ENTERGY

Entergy Operations, Inc.
River Bend Station
5485 U.S. Highway 61
P.O. Box 220
St. Francisville, LA 70775
(504) 381-4374
FAX (504) 381-4872

JOHN R. McGAHA, JR.
Vice President
Operations

September 6, 1995

U. S. Nuclear Regulatory Commission
Region IV
611 Ryan Plaza Drive, Suite 400
Arlington, TX 76011

Subject: River Bend Station Submittal of Formal NRC Examination Comments
River Bend Station - Unit 1
Docket No. 50-458

File No.: G9.5, G1.41.26

RBF1-95-214
RBG-41935
RBEXEC-95-137

Gentlemen:

In accordance with NUREG-1021, Revision 7, Supplement 1(ES-402, Attachment 3), enclosed are four questions submitted for your review and consideration. The enclosed questions (and supporting documentation) are a result of the NRC administered licensing examination conducted at the River Bend Station the week of August 18, 1995. The chief examiner during the examination was Mr. Howard Bundy.

If you have any questions regarding the attached, please contact Mr. L. Grant Lewis at (504) 381-4752.

Sincerely,

Mike Sellman
for JR McGaha

JRM/JJF
enclosure

cc: U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
Mail Stop P1-37
Washington, DC 20555

NRC Sr. Resident Inspector
P. O. Box 1051
St. Francisville, LA 70775

David Wigginton
NRR Project Manager
U. S. Nuclear Regulatory Commission
NRR Mail Stop 13-H-3, One White Flint North
11555 Rockville Pike
Rockville, MD 20852

Mr. Howard Bundy
U. S. Nuclear Regulatory Commission
Region IV
611 Ryan Plaza Drive, Suite 400
Arlington, TX 76011

1) Question #24 RO/#27 SRO

If total feedwater flow drops below the RRS interlock level, the Reactor Recirculation pumps will downshift to slow speed.

What is the PRIMARY reason for this interlock?

a. pump cavitation.
b. flow control valve cavitation.
c. excessive axial thrust on the pump.
d. inaccurate wide range level indication.

Answer: b

Reference: LOTM-7-5, page 12 of 38, section V.A.1
Recirculation System Enabling Objective 3.3
K/A 202002K108, (3.1/3.2)

Comment:

Both (a) and (b) are correct. LOTM-7-5 page 12 of 38, section V.A.1 does give the key's answer but does not mention anything about being the PRIMARY reason for the interlock. There is some conflicting information that supports answer (a). LOTM 34-6 (Feedwater Level Control System), page 5 of 16, B.3 specifically states that the low total feedwater flow limit is based on preventing cavitation of the Reactor Recirculation pumps due to inadequate subcooling. HLO-005-05 (lesson plan), page 10 of 27, 4.1.4 does not give a PRIMARY component. Also HLO-060-5 (lesson plan), page 9 of 26 "NOTE" reinforces answer (a) to the students.

Recommendation:

Accept both answer (a) and (b).

- Reactor Water Cleanup (Chapter 14),
- Floor and Equipment Drains (Chapter 49), and
- Nuclear Boiler Instrumentation (Chapter 3).

V. <u>SYSTEM OPERATION</u>

A. Normal Operation

1. Cavitation Interlocks

   a. Cavitation is prevented by inhibiting high speed operation with total feed water flow less than 25%.

      - If pumps are operating at high speed and feed flow drops below 25%, pumps auto transfer to low speed after a 15-second time delay.
      - Starting or transferring to high speed is inhibited.
      - White light illuminates above the cavitation reset pushbutton on P680 panel, indicating a "seal-in" condition.
      - This interlock downshifts the pumps because feed flow of less than 25 percent will not provide adequate NPSH to prevent cavitation of the FCVs during fast speed recirc pump operation.
      - This interlock may be bypassed by taking SW-127A(B), Low Total Flow Interlock Bypass switch, located on panel H22-B33-P001A(B) to BYPASS.

