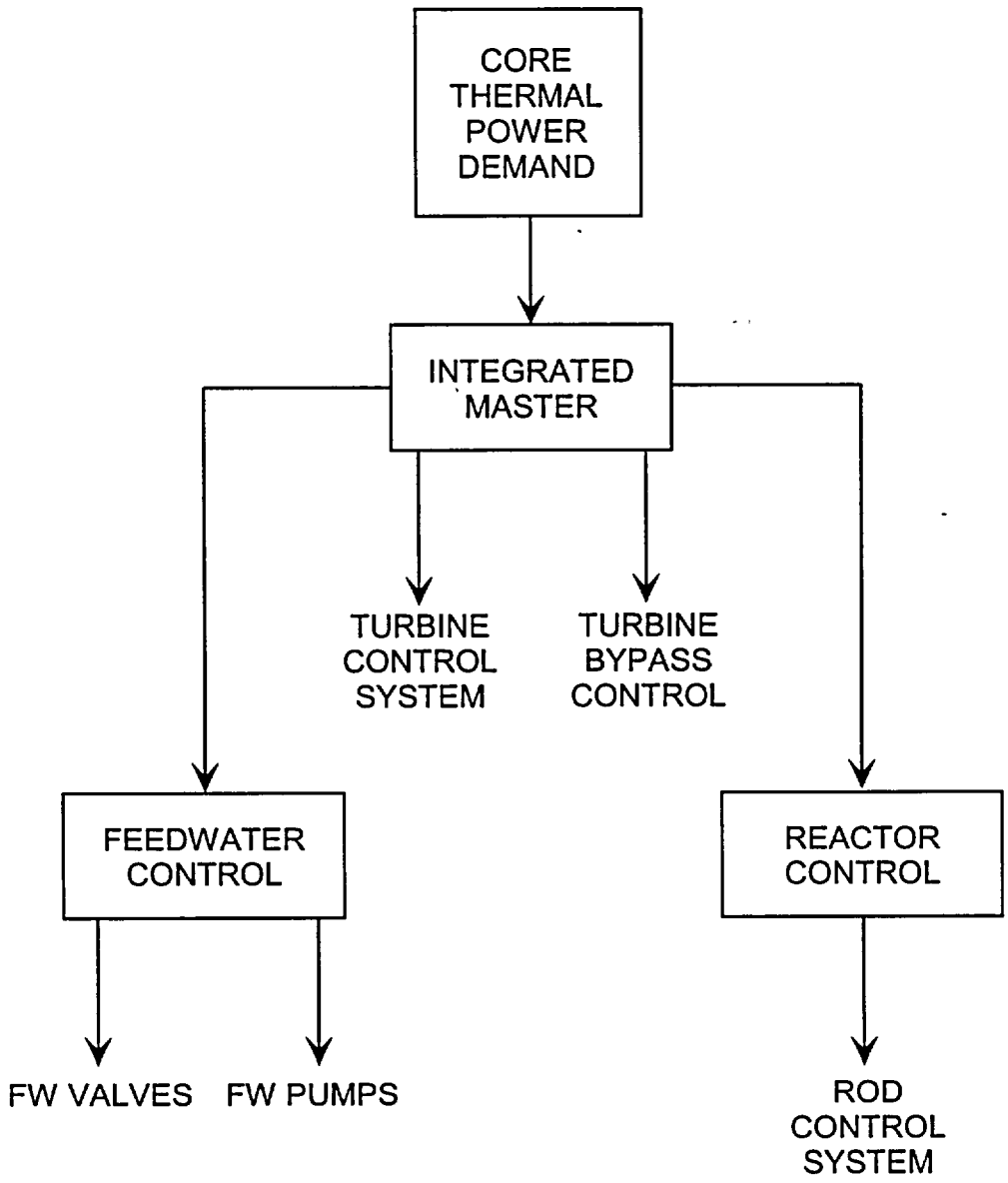


Figure 21-1 ACS Block Diagram

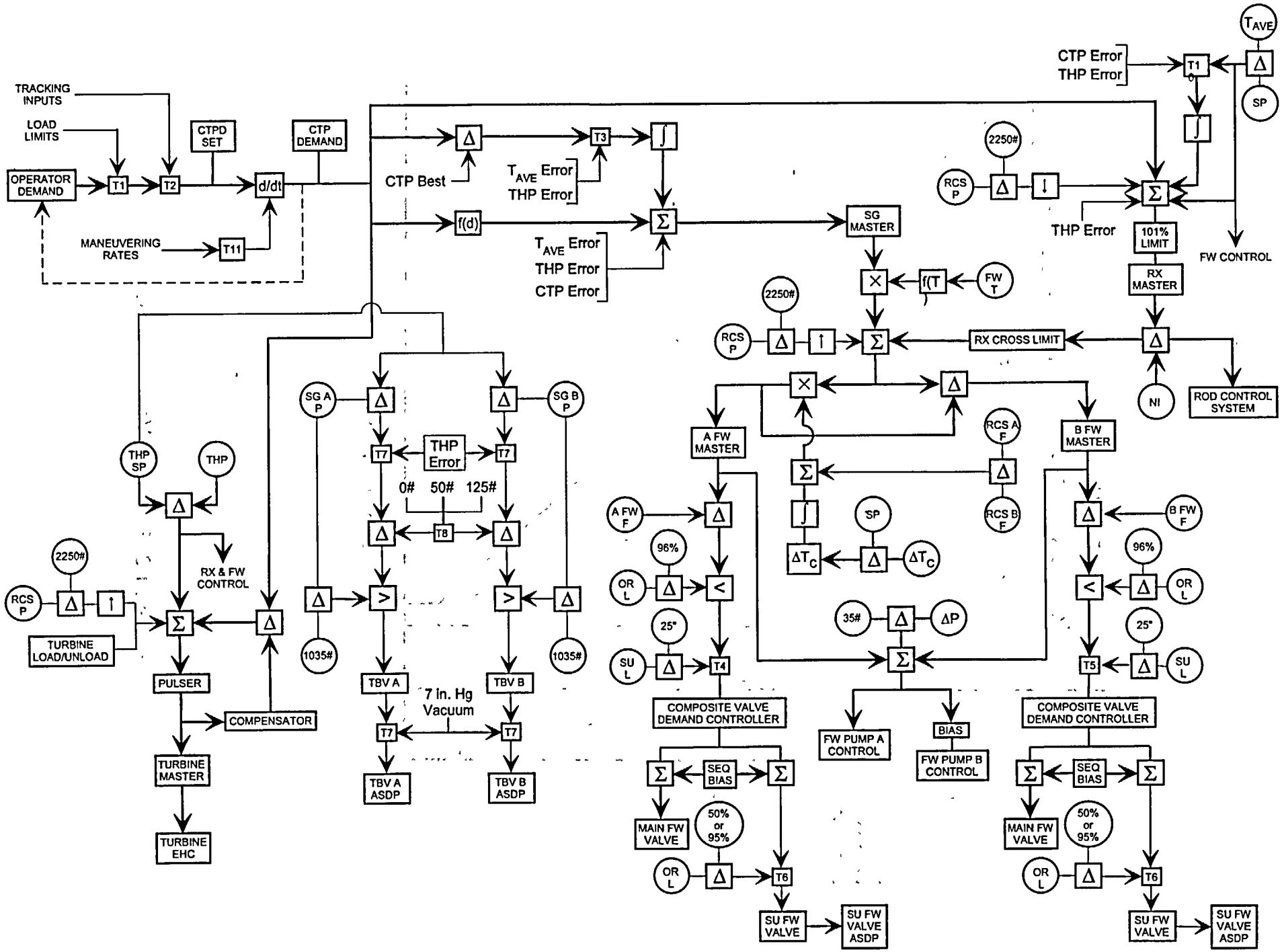


Figure 21-2 Advanced Control System

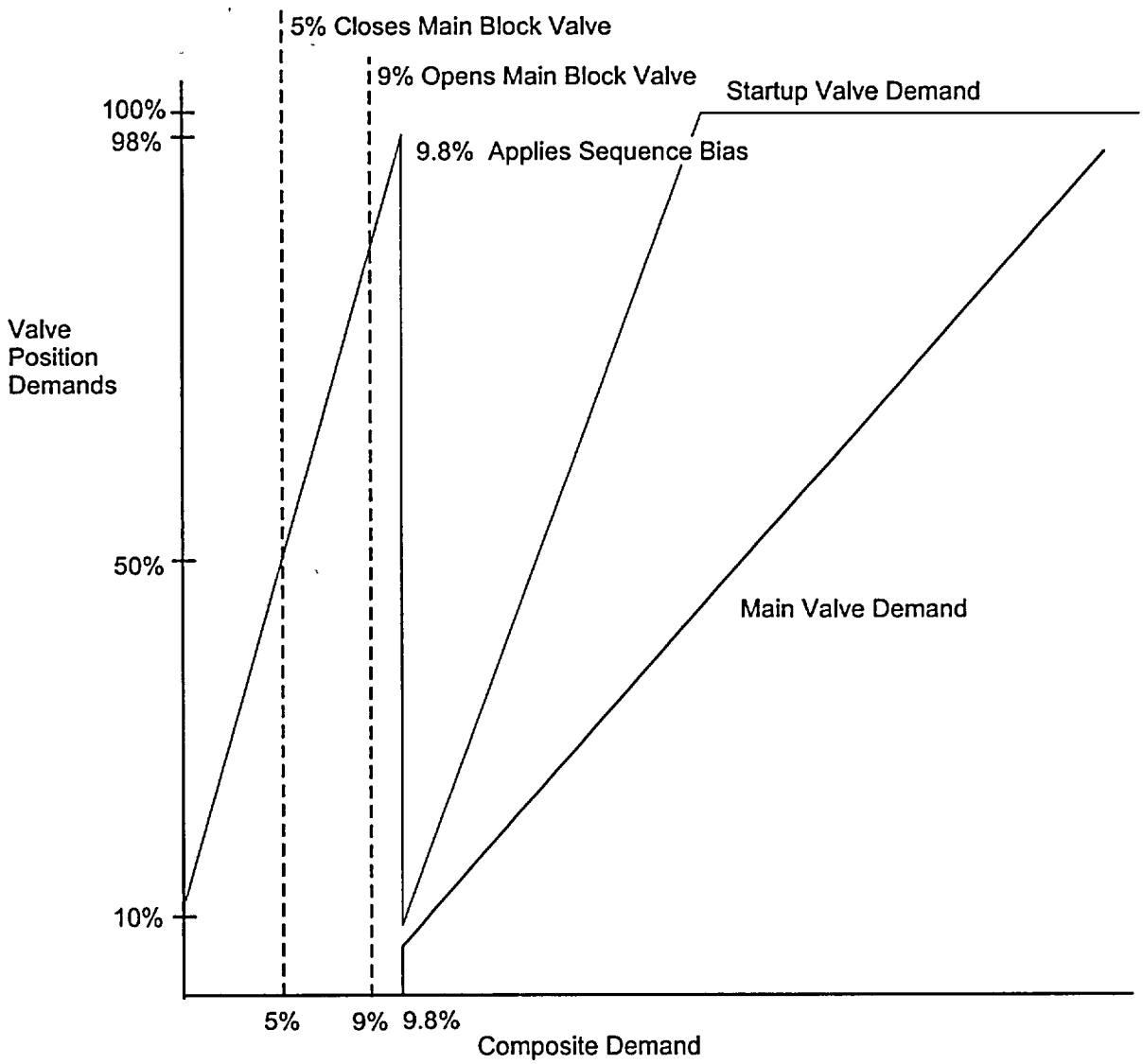


Figure 21-3 Feedwater Valve Operations

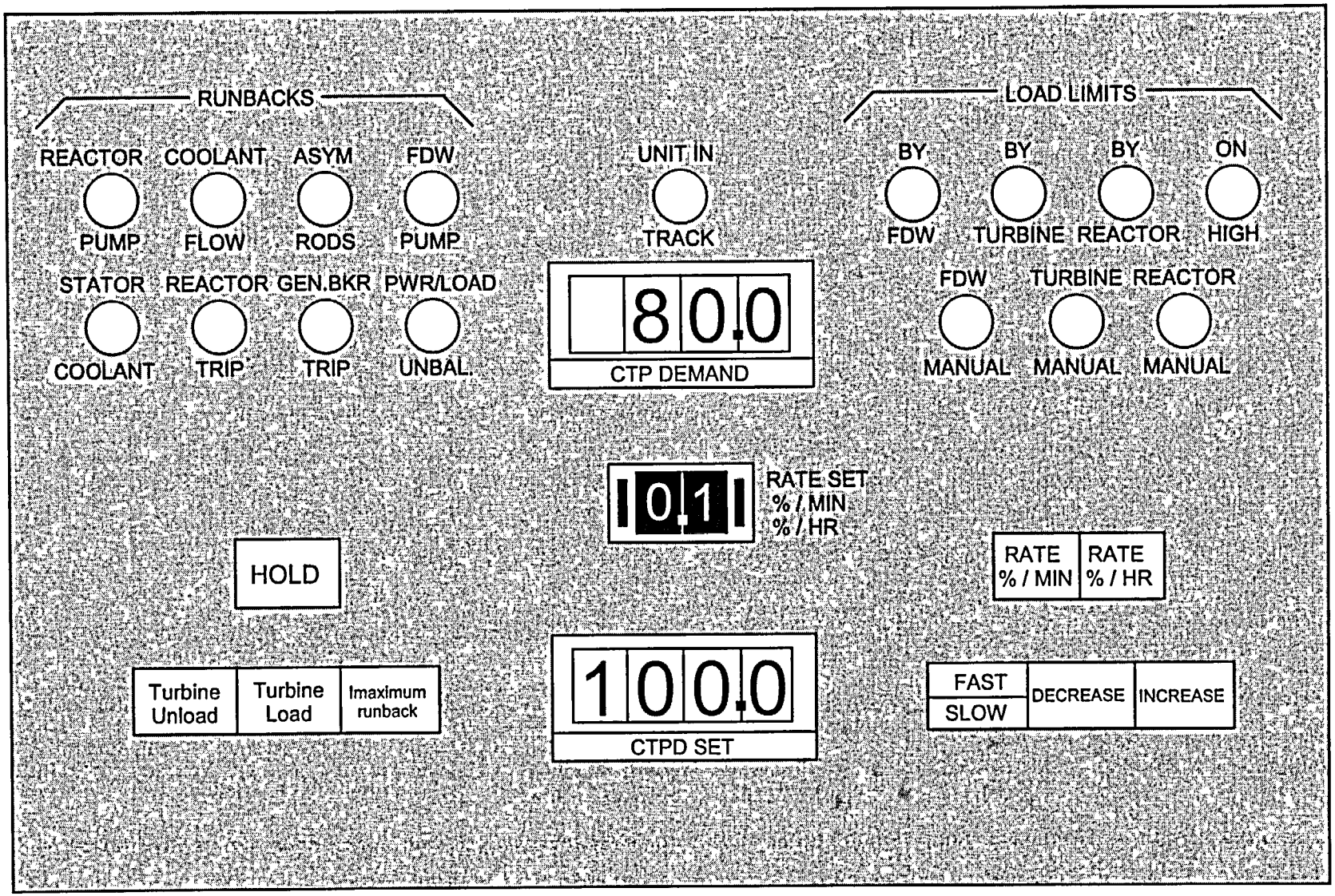


Figure 21-4 Load Control Panel

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CHAPTER 22 Plant Operating Limits

TABLE OF CONTENTS

22.0 PLANT OPERATING LIMITS
22.1 Introduction 22-1
22.2 Shutdown Margin 22-1
22.3 Ejected Rod Worth Limits 22-2
22.4 Linear Heat Rate Limits 22-3
22.4.1 Centerline Fuel Melting Limits 22-3
22.4.2 Emergency Core Cooling Systems Acceptance Criteria 22-4
22.4.3 Loss of Coolant Accident Linear Heat Rate Limits 22-4
22.5 Combined Operating Limits 22-5
22.6 Departure From Nucleate Boiling 22-5
22.6.1 Reactor Coolant Pressure 22-6
22.6.2 Reactor Coolant Temperature 22-6
22.6.3 Reactor Coolant Flow 22-6
22.6.4 Reactor Power and Power Shape 22-6
22.6.5 Hot Channel Factors 22-7
22.6.6 Quadrant Power Tilt 22-8
22.6.7 Departure from Nucleate Boiling Ratio Protection 22-8
22.7 Summary 22-8

TABLE

22-1 Ejected Rod Worths at Various Powers 22-3

LIST OF FIGURES

22-1 Shutdown Margin Rod Index Requirements
22-2 Ejected Rod Worth Rod Index Requirements
22-3 Centerline Fuel Temperature vs. Linear Heat Rate
22-4 Offset Limits
22-5 Linear Heat Rate Rod Index Limits
22-6 Axial Power Shaping Rod Insertion Limits
22-7 Combined Rod Index Limits
22-8 Pool Boiling Curve
22-9 DNBR as a Function of Temperature and Pressure
22-10 DNBR vs. Flow
22-11 DNBR vs. Fuel Height (Cosine Distribution)

22.0 PLANT OPERATING LIMITS

Learning Objectives:

1. Define the following terms:
   a. Shutdown Margin
   b. Nucleate Boiling
   c. Departure from Nucleate Boiling
   d. Departure from Nucleate Boiling Ratio
   e. Power Density (linear heat generation rate)
   f. Axial Flux Imbalance and Axial Offset
   g. Heat Flux Hot Channel Factor
   h. Enthalpy Rise Hot Channel Factor
   i. Quadrant Power Tilt
2. State the basis for requiring a shutdown margin.
3. Explain why the required rod index increases with increasing power.
4. Explain why an ejected rod is worse at high power than low power.
5. List the 5 emergency core cooling acceptance criteria.
6. Discuss the effect on departure from nucleate boiling ratio from changes in the following parameters:
   a. reactor coolant system pressure
   b. reactor coolant system temperature
   c. reactor coolant system flow
   d. reactor power

22.1 Introduction

The transfer of heat from the reactor core presents a difficult problem for the designers of a commercial nuclear power plant. First of all, the core has to generate enough energy to produce the desired electrical output after the heat energy has been converted to mechanical energy. Second, this energy production must be accomplished in a manner that protects the public from the highly radioactive fission products that result from the heat generation process. To protect the public, operational limits that prevent fuel damage are imposed on the core to preclude the release of fission products. These operational limits are shutdown margin, ejected rod protection, linear heat rate limits, and departure from nucleate boiling limits. Shutdown margin requirements ensure that the reactor is or can be shut down and that nuclear excursions will not lead to fuel damage. Ejected rod protection is necessary to prevent nuclear excursions from damaging the fuel when a large rapid reactivity addition takes place. Limits on linear heat rate and departure from nucleate boiling prevent damage to the protective cladding that surrounds the fuel. Each of these limits are discussed in this section.

