2001 SSES NRC Written Examination – August 10, 2001

Facility Summary

Prior to Written Examination Administration

Prior to administration of the written examination to the candidates, a change to the answer key was made. The question is included in Enclosure A. Typographical errors were also identified and corrected but are not included in this report.

- Developers identified one answer key change and corrected the question and key; the NRC lead examiner was notified.
- Developers identified typographical errors and corrected them.

Administered Examination to the Candidates

During the administration of the examination the proctors made minor clarifications to nine questions in response to questions asked by the candidates. Refer to Enclosure B for the "Questions/Comments during Administration."

- During administration candidate questions were captured and nine clarifications were made to the examination. These clarifications were minor. Refer to questions RO2/SRO2, RO3/SRO3, RO30/SRO27, RO38, RO41/SRO33, RO53/SRO42, RO60, RO79/SRO58, and RO86/SRO64.

Post Examination Review with Candidates / Concurrent and Independent Review

As a result of the post-examination review with the students and an independent and concurrent review by several instructors, changes were made to eight questions on the examination. Of these changes, the changes to five questions were considered significant and the changes to three questions were considered not significant. Refer to Enclosure C for "Post Examination Comments" and the resolution of these comments. The concurrent and independent review by instructors identified question comments similar to those identified by the candidates and also identified some questions comments not identified by the candidates.

- Three questions had changes that were identified as not significant because although the questions had the wrong answer this was identified and reflected on the answer key before grading the examination. This required that the correct answer be changed but did not affect the question and answers for each of these questions. Refer to questions RO23/SRO21, RO54/SRO43, and RO50. The comments that resulted in these changes were identified by the candidates in two instances and were identified by the instructors for all three questions.
- One question had a change considered significant change because there was no correct answer and the question was deleted. Refer to question RO92/SRO68. The comments that resulted in this question change were not identified during the review with the candidates but was identified by the instructors. It is not known if there is a knowledge deficiency in this area but this will be further evaluated to determine if there is one.
- Four questions had changes considered significant because there were two correct answers for each of these four questions. Refer to questions RO95/SRO70, RO33/SRO29, RO67, and RO55/SRO44. The comments that resulted in these changes were identified by both the candidates during the post-examination review and by the instructors during the concurrent and independent examination review.

2001 SSES NRC Written Examination – August 10, 2001

Examination Analysis Results:
The examination analysis confirmed the changes previously identified and showed no additional examination changes that warranted review. The examination analysis also revealed three topics where a knowledge deficiency may be present. RO15/SRO13 revealed a possible deficiency in limiting safety system settings and the bases for the settings for all candidates. RO88/SRO66 revealed a possible deficiency in required on-shift time to maintain an active license for the reactor operator candidates. RO66/SRO51 revealed a possible deficiency in RWM operation for the reactor operator candidates.

WRITTEN EXAM

CHANGES MADE DURING

PROTORING

WRITTEN EXAM Changes made during PROTORING

SSES 2001 NRC License Written Examination
Questions / Comments During Administration

Applicant Level: ☑ RO ☐ SRO

Applicant Name

Question Type: ☑ Common ☐ RO only ☐ SRO only

Question #: RO 2 (enter number, if SRO only enter N/A)
SRO 2 (enter number, if RO only enter N/A)

Question (enter verbatim):

We are talking about Main turbine Lube Oil here?

Response:

☐ Withheld additional guidance. Instructed the applicant to do his best with information provided.
☑ Did NOT withhold additional guidance. Provided the following response (enter verbatim).

Yes, MAIN TURBINE LUBE OIL System

Changes / clarifications made to examination:

☑ Identified on the white board
☑ Called to the attention of the applicants

Proctor:

☐ Post-Examination Comments form initiated

Edwin Bowles
~~Print~~ Name

Edwin W Bowles 8/10/2001
Signature Date

RO 2 | SRO 2

QUESTION 2.

Unit 1 is operating at 35% power when a loss of MAIN turbine lubricating oil system occurs. Which one of the following RPV pressure responses would occur over the **first five (5) minutes** of this event? **Assume all system operate as designed.**

RPV pressure will...

a. be controlled at approximately 955 psig.

b. be initially controlled at 1005 psig then lower to 915 psig.

c. rise to 1116 psig then lower and be maintained at 1106 psig.

d. initially rise to 1106 psig then cycle between 1070 psig and 1106 psig.