   b. If reactor level falls to level 3 (9.7"), auto transfer to slow speed is initiated. This also allows for more accurate wide range level reading. The above interlocks will downshift both pumps to slow speed.

   c. If both loops receive a main steamline temperature/recirc pump suction differential temperature signal of <8°F, auto transfer to slow speed will be initiated after a 15-second time delay.

differential pressure transmitter (1C33-FT-N003A,B,C,D) is sent to the Main Control Room where it passes through an SRU to be converted from current to voltage. The following relationship exists between ΔP and flow: Flow = constant x $\sqrt{\Delta P}$. Since the resultant flow signal is not linear, the signal then passes through a square root extractor (Figure 1). The square root extractor linearizes the signal such that flow indications are linear and the control system responds in a linear fashion.

When approximately 40% rated steam flow is sensed in all 4 steam lines, a close signal is sent to B21-F033 (steam line inboard drain) and B21-F069 (steam line outboard drain).

Each of the four flow signals is then sent to a summing circuit whose output is a total steam flow signal. In addition, the four steam line flows are displayed on meters on P680 (pounds/hr x 10^6).

3. Feedwater Flow (Figure 4)

Feedwater flow is measured by a venturi type flow element (1C33-FEN001A,B) located in each of the two feedwater lines to the reactor vessel. Each venturi differential pressure signal is measured by a transmitter (1C33-FT-N002A,B). Similar to steam flow, the feedwater flow signals pass through an SRU and a square root extractor (Figure 1). The output signal from the square root extractor is sent to a summer where the two feedwater flow signals are summed to give the total feedwater flow. The two feedwater flows are also displayed on meters on P680 (figure 10).

→ Low total feedwater flow ($\leq$25% rated for 15 seconds) provides a signal to transfer the Reactor Recirculation pumps from fast to slow speed during downpower. This limit is based on preventing cavitation of the Reactor Recirculation pumps due to inadequate subcooling. It also prevents upshift from slow to fast speed during power ascension if this permissive is not met.

C. Component Description

1. Feed Flow/Steam Flow Summer(Figure 1 - K602)

The total feed water flow and steam flow signals are input into the feed flow/steam flow summer.

4.0 Description of controls and interlocks

OBJ #4a. 4.1 Pump Cavitation Interlocks

CB **NOTE:** List the cavitation interlocks on the board as each is discussed.

4.1.1 Cavitation is prevented by inhibiting high speed operation with total feed water flow less than 25%.

- If pumps are operating at high speed and feed flow drops below 25%, pumps auto transfer to low speed after a 15-second time delay.
- Starting or transferring to high speed is inhibited.
- White light illuminates above the cavitation reset pushbutton on P680 panel, indicating a "seal-in" condition.
- This interlock downshifts the pumps because feed flow of less than 25 percent will not provide adequate NPSH in the downcomer for fast speed recirc pump operation.
- This interlock may be bypassed by taking SW-127A(B), located on panel H22-B33-P001A (B) to BYPASS.

4.1.2 If main steamline temperature/recirc pump suction differential temperature falls to 8°F, auto transfer to slow speed will be initiated after a 15-second time delay.

- This interlock is provided because the low pressure area at the recirc pump suction might fall below saturation pressure of the coolant under such a low differential temperature condition
- This interlock may be bypassed by taking SW-125A(B located on panel H22-B33-P001A(B) to BYPASS.

4.1.3 If reactor level falls to level 3 (9.7") auto transfer to slow speed is initiated.

4.1.4 All of the above interlocks are specifically designed to prevent or limit cavitation at the:

- Jet pumps
- Recirculation pumps
- Flow Control Valves

linearize the signal.

TP-06 3.0.4 Two meters on P680-3B indicate feedwater flow is the two lines with a scale of pounds/hr x 10^6.

3.0.5 The two feedwater flow signals are input to a total feed flow summer to provide a total feedwater flow signal. The output of this summer is sent to:

- Total feedwater flow indication (steam flow/feed flow recorder P680-3B).
- Feed flow/steam flow summer
- Bailey alarm cards K618A/B send a low total feedwater flow signal to the Recirculation System. This signal will transfer recirculation pumps to LFMG during a downpower. It also prevents upshift from slow to fast speed during power ascension (if this permissive is not met).