22.2 Shutdown Margin

Plant technical specifications require a shutdown margin of 1% $\Delta K/K$ and define shutdown margin as follows: shutdown margin shall be the instantaneous amount of reactivity by which the reactor is or would be subcritical from its present condition assuming:

- No change in axial power shaping rod position, and
- All control rod assemblies are fully inserted except for the single rod assembly of the highest reactivity (i.e., most reactive rod), which is assumed to be fully withdrawn.

The shutdown margin requirement is based on the ability to:

1. make the reactor subcritical from all operating conditions,
2. control postulated reactivity transients, and
3. prevent inadvertent criticality from a shutdown

condition.

The shutdown margin requirements apply in all operating modes of the plant. During subcritical operations, the shutdown margin requirements are met by performing a reactivity balance. This reactivity balance algebraically sums all core reactivities:

$$\rho_{net} = \rho_{fuel} + \rho_{xenon} + \rho_{samarium} + \rho_{temperature} + \rho_{rods} + \rho_{boron}$$

The result of the summation (ρ_{net}) represents the core's negative reactivity and its magnitude must be greater than or equal to 1% ΔK/K. A similar calculation can be made when the reactor is at power, but other factors need to be considered. For example, the reactor can be operated with the control rods in almost any position by adjusting the soluble boron concentration. But consider the case where the plant is operating at 100% power and an automatic shutdown (reactor trip) occurs. The trip de-energizes the control rod drive mechanisms, and the rods fall into the core. The amount of negative reactivity added by the rods, when the reactor trips, is proportional to the total rod withdrawal before the trip. Positive reactivity is added by two methods when the trip occurs: (1) the reactor coolant temperature is reduced to its post-trip value, which adds positive reactivity through the moderator temperature coefficient, and (2) positive reactivity is added by the Doppler coefficient as a result of the reduction in fuel temperature when power is decreased. The negative reactivity added by the control rods must be in excess of the positive reactivity added by the moderator temperature and Doppler coefficients. Sufficient control rod negative reactivity can be ensured by placing a minimum insertion limit on the control rods. The rod position required to ensure the minimum shutdown margin is calculated as follows:

1. Available rod worth calculation:

$$\rho_{ARW} = 0.9(\rho_{tcrw} - \rho_{sr})$$

where ρ_{ARW} = available rod worth
ρ_{tcrw} = total control rod worth
ρ_{sr} = most reactive rod worth

(The 0.9 factor in the equation indicates that the available rod worth is reduced by 10% to allow for calculational uncertainty.)