SSES 2001 NRC License Written Examination
Questions / Comments During Administration

Applicant Level: ☑ RO ☐ SRO

Applicant Name:

Question Type: ☑ Common ☐ RO only ☐ SRO only

Question #: RO 3 (enter number, if SRO only enter N/A)
SRO 3 (enter number, if RO only enter N/A)

315 KEY

Question (enter verbatim):
What do you mean by throttle pressure?

Response:
☐ Withheld additional guidance. Instructed the applicant to do his best with information provided.
☑ Did NOT withhold additional guidance. Provided the following response (enter verbatim).

throttle pressure can also be called Pressure Averaging manifold pressure

Changes / clarifications made to examination:

☑ Identified on the white board
☑ Called to the attention of the applicants

Proctor:
☐ Post-Examination Comments form initiated

Edwin Bowles
Print Name
Edwin W. Bowles 8/10/2001
Signature Date

* K03 / SR03

QUESTION 3.

The Unit 1 is operating at 100% power when a failure of the in-service EHC Pressure Regulator causes the controller output to lower to zero. Which one of the following will occur?

a. The MSIVs will isolate when reactor pressure lowers to 860 psig.

b. The reactor will scram on either high APRM power or high RPV pressure.

c. Pressure Averaging Manifold ~~Throttle pressure~~ will rise about 4 psig and be controlled by the standby regulator.

d. Pressure Averaging Manifold ~~Throttle pressure~~ will lower about 4 psig and be controlled by the standby regulator.

SSES 2001 NRC License Written Examination
Questions / Comments During Administration

Applicant Level:	☐ RO	☑ SRO	
Applicant Name:			
Question Type:	☑ Common	☐ RO only	☐ SRO only

Question #: RO 30 (enter number, if SRO only enter N/A) 30/41-KEY
SRO 27 (enter number, if RO only enter N/A)

Question (enter verbatim):
Questions asks for process monitors. Answers (some) have ventilation system. What should I be looking for? Sources?

Response:
☐ Withheld additional guidance. Instructed the applicant to do his best with information provided.
☑ Did NOT withhold additional guidance. Provided the following response (enter verbatim).
Delete the Last sentance of the question stem and Replace it with: Discharges from ~~the~~ Ew? which one of the following will cause this alarm condition?

Changes / clarifications made to examination:
☑ Identified on the white board
☑ Called to the attention of the applicants

Proctor:
☐ Post-Examination Comments form initiated

Edwin W. Bowles
Print Name

8/10/2001
Signature Date

=RO 30/SRO 27

QUESTION 30.

An alarm condition exits on the Unit 1 Reactor Building Stack Monitor on 0C630. You are directed to go to panel 1C600 and determine the source of the high radiation condition. ~~Which one of the following process monitors are an input to this alarm condition?~~
change to: Discharges from which one of the following will cause this alarm condition.

a. Zone 3, Railroad Access Shaft.

b. Main Steam Line Radiation Monitors.

c. Radwaste Building Ventilation System.

d. Standby Gas Treatment Ventilation Exhaust.

SSES 2001 NRC License Written Examination
Questions / Comments During Administration

Applicant Level: ☑ RO ☐ SRO

Applicant Name:

Question Type: ☐ Common ☑ RO only ☐ SRO only

Question #: RO 38 (enter number, if SRO only enter N/A)
SRO N/A (enter number, if RO only enter N/A)

38 KEY

Question (enter verbatim):
drive water d/p 240 psig - should this be 240 psid?

Response:
☐ Withheld additional guidance. Instructed the applicant to do his best with information provided.
☑ Did NOT withhold additional guidance. Provided the following response (enter verbatim).

change 240 psig to 240 psid

Changes / clarifications made to examination:

☑ Identified on the white board
☑ Called to the attention of the applicants

Proctor:
☐ Post-Examination Comments form initiated

Edwin Bowles
Print Name

[signature] 8/10/2001
Signature Date

QUESTION 38. RO only

An ATWS has occurred on Unit 2 and reactor power is approximately 19%. The operator attempts to fully insert control rod 32-27 with the CONTINUOUS-IN switch and the rod fails to move. The operator then notes the following conditions:

• Drive Water D/P	240 ~~psig~~ psid
• Reactor Mode Switch	Shutdown
• CRD Flow	120 gpm
• CRD Flow Control Valves	Closed
• CRD pump	2A and 2B Running
• RWM	Normal

Control Rod 32-27 will not move because the...

a. RWM is enforcing an insert block.

b. Reactor Mode Switch is enforcing a rod block.

c. Drive Water D/P is much lower when rod movement is attempted.

d. CRD flow control valves are closed shutting off CRD flow to the HCUs.