NOTE: Ask students what the purpose of the downshift is and what the setpoint is? **ANS**: Purpose is to prevent pump cavitation due to inadequate subcooling (≤25% rated flow for 15 seconds).

4.0 Feedwater Level Control System Components

OBJ#3.1 4.1 **Feed Flow/Steam Flow Summer**

TP-04 4.1.1 The total feedwater flow and steam flow signals are input into the feed flow/steam flow summer.

4.1.2 Its purpose is to provide a method for the Feedwater Level Control System to anticipate Rx level changes and take appropriate corrective action.

4.1.3 The principle is simple. If feed flow matches steam flow, level should be relatively constant. If a match does not exist, a change in level can be expected.

4.1.4 The feedwater flow input is negative

2) Question #47 RO/#46 SRO

Which of the following will prevent RCIC discharge to the CST through the test line return valves (F022, F029)?

a. The CST suction valve is open (F010)
b. RCIC minimum flow valve open (F019)
c. CST level is less than 6.5 inches.
d. Reactor vessel level is 55 inches.

Answer: d

Reference: LOTM-20-4, RCIC, page 3, II.A.2 and table #1
RCIC objective 4 and 5.j
K/A 217000A301, (3.5/3.5)

Comment:

Answer (c) could also be correct. Operators know that RCIC pump suction is normally lined up to the CST as long as there is at least about 3 and a half feet in the CST. This lineup automatically shifts the suction path from the CST to the suppression pool when there is less than 3'5" in the CST which correlates to a trip instrument "0" on the back panels. When the swap of the suction valves takes place, the test return valves are interlocked with the suppression pool suction valve to go shut and will prevent RCIC discharge to the CST. Since 6.5" is less than 3'5" this would be a correct answer. The confusion is that the response choice is not clear if this is actual tank level or a trip instrument level. To get actual CST level, the operator calls the auxiliary control room which gives actual CST tank level in feet. The operator can also call up a computer point which is also in units of feet (actual tank level). See LOTM-20-4, page 3, II.A.2. Also see table 2 (LOTM-20-4, page 15 of 24). Also see Tech. Spec. table 3.3.3-2 note ** page 3/4 3-39 for tank level correlation. Also see Tech. Spec. table 3.3.5-2, page 3/4 3-57 for trip setpoint.

Recommendation:

Accept both answer (c) and (d)

C. General Description

The RCIC system is started automatically upon receipt of a low reactor water level signal (level 2) or manually by the operator. Water from the CST or suppression pool is pumped into the core by a turbine-driven pump powered by reactor steam.

D. Basic System Flow Path

The RCIC pump suction is normally lined up to the Condensate Storage Tank (CST). This provides an adequate supply of high purity water for system operation. Steam to operate the turbine is supplied via piping from Main Steam Line (MSL) A upstream of the inboard Main Steam Isolation Valve (MSIV). The RCIC pump discharges to the reactor vessel upper head spray nozzle. A backup source of water for the RCIC pump is available from the suppression pool. Shifting to this source of water under normal conditions requires deliberate operation of valves by the operator.

II. SYSTEM DETAILS

A. Detailed Flow Path (Figure 1)

1. Steam Flow

Steam for operation of the RCIC turbine is provided from MSL A inside the drywell upstream of the inboard MSIV (B21-F022A). The steam piping size is 8" up to the point that taps off to the RHR system, and then reduces to 4" to supply the RCIC turbine.

The steam piping to the turbine is kept hot to allow rapid starting of the turbine, therefore allowing rated RCIC system flow to be attained in less than 30 seconds when needed. This is accomplished by a normal valve lineup and piping and steam trap arrangement which maintains the steam piping at near normal operating temperature.

The turbine is designed for immediate starting with no warmup prior to operation at rated speed. Turbine exhaust steam is directed to the suppression pool for condensation via 12" piping.