2. Reactivity added as a result of the trip:

$$\rho_{power} = \rho_{mod} + \rho_{doppler}$$

where ρ_{mod} = positive reactivity added by the moderator temperature decrease
$\rho_{doppler}$ = positive reactivity added by the fuel temperature decrease

Note: ρ_{power} = power defect

3. Shutdown margin calculation:

$$\rho_{SM} = \rho_{ARW} + \rho_{power}$$

where: ρ_{SM} = shutdown margin

4. The shutdown margin is converted to a withdrawal limit by use of the rod worth curves. Babcock & Wilcox expresses this limit in terms of rod index, which is equal to the total regulating rod withdrawal percentage. If group 5 is withdrawn to100%, group 6 is withdrawn to 75%, and group 7 is withdrawn to 0%, then the rod index is 175 (100 + 75 + 0). The rod index required to maintain a 1% ΔK/K shutdown margin is shown in Figure 22-1.

22.3 Ejected Rod Worth Limits

The ejection of a control rod from the core results in a rapid reactivity excursion that could increase reactor power to a level that causes fuel

damage. The severity of this accident is dependent on the worth of the ejected rod and the initial power level of the core. Higher power levels are the worst case for this accident because the amount of energy that must be added to cause fuel damage is less than that required at low power levels. The worth of the ejected rod is proportional to its height in the core. When the reactor is shut down, all rods are inserted and, therefore, have the maximum rod worth. However, the maintenance of the required 1% ΔK/K shutdown margin prevents nuclear excursion. As criticality is approached, rods are withdrawn, and the worth of the remaining inserted rods decreases. At criticality, a rod worth of 1% ΔK/K is assumed to be ejected from the core. No core damage results from this accident. When power is escalated, rods are withdrawn to overcome the negative power coefficient; therefore, the maximum rod worth expected at 100% power is 0.5% ΔK/K. To add conservatism, a value of 0.65% ΔK/K is used in the safety analysis. The ejection of a 0.65% ΔK/K rod from 102% power causes fuel damage, but this damage and the subsequent release of activity to the environment are within the limits specified in 10CFR100. The following table summarizes the results of various ejected rod worths from various power levels.

Table 22-1
Ejected Rod Worths at Various Power Levels

Initial Power (%)	Rod Worth (% ΔK/K)	Peak Thermal Power (%)
0.1 (BOL)	1.0	68.4
0.1 (EOL)	1.0	82.9
102 (BOL)*	0.65	135.1
102 (EOL)	0.65	128.1

*If a 0.65% ΔK/K rod is ejected from 102% power at BOL, <5% of the fuel pins in the core will experience departure from nucleate boiling (DNB).

Note: BOL = beginning of core life;
EOL = end of core life.

Sufficient calculations are performed at various power levels with various regulating bank positions so that the ejected rod worths can be plotted against inserted worth and a bounding line can be drawn. The rod position limits that preserve the ejected rod criteria can be established by picking the inserted worths for points on the bounding line corresponding to the ejected rod worth limits for various power levels and using integral rod worth curves. These rod position limits are shown in Figure 22-2.

22.4 Linear Heat Rate Limits

The measure of the heat production in a fuel assembly can be expressed in terms of power per unit length, kilowatts per foot (kW/ft). Limits are placed on the allowable heat production to prevent centerline fuel melting and excessive core damage during a loss of coolant accident (LOCA). In general, centerline fuel melting is prevented by automatic reactor shutdowns (trips) generated by the reactor protection system, and heat rate limits are administratively maintained by the plant operators.

22.4.1 Centerline Fuel Melting Limits

Centerline fuel melting limits are imposed to prevent cladding damage, which could result in the release of fission products to the reactor coolant system. If the fuel melts, the volumetric change from a solid fuel pin to a liquid would cause excessive cladding stresses.

The melting point of the uranium dioxide fuel at the beginning of core life is approximately 5000°F, and this value decreases to approximately 4800°F at the end of core life. As shown in Figure

22-3, fuel temperature is a function of the linear heat rate; therefore, melting and its associated problems can be precluded by limiting kW/ft.

The reactor protection system generates a reactor trip signal based on core axial peaking (offset) to limit the linear heat rate to 20.9 kW/ft. Offset protection is basically a total power limit (heat production) that is reduced by axial flux shape to account for localized excessive heat production.

22.4.2 Emergency Core Cooling Systems Acceptance Criteria

During a large LOCA some cladding damage will occur; however, this cladding damage must be limited to ensure that the health and safety of the public is protected. The amount of damage experienced depends on the initial heat generation of the core and the post-LOCA heat removal by the emergency core cooling systems. A set of emergency core cooling performance acceptance criteria has been developed by the Nuclear Regulatory Commission and is listed below:

1. The peak cladding temperature shall not exceed 2200°F.
2. Only 17% of the cladding thickness shall oxidize.
3. Hydrogen generation shall be limited to 1% of the value that would be generated if all of the fuel cladding underwent a zirconium-water reaction.
4. The core shall remain in a coolable geometry.
5. Long term core cooling must be maintained to remove core decay heat.

These acceptance criteria must be met by the core designer by assuming that the accident starts from 102% power with a power distribution that results in the worst consequences. To comply with the acceptance criteria and LOCA analysis assumptions, the effect of quadrant flux tilts, axial flux tilts, and rod patterns have to be considered. Once worst-case conditions are determined, allowable plant parameters are calculated to ensure that LOCA acceptance criteria are met.

22.4.3 Loss of Coolant Accident Linear Heat Rate Limits

Figures 22-4 through 22-6 outline the restrictions on plant operation required to ensure that the LOCA acceptance criteria are maintained. Each of the figures is discussed below :

1. Offset Limits — The offset limits shown in Figure 22-4 are based on LOCA analyses that have defined the maximum heat rate limit so that cladding temperature will not exceed 2200°F.

NOTE:

$$OFFSET = \frac{power_{top} - power_{bottom}}{power_{total}}$$

2. Rod Insertion Limits — Although the full-power operating position of the group 7 rods is 90% withdrawn, rod insertion can create peaks that result in high local linear heat rates. Figure 22-5 shows power level restrictions required to minimize local power peaking when rods are inserted into the core.

3. Axial Power Shaping Rod Position Limits — Limits are required on axial power shaping rod (APSR) positions to avoid axial power "pinching" effects whereby large peaks are produced at small values of offset. Pinching of the power would result if the regulating rods were inserted into the top of the core and the APSRs were inserted into the bottom of the core. This condition would cause a flux peak near the center of the core. Flux peaks at the center of the core represent a low value of offset. APSR limits are shown in Figure 22-6.

22.5 Combined Operating Limits

Thus far rod position has been limited by shutdown margin requirements, ejected rod worth requirements, and linear heat rate requirements. Unfortunately, one rod index will not necessarily satisfy all requirements. For example, a rod index of 125 at 30% power will meet linear heat rate and shutdown margin requirements, but this rod index violates ejected rod worth criteria. Therefore, a rod index curve that satisfies all criteria is provided for plant operations personnel. This curve is shown in Figure 22-7.

22.6 Departure From Nucleate Boiling

As previously stated, heat that is produced within the fuel pellets must be removed. In most commercial reactors, the heat is transferred across the cladding to water. In a pressurized water reactor, the normal method of heat transfer is forced convection. The reactor coolant pumps circulate coolant through the core to remove its heat; however, at high power levels heat transfer is also accomplished by a process known as nucleate boiling. Nucleate boiling is the formation of steam bubbles on the surface of the fuel rod at a rate which allows the bubbles to be swept into the flow channel where they collapse. This process produces an agitated flow that increases the heat transfer rate across the fuel cladding. As long as only nucleate boiling is occurring, the heat removal from the core is ensured. The effects of exceeding the nucleate boiling region can be illustrated by examining the heat transfer mechanisms that occur if a heated wire is inserted into a pool of boiling liquid. Traditionally, a graph of heat transfer versus ΔT is plotted for this experiment. This graph is illustrated in Figure 22-8. The difference between the wire's surface temperature (T_{wall}) and the pool's saturation temperature (T_{sat}) is plotted on the horizontal axis, whereas the rate of heat transfer per unit area (Btu/hr-ft^2) is plotted on the vertical axis. When this is done, five regions of heat transfer can be identified. The first region, from point A to point B, is called the natural convection regime. Here, the hot fluid in contact with the surface of the wire rises and is replaced with colder water. As the heat input to the wire is increased, the amount of heat transferred increases with the temperature differential between the wire's surface and the saturation temperature. The second region, from point B to point C, is called the nucleate boiling regime. In this area of heat transfer, bubbles of steam vapor start to form on the surface of the wire. In the nucleate boiling process, a high rate of heat transfer is obtained from intense convection currents at the heating surface and from bubble agitation and liquid movement to the surface as vapor bubbles leave the heated area. The following two points should be considered in this discussion: (1) as seen in Figure 22-8, the increase in ΔT required for a large increase in heat transfer is very small, and (2) the bubbles that form are actually superheated steam bubbles.

At point C on the curve, the heat transfer levels off, and further increases in the heat input into the wire will actually cause a decrease in the heat transfer rate. Point C is known as the departure from nucleate boiling (DNB). The formation of steam bubbles is at a greater rate, and the bubbles tend to blanket the wire. This blanket of steam bubbles prevents the entry of water to the surface of the wire, and the heat transfer rate is impeded. This third region, from point C to point D, is known as the partial film boiling regime. The fourth region of heat transfer is represented by the curve that connects point D to point E. This region is known as the stable film boiling regime. Here, a continuous steam blanket covers the wire. The final region, point E to point F, is known as the film and radiation regime. Here, the temperature of the heated surface of the wire is so high that radiative heat transfer becomes significant. Thermal radiation is a strong function of temperature, and

the heat transfer rate rises rapidly with ΔT.