SSES 2001 NRC License Written Examination
Questions / Comments During Administration

Applicant Level:	☐ RO	☑ SRO

Applicant Name:

Question Type:	☑ Common	☐ RO only	☐ SRO only

Question #: RO 41 (enter number, if SRO only enter N/A)
SRO 33 (enter number, if RO only enter N/A)

41/45 KEY

Question (enter verbatim):

436 psig is old setpoint (420± is new one)
Are below pressure permissive or not?

Response:

☐ Withheld additional guidance. Instructed the applicant to do his best with information provided.
☑ Did NOT withhold additional guidance. Provided the following response (enter verbatim).

change the pressure in the question stem from 436 psig to 410 psig.

Changes / clarifications made to examination:

☑ Identified on the white board
☑ Called to the attention of the applicants

Proctor:

☐ Post-Examination Comments form initiated

Edwin Bowles
Print Name

[signature] 8/10/2001
Signature Date

< RO 41 / SRO 33

QUESTION 41.

Unit 1 is operating at 100% power with the following conditions:

- SO-149-002, Quarterly RHR System Flow Verification, in progress.
- "A" RHR Pump discharging 12,200 gpm through HV-151-F024A, Test Return Valve.

During the test a valid LPCI initiation signal and loss of **250 VDC** Power occurs. Which one of the following is the response of the RHR system when RPV pressure lowers to ~~436~~ psig. 410 psig

	F015A, LPCI Injection Valve	F024A, Test Return Valve	F007A, Min Flow Valve	"A" RHR Pump
a.	REMAINS CLOSED	CLOSES	OPENS	TRIPS
b.	OPENS	CLOSES	OPENS	REMAINS RUNNING
c.	REMAINS CLOSED	REMAINS OPEN	REMAINS CLOSED	REMAINS RUNNING
d.	OPENS	REMAINS OPEN	REMAINS CLOSED	TRIPS

SSES 2001 NRC License Written Examination
Questions / Comments During Administration

Applicant Level: ☐ RO ☑ SRO

Applicant Name:

Question Type: ☑ Common ☐ RO only ☐ SRO only

Question #: RO 53 (enter number, if SRO only enter N/A)
SRO 42 (enter number, if RO only enter N/A)

Question (enter verbatim):

time remaining from?

Response:

☑ Withheld additional guidance. Instructed the applicant to do his best with information provided.
☑ Did NOT withhold additional guidance. Provided the following response (enter verbatim).

From time = 60 seconds

Changes / clarifications made to examination:

☑ Identified on the white board
☑ Called to the attention of the applicants

Proctor:
☐ Post-Examination Comments form initiated

Edwin Bowles
Print Name
Edwin W Bowles 8/10/2001
Signature Date

RO 53 / SRO 42

QUESTION 53.

Unit 1 was at 100% power when a LOCA occurred. The following events take place at the indicated times after the LOCA:

- Time = 2 seconds, High Drywell Pressure setpoint reached, ECCS pumps started and operate on minimum flow.
- Time = 20 seconds, RPV water level lowers to -129 inches.
- Time = 48 seconds, RPV water level recovers to -90 inches.
- Time = 60 seconds, RPV water level lowers to -129 inches.

Which one of the following is **the time remaining** before ADS initiates?

a. 42 seconds

Time remaining from T=60 secs

b. 44 seconds

c. 82 seconds

d. 102 seconds

SSES 2001 NRC License Written Examination
Questions / Comments During Administration

Applicant Level: ☑ RO ☐ SRO

Applicant Name:

Question Type: ☐ Common ☑ RO only ☐ SRO only

Question #: RO 60 (enter number, if SRO only enter N/A)
SRO N/A (enter number, if RO only enter N/A)

Question (enter verbatim):
Is the question asking the level right after the pushbutton is released or later?