A gland seal air system is provided to prevent steam leakage from the turbine glands, governor and throttle valves. Air is provided by a compressor to counteract steam leakage from the above points.

2. Water Flow

The RCIC pump suction is normally lined up to the CST. Suction automatically shifts to the suppression pool when:

- a low level exists in the CST (0"), or
- a high level exists in the Suppression Pool (+6.5") with a RCIC isolation signal not present.

<u>NOTE</u>: Tech Specs require RCIC suction to shift at ≤6.5", or 20'-3.5" actual Suppression Pool level. Current setpoint is 19'-10.9", which satisfies this requirement.

Table 2 (continued)

1H13*P601/21A ANNUNCIATORS

DIV II RCIC ISOL STM SPLY PRESS LOW	60 psig (3 sec TD)	RCIC System Auto Isolation: • RCIC turbine trip. • RCIC and RHR steam supply inboard isol. valve (1E51*F063) closes. • RCIC steam line warmup isol. valve (1E51*F076) closes. • RCIC pump min flow to suppression pool valve (1E51*F019) closes. • RCIC injection isol. valve (F013) closes.
RCIC WARMUP LINE ISO VLV E51-F076 NOT FULLY CLOSED	Valve not closed	None.
RCIC TURBINE STEAM SPLY WATER DRAIN TRAP LVL HI	0" Increasing (mid-range on trap)	RCIC steam supply drain trap bypass valve (1E51*F054) opens.
RCIC TURB TRIP PMP SUCT PRESS LOW	20" Hg vac Decr. (0.5 second TD)	RCIC Turbine Trip.
RCIC SUCT XFER CST LEVEL LOW	0" on meter	• RCIC pump suppression pool suction valve (1E51*F031) opens. • RCIC pump CST suction valve (1E51*F010) closes. • RCIC test bypass valve to CST (1E51*F022) closes. • RCIC test return valve to CST (1E51*F059) closes.
RCIC ISOLATION RCIC RM HI AMB OR VENT DIFF TEMP	Ambient 182°F Vent Diff 96°F ΔT	RCIC System Auto Isolation: • RCIC turbine trips. • RCIC steam supply valves (1E51*F063 and 1E51*F064) close. • RCIC inject isol valve (1E51*F013) closes. • RCIC pump suppression pool suction valve (1E51*F031) closes. • RCIC min flow valve to suppression pool (1E51*F019) closes. • RCIC warmup line shutoff valve (1E51*F076) closes.

RECEIVED
JUN 03 1994
SDC

TABLE 3.3.3-2 (Continued)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
D. LOSS OF POWER (continued)		
2. Division III		
a. 4.16 kv Standby Bus Undervoltage (Sustained Undervoltage)	a. 4.16 kv Basis - 3045 ± 153 volts	3045 ± 214 volts
	b. 3 ± 0.3 sec. time delay	3 ± 0.33 sec. time delay
b. 4.16 kv Standby Bus Undervoltage (Degraded Voltage)	a. 4.16 kv Basis - 3777 ± 30 volts	3777 ± 75 volts
	b. 60 ± 6 sec. time delay (w/o LOCA)	60 ± 6.6 sec. time delay
	c. 3 ± 0.3 sec. time delay (w/LOCA)	3 ± 0.33 sec. time delay

*See Bases Figure B 3/4 3-1.

**(Bottom of CST is at EL 95'1".) The levels are measured from the instrument zero level of EL 98'6".

#(Bottom of suppression pool is at EL 70'.) The levels are measured from the instrument zero level of EL 89'9".

##These are inverse time delay voltage relays or instantaneous voltage relays with a time delay. The voltages shown are the maximum that will not result in a trip. Lower voltage conditions will result in decreased trip times.