How does a heated wire in a pool of boiling water relate to a forced convection system containing thousands of fuel rods? First the effect of exceeding DNB is very serious. As seen in Figure 22-8 , the heat transfer rate drops off as point C is passed; therefore, the temperature of the wire's surface will increase. If this happens to the surface of the fuel pin, the probability of cladding destruction is very high. Second, certain operational occurrences can result in a loss of forced coolant flow; the core can then be thought of as thousands of heated wires in a pool. Finally, if the independent variable on the curve is reversed and heat generation is increased, a level where DNB occurs will be reached and cladding temperatures will become unacceptably high. Once this level is known, protection to prevent exceeding DNB can be provided. DNB protection in PWRs is provided by limiting the allowable heat generation rate or heat flux, and a part of this protection lies in defining the ratio of the heat flux required for DNB to the actual heat flux reached in any given situation. This ratio is known as the departure from nucleate boiling ratio (DNBR) and is defined as:

$$DNBR = \frac{\text{heat flux required for DNB}}{\text{actual heat flux}}$$

Values of DNBR greater than 1.0 are acceptable; however, values equal to or less than 1.0 indicate the departure from nucleate boiling. Many factors affect the allowable heat flux in the core, which, in turn, affects the DNBR of the core. Among these factors are reactor coolant pressure, reactor coolant temperature, reactor coolant flow, and reactor power level and power shape. Each of these factors and their effects are discussed in the following sections.

22.6.1 Reactor Coolant Pressure

The heat flux required for DNB can be increased by a higher reactor coolant system pressure. Boiling can be prevented by increasing the pressure on the fluid, and, if the actual heat flux is held constant, DNBR will increase. Conversely, if pressure decreases, then DNBR will decrease. DNBR can be held constant with a higher actual heat flux if the heat flux required for DNB is increased.

22.6.2 Reactor Coolant Temperature

During normal operations, the temperature of the reactor coolant is below the saturation temperature for the existing reactor coolant pressure. If temperature is increased, then the margin to saturation will decrease. This decrease in margin is normal as water temperature increases from the bottom to the top of the core because of heat addition from the fuel. At some elevation in the core, nucleate boiling takes place. If the inlet temperature is increased with core power remaining constant, the the onset of nucleate boiling will occur at a lower elevation; therefore, with continued heat addition from the core, the actual heat flux will decrease the DNBR. Figure 22-9 illustrates the effects of temperature and pressure on DNBR.

22.6.3 Reactor Coolant Flow

The flow of coolant through the core is a function of the number of pumps that are in operation. Allowable pump combinations are two pumps per loop, three-pump operation, and one pump per loop. These operational modes determine the coolant flow and the heat removal from the core. The greater the amount of flow, the higher the allowable power (heat generation). In other words, DNBR increases with an increase in RCS flow. Figure 22-10 illustrates the variation of DNBR with RCS flow.

22.6.4 Reactor Power and Power Shape

The generation of heat in the core is directly proportional to the fission rate or power level of the core. If only the total core power is considered, as power is increased, the actual heat flux increases, which will cause a decrease in DNBR. Both the power shape and the total power must be considered. If a cosine flux distribution is assumed, the change in DNBR versus core height is as shown in Figure 22-11. The shape of the curve can be explained by examining the changes in heat addition as the coolant travels up through the core. Relatively cold water enters the bottom of the core at a pressure that is higher than the pressure at any other point in the core. This combination causes the DNBR to be very large. As the coolant travels up through the fuel, it picks up additional energy from the core because power is increasing as a function of the cosine flux distribution. Also, the higher the elavation, the greater the pressure drop and the lower the pressure at any particular point. The effect of all these parameters is to cause a rapid drop in DNBR. The decrease in DNBR continues to just slightly past core midplane to the point that represents the worst combinations of temperature, total heat flux and pressure. The increase in DNBR beyond this point results from the decreased heat flux input from the core. Thus far in this discussion, the effect of DNBR parameters has been considered in a macroscopic manner. In reality, DNBR is considered as a very localized occurrence. The heat flux required for DNB is lower at locations having the lowest flow and pressure along with the highest temperature and flux. Conservatisms are added to these parameters to create a theoretical location called the hot channel. If the actual heat flux in this channel can be prevented from exceeding the DNB heat flux, then the designer can be assured that the core is protected against low DNBR conditions.

22.6.5 Hot Channel Factors

The minimum allowable DNBR for the hot channel is 1.25 and is based on the Babcock & Wilcox BXC correlation of DNBR. This section discusses the factors that were considered in the evaluation of DNBR.

1. Core Flow — The flow rate of reactor coolant is measured in the hot leg of the RCS, but the measured system flow rate does not equal the flow available for core heat removal because of bypass paths in the reactor vessel. Flow through the core has been determined using vessel models. Data from the tests have been applied to computer analysis of total core flow. A value of 95.2% of total system flow is used.

2. Bundle Flow — Once the value of core flow is known, a value of flow through the limiting bundle (hot channel) must be calculated. The amount of flow is determined by such variables as pressure drops across spacer grids and reduction in areas because of manufacturing tolerances and fuel rod bowing. The hot channel is assumed to receive only 95% of normal fuel assembly flow.

3. Radial Hot Channel Factor — The ideal radial flux shape would be a cosine flux distribution; however, inserted control rods, inserted axial power shaping rods, and the fact that power peaks can exist at the edges prevent the perfect cosine distribution from existing. To account for the variation in heat flux caused by the variation in radial neutron flux, the term radial hot channel factor has been defined as:

$$F_R^N = \frac{\textit{average heat flux of the hot channel}}{\textit{average heat flux in the core}}$$

Values greater than 1.0 indicate that a radial power peak exists.

4. Axial Hot Channel Factor — The axial flux shape of the reactor also influences the amount of heat deposited in the coolant and the approach to DNB. Therefore, an axial hot channel factor has been defined as:

$$F_Z^N = \frac{\text{maximum heat flux of the hot channel}}{\text{average heat flux of the hot channel}}$$

5. Nuclear Heat Flux Hot Channel Factor — Radial and axial peaking factors are combined to yield a total nuclear heat flux hot channel factor.

$$F_Q = F_R^N \times F_Z^N$$
$$= \frac{\text{maximum heat flux in the hot channel}}{\text{average heat flux in the core}}$$

6. Enthalpy Rise Hot Channel Factor — Radial and axial hot channel factors by themselves do not give the designer the ability to predict DNB. For example, a high heat flux resulting from large radial and axial peaks can be tolerated if enough coolant flow is available. Enthalpy rise hot channel factor was developed to account for both heat addition and flow.

$$F_{\Delta H}^N = \frac{\text{maximum coolant enthalpy rise}}{\text{average coolant enthalpy rise}}$$

The worst case flow, pressure, and hot channel factors are combined in the calculation of hot channel DNBR. If, as previously stated, protection can be shown for this theoretical channel, then all locations in the core can be protected. DNBR protection is described in Section 22.6.7.

22.6.6 Quadrant Power Tilt

The quadrant power tilt (QPT) is a gross indication of whether the core is operating with an acceptable radial power distribution.

$$QPT = 100\left(\frac{\text{power in any quadrant}}{\text{average quadrant power}} - 1\right)$$

An unacceptably large QPT requires verification of the nuclear heat flux and enthalpy rise hot channel factors by the plant computer utilizing incore detector data.

22.6.7 Departure from Nucleate Boiling Ratio Protection

The reactor protection system (RPS) implements DNBR protection during normal and moderate frequency abnormal events. The 177 fuel assembly plants are protected by the following trips:

1. Flux/imbalance/flow
2. Flux/pumps
3. Variable Low Pressure

NOTE:

Axial Flux Imbalance = % $power_{top}$ - % $power_{bottom}$

Chapter 10.1 contains detailed RPS information. Hot channel factor values are calculated and used in the generation of reactor protection system setpoints. The values of F_Q and $F_{\Delta H}^N$ are explicitly checked by the plant computer utilizing incore detector data only when QPT or some other power distribution specification is not satisfied.

22.7 Summary

Operating limits are imposed to ensure that fuel damage is minimized. Minimizing fuel damage is necessary to protect the safety of the public. Limits on shutdown margin, ejected rod worth, linear heat rate, and departure from nucleate boiling are required to protect the core. A shutdown margin requirement of 1% ΔK/K is maintained by imposing a minimum rod withdrawal position that ensures that the positive reactivity addition caused by the power deficit during a reactor trip is overcome by the negative reactivity insertion of the rods. This rod position is a function of power and is maintained by the rod index curve. Ejected rod worth is also minimized by rod insertion limits. The greater the amount of rod withdrawal, the less the worth of potential ejected rods. The rod index curve also preserves ejected rod worth limits.

Linear heat rate requirements are based on centerline fuel melting and heat generation limits necessary to meet the acceptance criteria of a loss of coolant accident. The former requirement is met by automatic action of the reactor protection system, and the latter is met by administratively controlling power and power peaking factors. Peaking is controlled by limits on allowable axial power distribution, control rod position, and axial power shaping rod position. The departure from nucleate boiling requirements ensure that the ability to transfer heat from the cladding is maintained. DNBR limits are maintained by the RPS and the monitoring of peaking in the core.