Response:
☐ Withheld additional guidance. Instructed the applicant to do his best with information provided.
☑ Did NOT withhold additional guidance. Provided the following response (enter verbatim).
In the question stem delete "and holds" and delete "for 5 seconds".
In Selection "c", change the second +18 inches to +14 inches or higher.

Changes / clarifications made to examination:
☑ Identified on the white board
☑ Called to the attention of the applicants

Proctor:
☐ Post-Examination Comments form initiated
Edwin N. Bowles
Print Name
Edwin W. Bowles 8/10/2001
Signature Date

QUESTION 60. RO ONLY

The following conditions occurred on Unit 1 following a reactor scram from 95% power:

- RPV water level lowered to zero (0) inches before rising.
- Reactor Water Level Control System remains in AUTOMATIC.
- NO operator actions have been taken for the Reactor Water Level Control System.
- With RPV water level at +5 inches the PCO depresses ~~and holds~~ the SETPOINT SETDOWN reset pushbutton (HS-C32-1S08) ~~for five (5) seconds~~ and then releases it

Which one of the following describes where RPV water level will stabilize and why?

a. +5 inches because it can't be reset.

b. +13 inches because level hasn't recovered.

c. +18 inches because it can't be reset until level is ~~+18~~ +14 or higher inches.

d. +35 inches because level will reset and return to the normal setpoint.

SSES 2001 NRC License Written Examination
Questions / Comments During Administration

Applicant Level:	☑ RO	☐ SRO	
Applicant Nam			
Question Type:	☑ Common	☐ RO only	☐ SRO only
Question #:	RO 79 (enter number, if SRO only enter N/A)	SRO 58 (enter number, if RO only enter N/A)	79/76 KEY

Question (enter verbatim):

They are there but I don't know if we use them anymore. Go with location?

Response:

☐ Withheld additional guidance. Instructed the applicant to do his best with information provided.
☑ Did NOT withhold additional guidance. Provided the following response (enter verbatim).

IN the No. 1 & No. 2 Statements in the Question Stem replace the word "Monitor with Sampling.

Changes / clarifications made to examination:

☑ Identified on the white board
☑ Called to the attention of the applicants

Proctor:

☐ Post-Examination Comments form initiated

Edwin Bowles
Print Name

Edwin W. Bowles 8/10/2001
Signature Date

RO 79 / SRO 58

QUESTION 79.

For Unit 1 which one of the following are the OFFGAS PRE-TREATMENT and OFFGAS POST-TREATMENT PROCESS RADIATION MONITOR sample point locations?

sampling

(1) The Pre-Treatment ~~Monitor~~ takes a sample from...
(2) The Post-Treatment ~~Monitor~~ takes a sample from...

sampling

a. (1) The inlet to the Steam Jet Air Ejectors.
(2) The inlet to the Offgas HEPA Filter.

b. (1) The inlet to the Offgas Recombiners.
(2) The outlet from the Charcoal Adsorbers.

c. (1) The outlet from the Offgas Recombiners.
(2) The outlet from the Offgas HEPA Filter.

d. (1) The outlet from the Steam Jet Air Ejectors.
(2) The inlet to the Charcoal Adsorbers.

SSES 2001 NRC License Written Examination
Questions / Comments During Administration

Applicant Level: ☐ RO ☑ SRO

Applicant Name:

Question Type: ☑ Common ☐ RO only ☐ SRO only

Question #: RO 86 (enter number, if SRO only enter N/A)
SRO 64 (enter number, if RO only enter N/A)

Question (enter verbatim):
I don't think the ~~Hot~~ Ex 3 heading on this first column is correct.

Response:
☐ Withheld additional guidance. Instructed the applicant to do his best with information provided.
☑ Did NOT withhold additional guidance. Provided the following response (enter verbatim).
Change the heading on the first column from ZONE I HVAC FANS to Recirc FANS

Changes / clarifications made to examination:

☑ Identified on the white board
☑ Called to the attention of the applicants

Proctor:
☐ Post-Examination Comments form initiated

Edwin Bowles
Print Name
Edwin W Bowles 8/10/2001
Signature Date

RO 86 / SRO 64

QUESTION 86.

Both units are operating at 100% power when a Unit 1 Railroad Access Shaft Radiation Monitor reaches 6 mr/hr. Which one of the following is the Reactor Building HVAC response?