TABLE 3.3.5-2

REACTOR CORE ISOLATION COOLING SYSTEM ACTUATION INSTRUMENTATION SETPOINTS

FUNCTIONAL UNITS	TRIP SETPOINT	ALLOWABLE VALUE
1. Reactor Vessel Water Level - Low Low Level 2	$\geq$ -43 inches*	$\geq$ -47 inches
2. Reactor Vessel Water Level - High Level 8	$\leq$ 51 inches*	$\leq$ 52 inches
3. Condensate Storage Tank Level - Low	$\geq$ 0 inches	$\geq$ -4.5 inches
4. Suppression Pool Water Level - High	$\leq$ 6.5 inches	$\leq$ 8 inches
5. Manual Initiation	NA	NA

*See Bases Figure B 3/4 3-1.

3) Question #52 RO/#49 SRO

While in a refueling outage, which of the following requires Primary Containment Integrity to be established?

a. Both trains of Standby Gas Treatment becomes inoperable.
b. One LPRM detector will be replaced with a new one.
c. All source range detectors are discovered to be inoperable.
d. Irradiated fuel is to be moved in the fuel pool.

Answer: b

Reference: TS 3.6.1.2, page 6-2 amendment 35 and definition 1.7 Core Alterations
Primary containment objective 9A and 9B
K/A 223001G011, (3.3/4.2)

Comment:

No correct answer. Answer (b) assumes that replacing an LPRM is a core alteration. For our plant (BWR/6), the LPRM resides in a dry tube and if removed during a refuel outage, would have only a negligible (if any) effect on core reactivity. Replacing an LPRM is not a core alteration. See attached NRC safety evaluation of T.S.amendment No. 29, dated October 12,1988.

Recommendation:

Throw out the question since there is no correct answer.

Attached to: RBC-37672

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 29 TO FACILITY OPERATING LICENSE NO. NPF-47

GULF STATES UTILITIES COMPANY

RIVER BEND STATION, UNIT 1

DOCKET NO. 50-458

1.0 INTRODUCTION

By letter dated August 5, 1988, Gulf States Utilities Company (GSU) (the licensee) requested an amendment to Facility Operating License No. NPF-47 for the River Bend Station, Unit 1. The proposed amendment would modify the Technical Specifications (TSs) to revise the definition of core alteration to exclude the normal movement (including replacement) of local power range monitors (LPRMs) from this definition.

2.0 EVALUATION

Technical Specification Definition 1.7, CORE ALTERATION, currently does not consider normal movement of the source range monitors, intermediate range monitors, traversing in-core probes, or special moveable detectors to be considered a core alteration. This change request would provide the same exclusion for LPRMs.

River Bend Station is a BWR/6 boiling water reactor which incorporates certain design changes compared to earlier boiling water reactors. One of these changes is the introduction of a dry tube that houses the LPRM strings. The dry tubes extend from the bottom of the reactor pressure vessel vertically to the top of the core. Thus, removal and installation of the LPRMs from underneath the reactor pressure vessel can be accomplished without the removal of the reactor vessel head and fuel does not need to be moved from around the dry tube for maintenance or replacement of LPRMs. The LPRM strings are only removed from the core when they are being replaced and they have no normal drive mechanisms. Based on the above discussion, the staff concludes that the exclusion of the LPRMs in the definition of core alteration is acceptable.

With the modification of the definition of core alteration discussed above, the footnote excepting replacement of LPRM strings applicable to Action 3 and Action 9 of Table 3.3.1-1 is no longer necessary. The staff concludes that deletion of the footnote is acceptable.

3.0 ENVIRONMENTAL CONSIDERATION

The amendment involves a change in the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposures. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration and there has been no public comment on such finding. Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR Section 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

4.0 CONCLUSION

The staff has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulations, and the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Date: October 12, 1988

Principal Contributor: W. Paulson

4) Question #75/#71 SRO

With the reactor initially at 100% power, a loss of instrument air to which of the following will NOT eventually result in an automatic REACTOR PROTECTION SYSTEM SCRAM?

a. Condensate and heater drain pump recirc valves.
b. Feedwater regulating valves.
c. Turbine steam seals and SJAE.
d. Scram inlet and outlet valves.