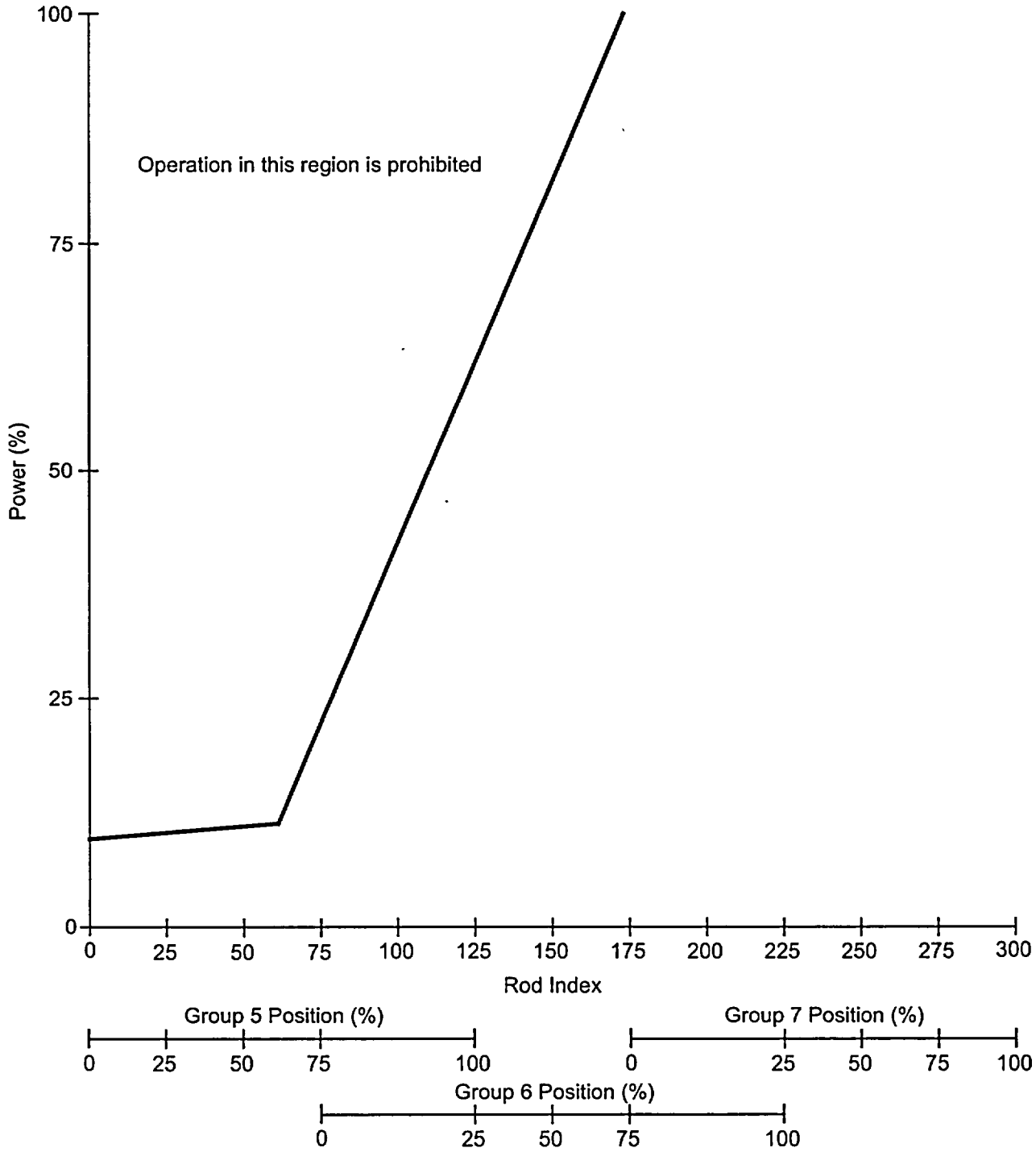


Figure 22-1 Shutdown Margin Rod Index Requirements

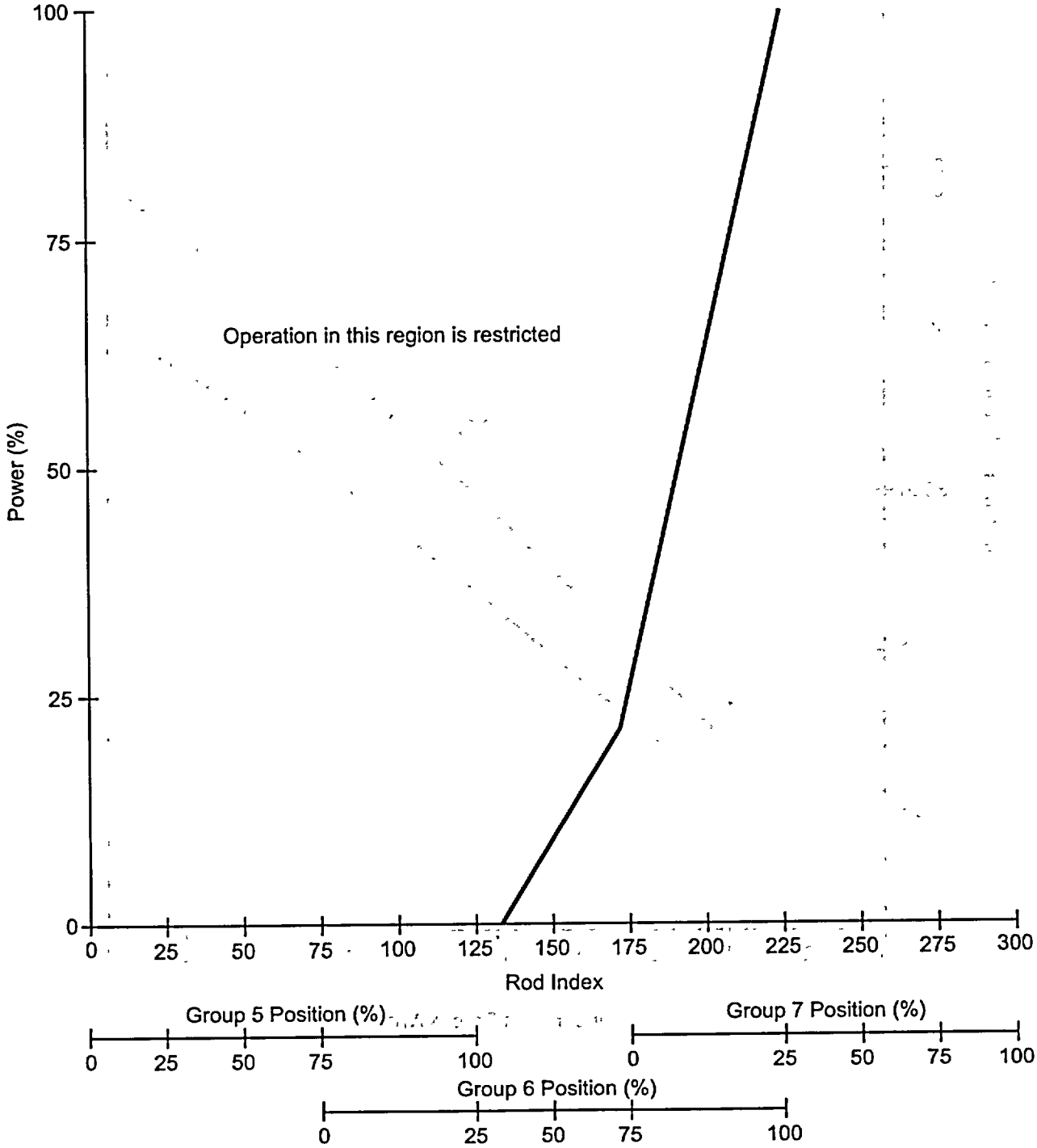


Figure 22-2 Ejected Rod Worth Rod Index Requirements

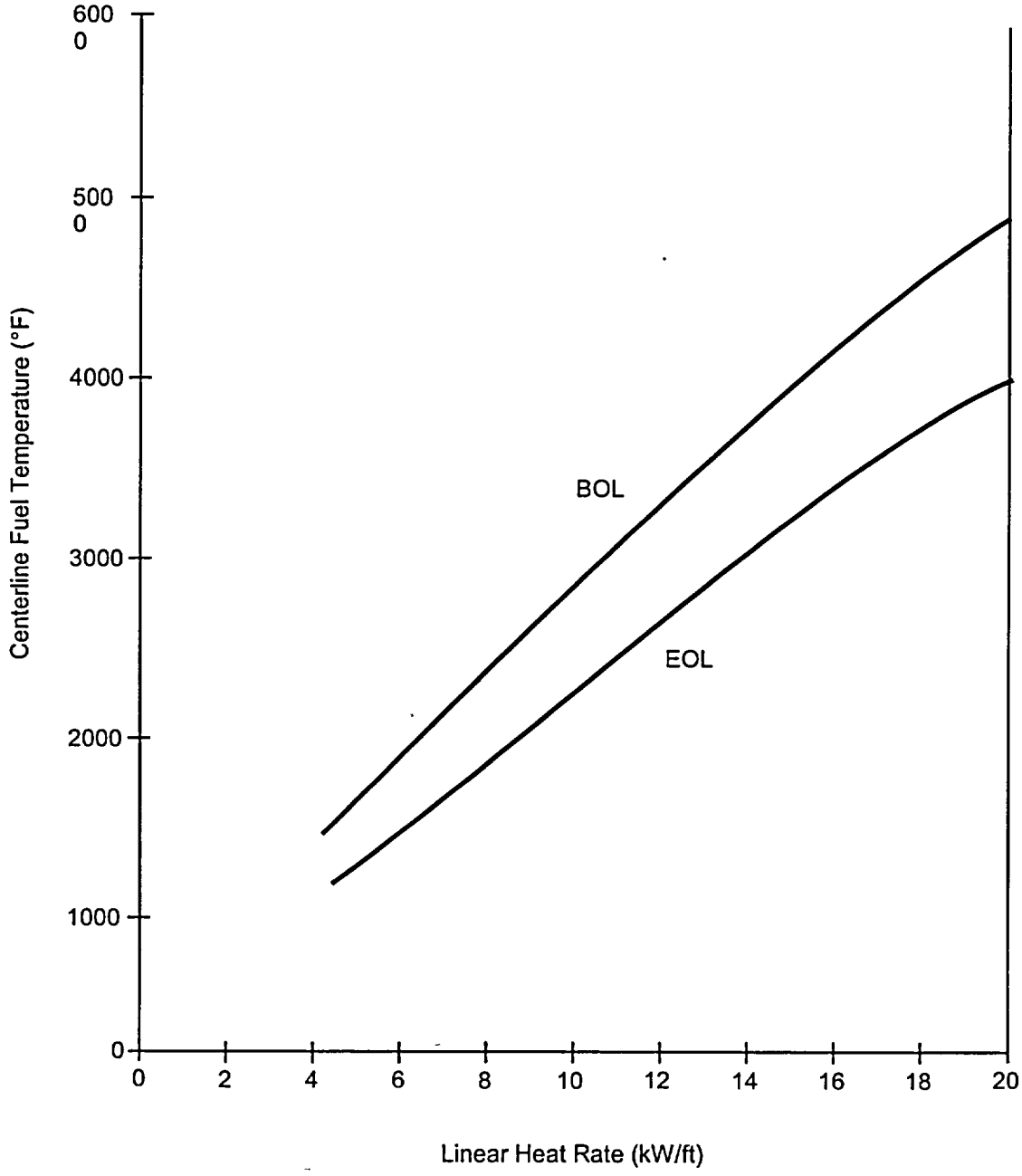


Figure 22-3 Centerline Fuel Temperature vs. Linear Heat Rate