Answer: d

Reference: AOP-0008, R7A, section 4.0
AOP-0008 objective 3.a
K/A 295019K201, (3.8/3.9)

Comment:

The words "will not eventually" leaves this question with no correct answer. Answer (a) does not send a direct RPS trip but will eventually cause a feedpump trip and eventually a RPV level 3 scram (see AOP-0008, Rev.7A page 3 of 16, section 3.2). Answer (b) will cause the feedwater regulating valves to lock up and fail "as is". Changes in power, pressure, temperature, and other non-constant external and internal forces (even fuel depletion) will cause a steam flow to feed flow mismatch and you'll get either a high or low RPV level scram. Also the air pressure that is locked up, keeping the feedwater regulating valves in a steady, constant position, will eventually bleed slowly. Answer (c) will result in a loss of condenser vacuum which does not send a direct RPS trip but will eventually cause a turbine trip and a resultant Rx scram (see AOP-0008, Rev. 7A page 3 of 16, section 3.5). Answer (d) will cause the rods to go in, and will cause either the scram discharge volume to fill faster than it can drain (assuming that the scram discharge volume vent & drain valves did not fail closed since the air line is common to the scram inlet and outlet valves which lost air) and cause a RPS scram or because the turbine is on the line and Rx power level is going down due to rods going in will cause Rx pressure to go below 849 psig (eventually because of other steam loads) and cause the MSIV's to go closed and cause a RPS scram (see LOTM-5 Fig. 12, Tech.Spec table 2.2.1-1(6)&(9), pages 2-4&5, and Tech.Spec.table 3.3.2-2(2.c), page 3/4 3-19).

Recommendation:

Throw out the question since there is no correct answer.

1.0 PURPOSE/DISCUSSION

1.1 The purpose of this procedure is to provide guidance to the operators in the event that Instrument Air System air pressure is lowering or lost.

1.2 A total loss of instrument air pressure may be caused by a break in the instrument air header, or by a loss of all air compressors. A multitude of actions occur as a result of low instrument air supply pressure. These actions occur at various times depending upon the rate of the instrument air pressure drop. The actions listed in 3.0, Automatic Actions, are listed in decreasing order of significance to the Nuclear Steam Supply System.

2.0 SYMPTOMS

2.1 Lowering Instrument Air Header Pressure.

2.2 Amber indicating lights for compressor 1IAS-C1A, C1B and/or C1C.

2.3 Various AOV's will fail (see Attachment 1) in a random manner.

3.0 AUTOMATIC ACTIONS

→ 3.1 Control rods individually scram as the scram valves fail open. The Scram Discharge Volume vent and drain valves fail closed. CRD flow control valves fail closed.

→ 3.2 Condensate and heater drain pumps recirc valves will open and "starve" the reactor feed pumps; causing them to trip

→ 3.3 The Feedwater Reg Valves will lock up and fail "as is" on low air pressure (85 psig).

3.4 All normal HVAC will fail due to closure of AOD's.

→ 3.5 Loss of steam seals and SJAE will result in a loss of condenser vacuum.

3.6 Drywell and containment drains will isolate.

4.0 IMMEDIATE OPERATOR ACTIONS

4.1 Make a plant wide Gaitronics announcement to cease non-essential use of air.

4.2 If any of the following occurs, Scram the Reactor and enter AOP-0001 REACTOR SCRAM:

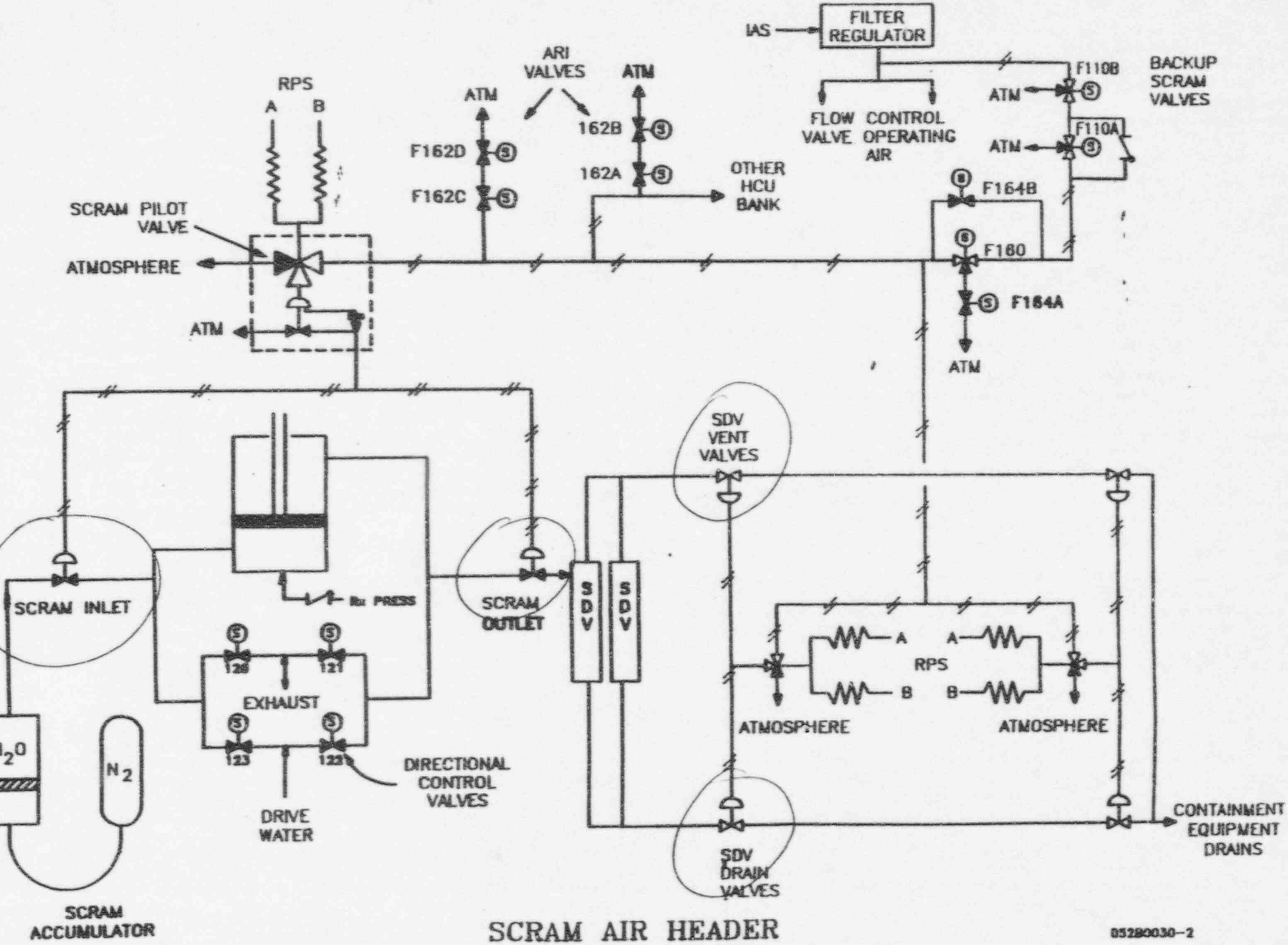
4.2.1 When individual rod movement is observed.