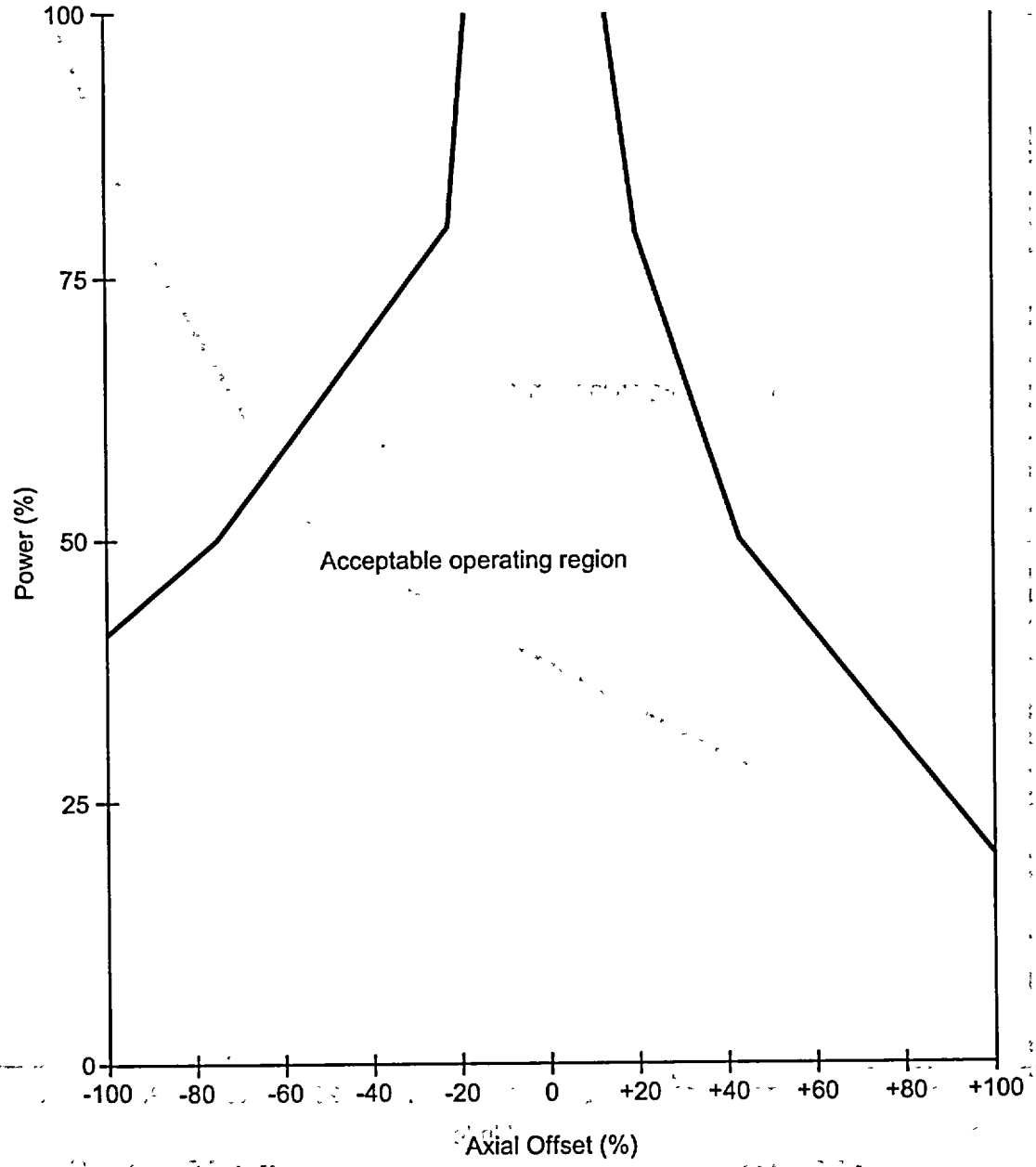


Figure 22-4 Offset Limits

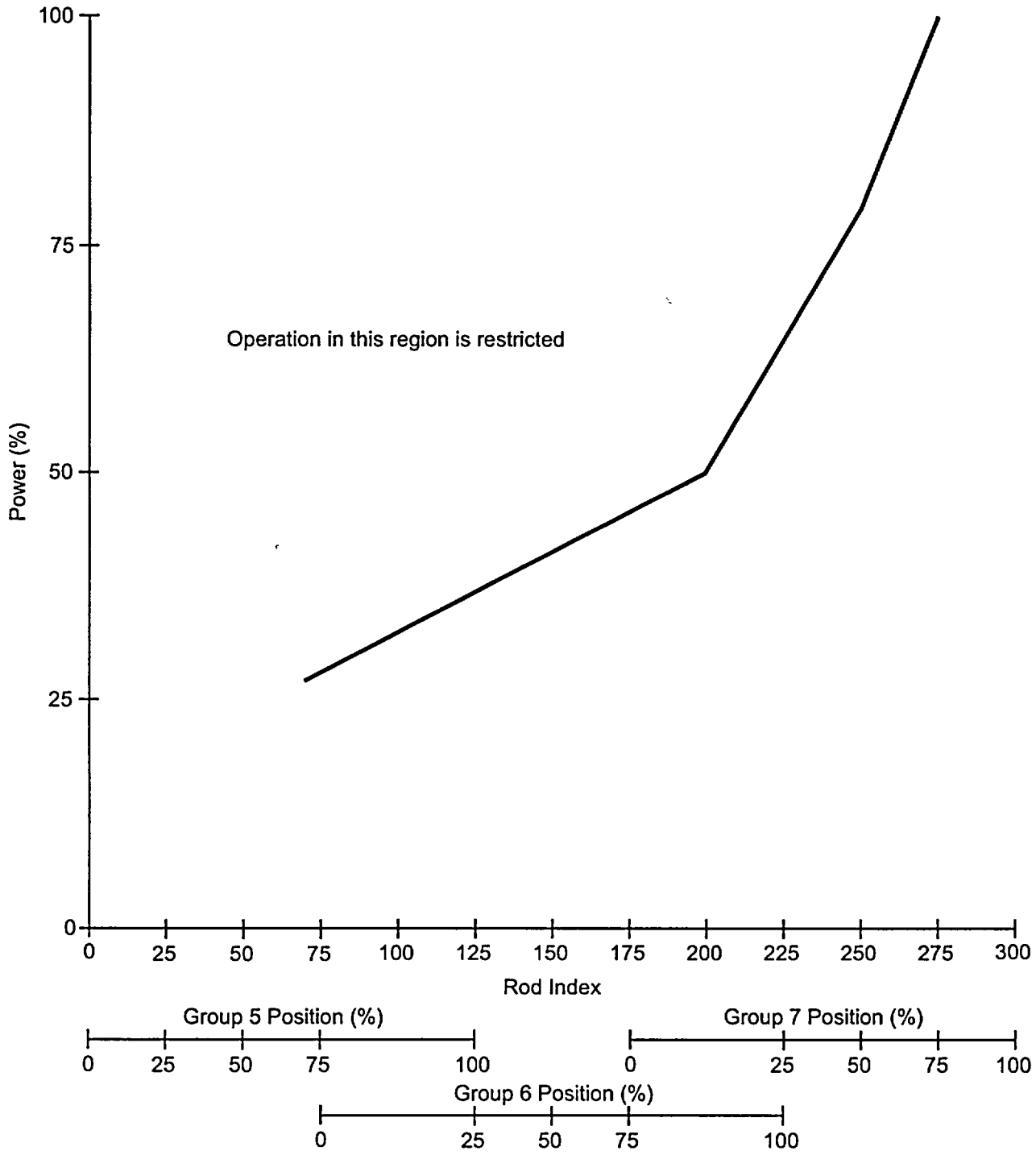


Figure 22-5 Linear Heat Rate Rod Index Requirements

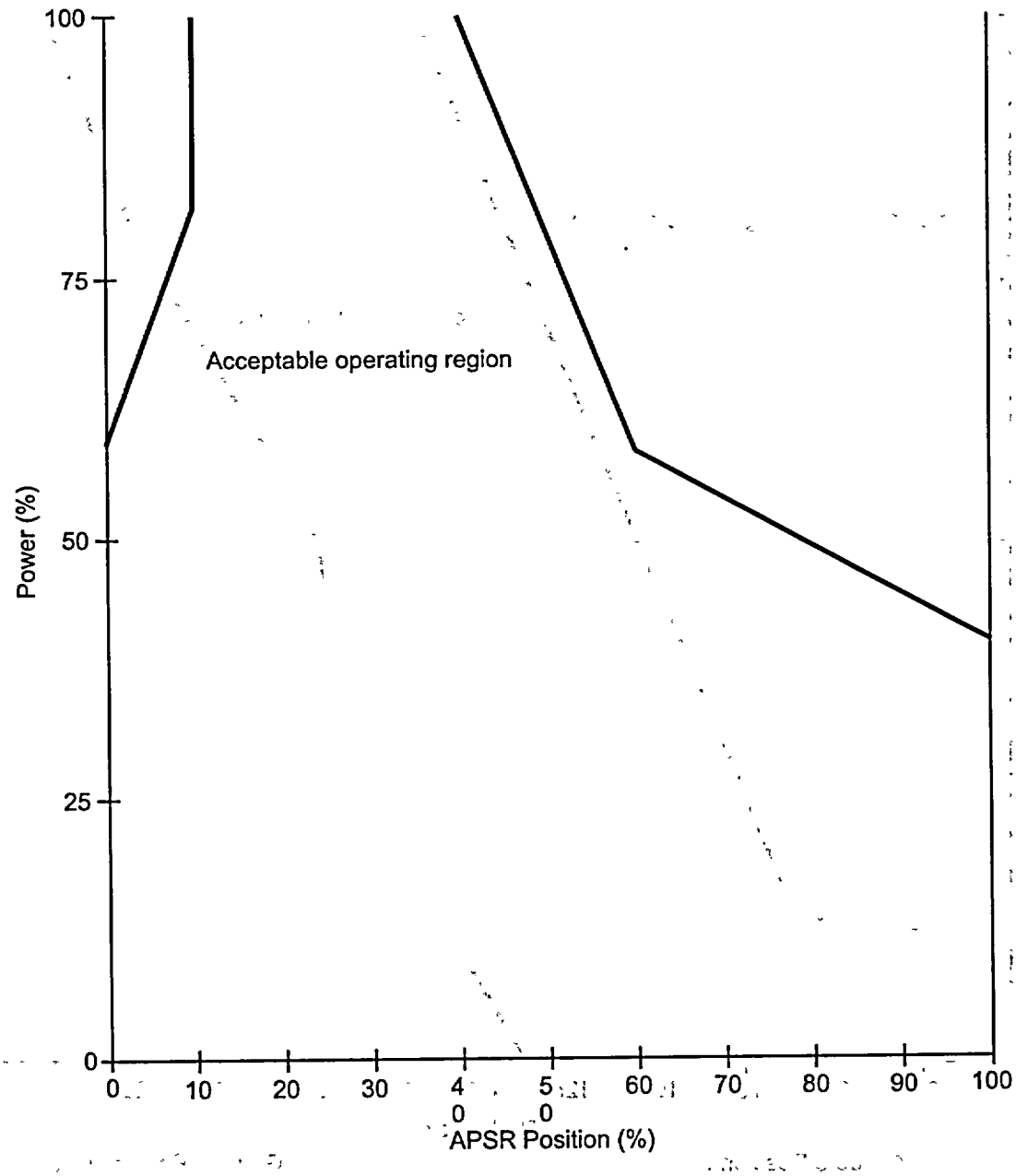


Figure 22-6 Axial Power Shaping Rod Insertion Limits

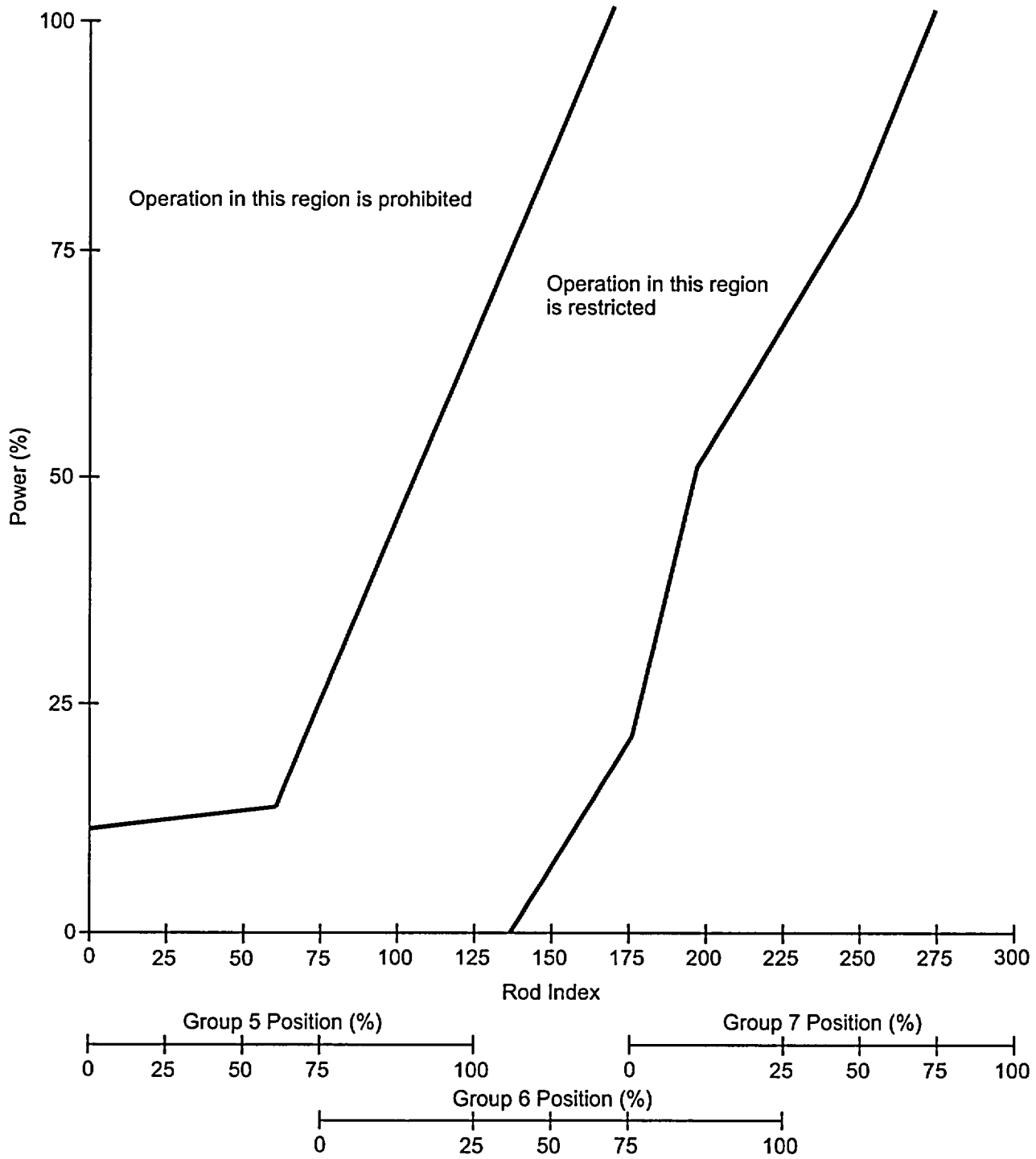
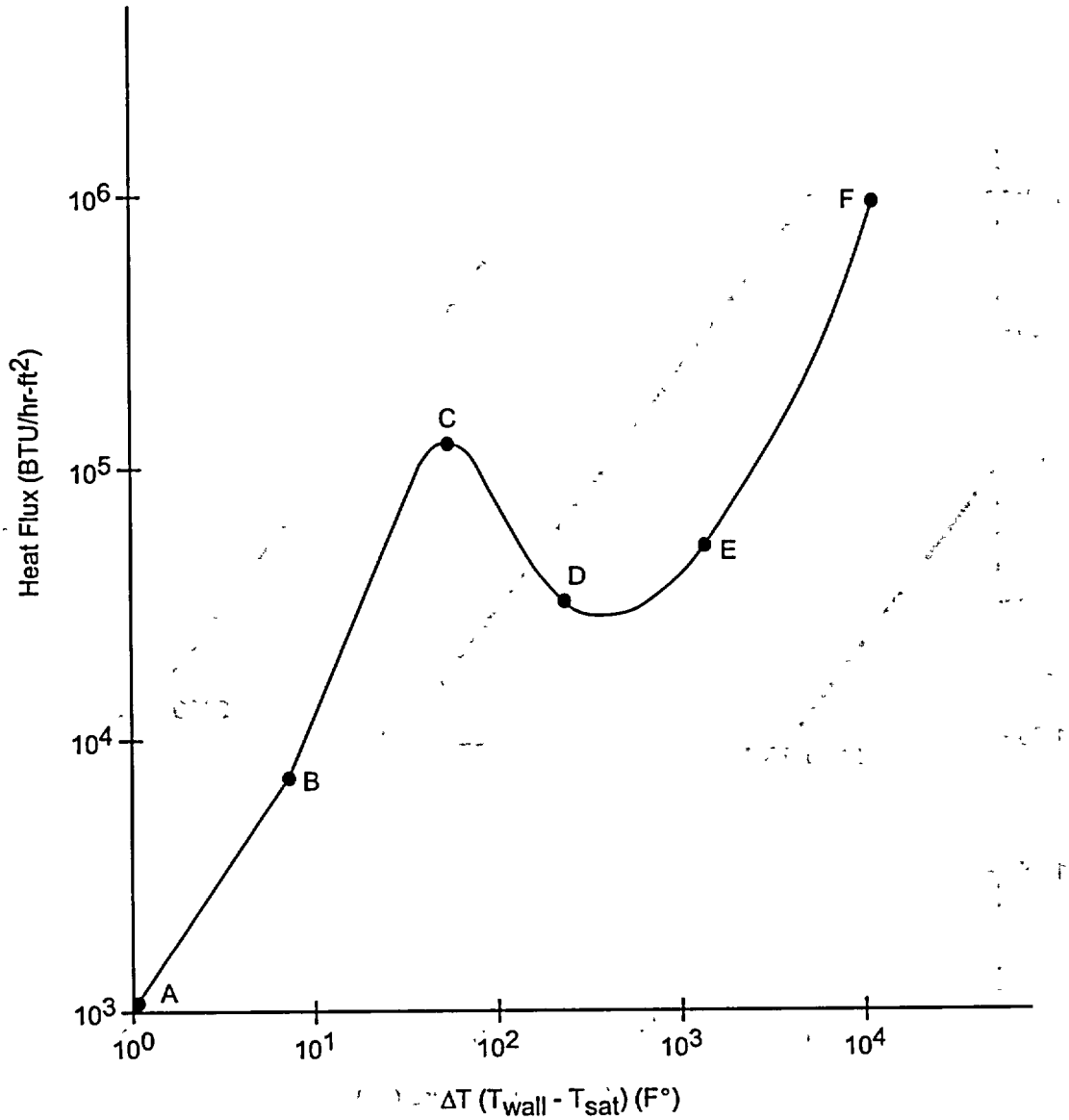


Figure 22-7 Combined Rod Index Limits



A to B – Natural Convection Regime
B to C – Nucleate Boiling Regime
C – Departure From Nucleate Boiling
C to D – Partial Film Boiling Regime
D to E – Stable Film Boiling Regime
E to F – Film Radiation Regime