4.2.2 When the instrument air header pressure decreases to 65 psig (1IAS-PI105) on 1H13*P870.

5.0 SUBSEQUENT OPERATOR ACTIONS

5.1 Monitor air header pressure and if it lowers to 50 psig, verify closed or close the MSIV's.

LOTM-5



SCRAM AIR HEADER

Figure 12

TABLE 2.2.1-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. Intermediate Range Monitor, Neutron Flux-High	$\leq$ 120/125 divisions of full scale	$\leq$ 122/125 divisions of full scale
2. Average Power Range Monitor:		
a. Neutron Flux-High, Setdown	$\leq$ 15% of RATED THERMAL POWER	$\leq$ 20% of RATED THERMAL POWER
b. Flow Biased Simulated Thermal Power-High		
1) Two Recirculation Loop Operation		
a) Flow Biased	$\leq$ 0.66 W+48%, with a maximum of	$\leq$ 0.66 W+51%, with a maximum of
b) High Flow Clamped	$\leq$ 111.0% of RATED THERMAL POWER	$\leq$ 113.0% of RATED THERMAL POWER
2) Single Recirculation Loop Operation		
a) Flow Biased	< 0.66 W+42.7%, with a maximum of	< 0.66 W+45.7%, with a maximum of
b) High Flow Clamped	< 111.0% of RATED THERMAL POWER	< 113.0% of RATED THERMAL POWER
c. Neutron Flux-High	$\leq$ 118% of RATED THERMAL POWER	$\leq$ 120% of RATED THERMAL POWER
d. Inoperative	NA	NA
3. Reactor Vessel Steam Dome Pressure - High	$\leq$ 1064.7 psig	$\leq$ 1079.7 psig
4. Reactor Vessel Water Level - Low, Level 3	$\geq$ 9.7 inches above instrument zero*	$\geq$ 8.7 inches above instrument zero
5. Reactor Vessel Water Level-High, Level 8	$\leq$ 51.0 inches above instrument zero*	$\leq$ 52.1 inches above instrument zero
6. Main Steam Line Isolation Valve - Closure	$\leq$ 8% closed	$\leq$ 12% closed

*See Bases Figure B 3/4 3-1.

RECEIVED
NOV 30 1988
SDC

TABLE 2.2.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
7. Main Steam Line Radiation - High	$\leq$ 3.0 x full power background	$\leq$ 3.6 x full power background
8. Drywell Pressure - High	$\leq$ 1.68 psig	$\leq$ 1.88 psig
9. Scram Discharge Volume Water Level - High		
a. Level Transmitter - LISN601A and B	$\leq$ 49"	$\leq$ 53"
LISN601C and D	$\leq$ 49"	$\leq$ 51.7"
b. Float Switches - LSN013A and B	$\leq$ 48.76"	$\leq$ 53.50"
LSN013C and D	$\leq$ 46.88"	$\leq$ 49.00"
10. Turbine Stop Valve - Closure	$\leq$ 5% closed	$\leq$ 7% closed
11. Turbine Control Valve Fast Closure, Trip Oil Pressure - Low	$\geq$ 530 psig	$\geq$ 465 psig
12. Reactor Mode Switch Shutdown Position	NA	NA
13. Manual Scram	NA	NA

*See Bases Figure B 3/4 3-1.

TABLE 3.3.2-2
ISOLATION ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
1. PRIMARY CONTAINMENT ISOLATION		
a. Reactor Vessel Water Level - Low Low, Level 2	$\geq$-43 inches*	$\geq$-47 inches
b. Drywell Pressure - High	$\leq$ 1.68 psig	$\leq$ 1.88 psig
c. Containment Purge Isolation Radiation - High	$\leq$ 1.3 R/hr	$\leq$ 1.57 R/hr
2. MAIN STEAM LINE ISOLATION		
a. Reactor Vessel Water Level - Low Low Low, Level 1	$\geq$-143 inches*	$\geq$-147 inches
b. Main Steam Line Radiation - High	$\leq$ 3.0 x full power background	$\leq$ 3.6 x full power background
c. Main Steam Line Pressure - Low	$\geq$ 849 psig	$\geq$ 837 psig
d. Main Steam Line Flow - High		
1. Line A	$\leq$ 146 psid	$\leq$ 151 psid
2. Line B	$\leq$ 156 psid	$\leq$ 161 psid
3. Line C	$\leq$ 153 psid	$\leq$ 158 psid
4. Line D	$\leq$ 164 psid	$\leq$ 169 psid
e. Condenser Vacuum - Low	$\geq$ 8.5 inches Hg. vacuum	$\geq$ 7.6 inches Hg. vacuum
f. Main Steam Line Tunnel Temperature - High	$\leq$ 141°F	$\leq$ 148.5 °F
g. Main Steam Line Tunnel Δ Temperature - High	$\leq$ 57°F	$\leq$ 61 °F

RECEIVED
JUN 0 8 1988
CIRC