Figure 22-8 Pool Boiling Curve

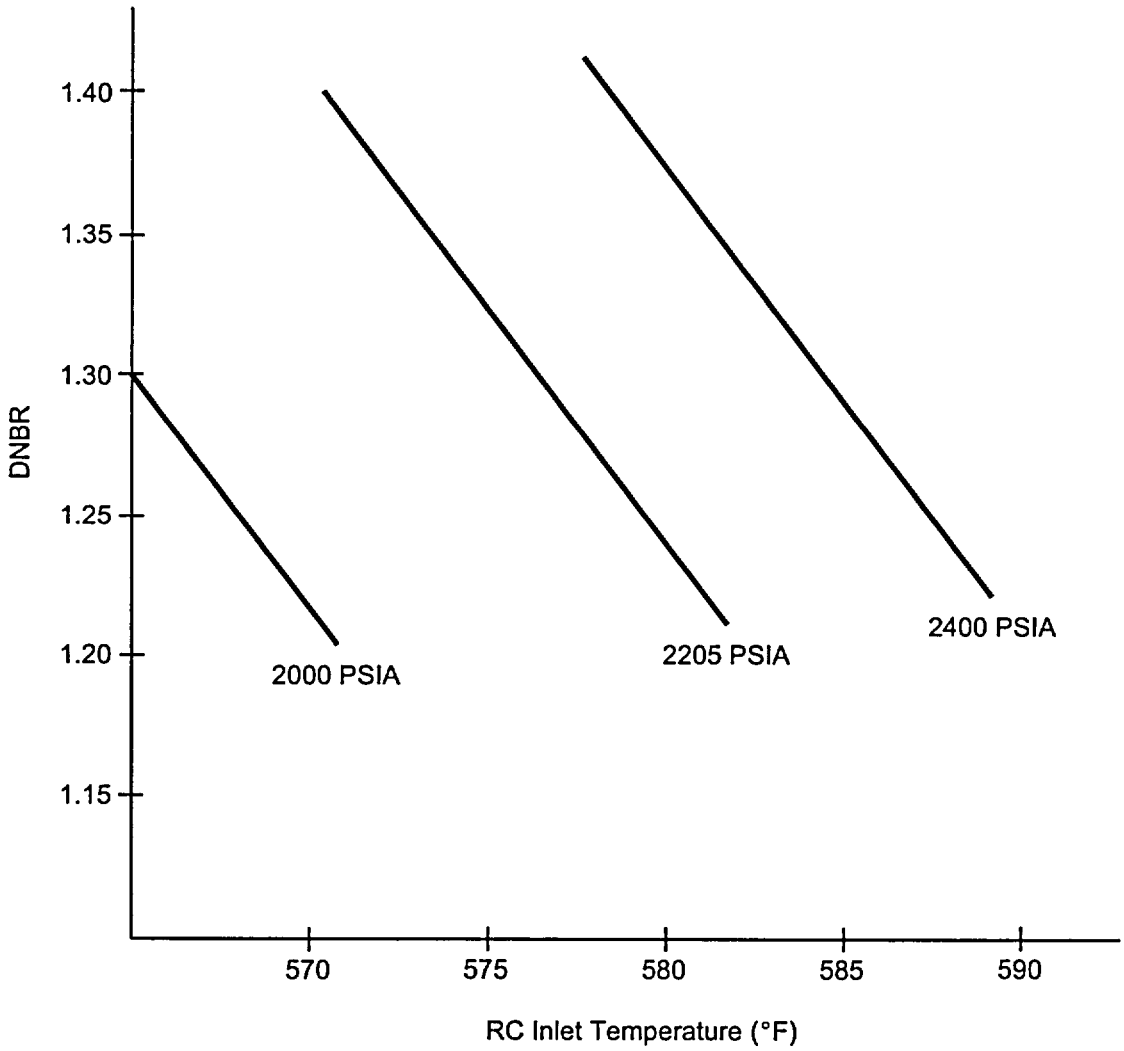


Figure 22-9 DNBR as a Function of Temperature and Pressure

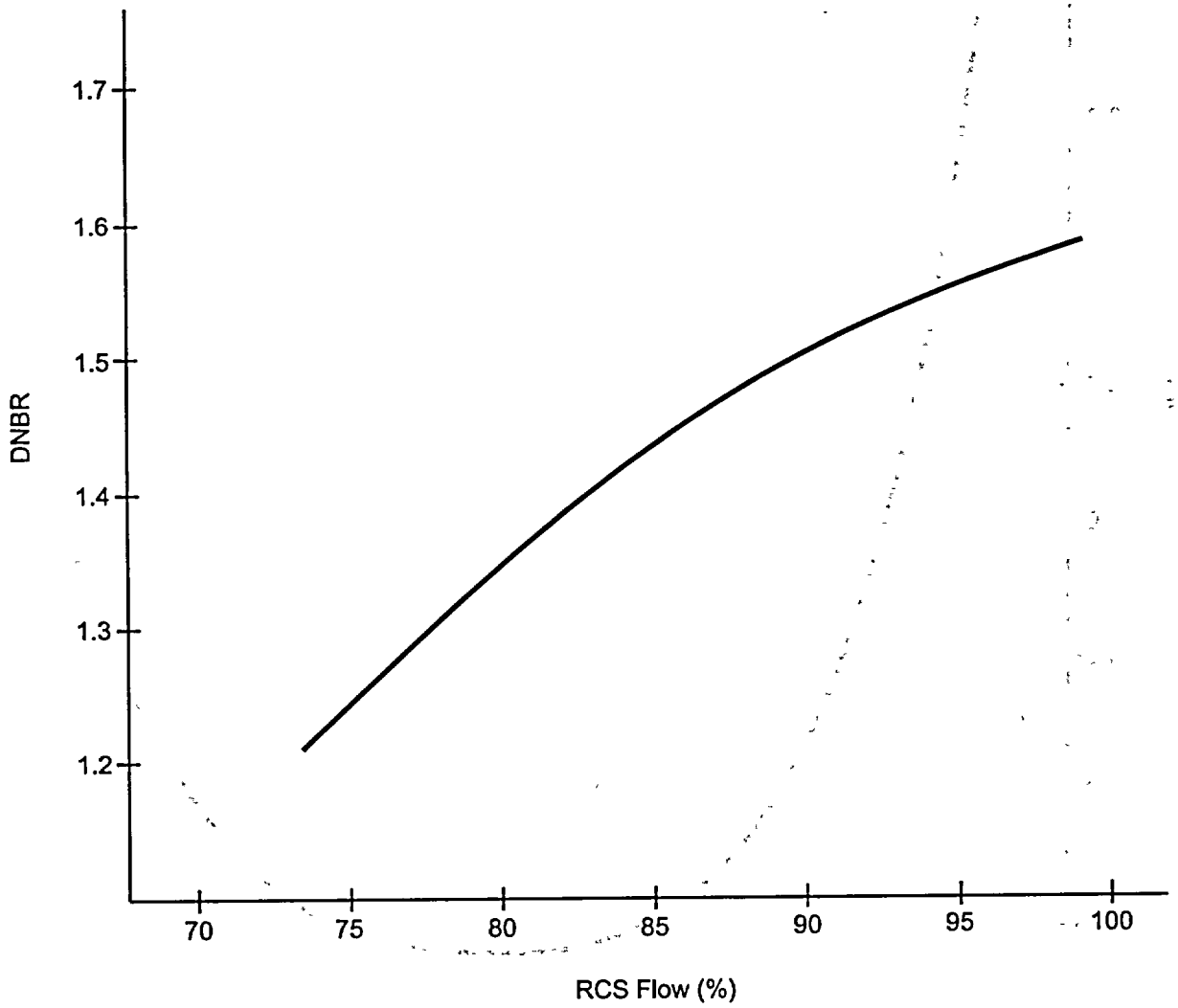


Figure 22-10 DNBR vs. Flow

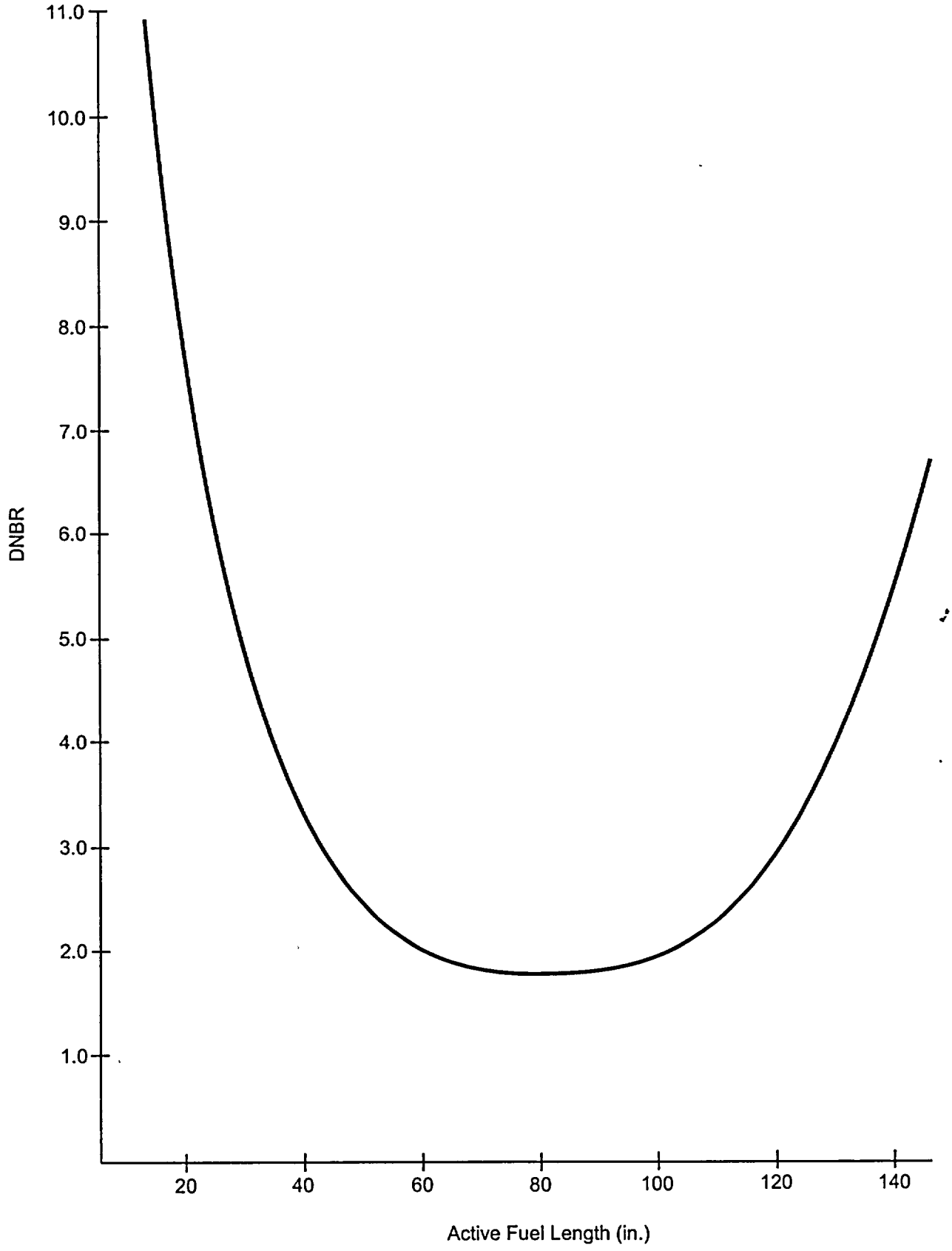


Figure 22-11 DNBR vs. Fuel Height (Cosine Distribution)