	~~Zone 1 HVAC Fans~~ RECIRC FANS	Zone 3 HVAC Fans	SBGT	CREOASS
a.	STOP AND ISOLATE	NO CHANGE	STARTS	NO CHANGE
b.	RECIRC INITIATES	STOP AND ISOLATE	NO CHANGE	STARTS
c.	STOP AND ISOLATE	NO CHANGE	NO CHANGE	NO CHANGE
d.	RECIRC INITIATES	STOP AND ISOLATE	STARTS	STARTS

SSES 2001 NRC License Written Examination
Questions / Comments During Administration

Applicant Level: ☐ RO ☑ SRO

Applicant Name:

Question Type: ☐ Common ☐ RO only ☑ SRO only

Question #: RO N/A (enter number, if SRO only enter N/A) SRO 19 KEY
SRO 78 (enter number, if RO only enter N/A)

Question (enter verbatim):
Am I beyond anticipating rapid depress?
Do I know this?

Response:
☑ Withheld additional guidance. Instructed the applicant to do his best with information provided.
☐ Did NOT withhold additional guidance. Provided the following response (enter verbatim).

Changes / clarifications made to examination:

☐ Identified on the white board NO
☐ Called to the attention of the applicants NO

Proctor:
☐ Post-Examination Comments form initiated

Edwin Bowles
Print Name
Edwin W Bowles 8/10/2001
Signature Date

WRITTEN EXAM

FACILITY POST EXAM COMMENTS

Post Exam Comments - WRITTEN

#1

SSES 2001 NRC Initial Written Examination
Post-Examination Comments

Applicant Level: ☑ RO ☑ SRO

Applicant Name:

Question Type: ☑ Common ☐ RO only ☐ SRO only

Question #: RO 15 (enter number, if SRO only enter N/A)
SRO 13 (enter number, if RO only enter N/A)

Answer: (circle the answer key response) A B C D

Reference: (enter the answer key reference below)
T.S. BASES 3.3.1

Comment: (enter the comment below)
Re-evaluate bases of answer.
Turb. Trip >30% anticipates Pressure/power transient

Recommendation: (The grader is encouraged to discuss the matter with the NRC Chief Examiner before proceeding with the grading)
☐ Change the correct answer. ☑ Do NOT change the correct answer.
☐ Accept two correct answers. ☐ Delete the question
☑ Make clarifications to the question.

Changes / clarifications made to examination: (provide a description)

Reference(s) to support change / clarification made to examination:

Justification for rejection of an applicant's comment:
Pressure/Power Transient Caused by Loss of Heat Sink.

Proctor: ☐ Change made in INK on the master examination copy
Edwin W. Bowles
Print Name
Edwin W Bowles 8/10/2001 [signature] 08/14/01
Signature Date

OPERATIONS QUESTION AND ANSWER INPUT FORM

RO 15 SRO ~~28~~ 13

(A) SY017 L-5 (B) 20
Course Objective

(C) Question Type (check one)
[X] Multiple Choice
[ ] Matching
[ ] Free Format (Essay)

(D) Bank
Operations [X]
OP002 [ ]

(E)	1	2	3	4	5	6	7	8
Keywords :	Category	Topic 1	Topic 2	JTA	Setting	Other Objs.	Quiz Only	Retired
(≤9 characters)	E/APE	Loss of Vacuum						

(F) Point Value: 1 (G) Answer Time: (Minutes)

(I) Review Date (YYMM):

(H) Cognitive Level: (Check one)
[X] 1 Memory
[ ] 2 Comprehension
[ ] 3 Application
[ ] 4 Analysis
[ ] 5 Problem Solving

(J) QUESTION:

Which one of the following is the bases for a reactor scram on a main turbine trip above 30% reactor power?

a. Provides a backup to the RPV pressure and APRM high scrams.

b. Ensures RPV water level remains above the dryer separator skirt.

c. Protects the reactor from the pressure effects of a loss of heat sink.

d. Anticipates a positive reactivity addition from a loss of feedwater heating.

(K) ANSWER: c. ok

(L) REQUIRED MATERIALS: None

(M) K&A NUMBER/RATING: 295002, AK1.03/ 3.6

(N) NOTES:

JUSTIFICATION:	Scramming the reactor above 30% power prevents pressure transients caused by loss of the main condenser as the heat sink because turbine bypass capacity is limited. (Turbine trips on low vacuum, 21.7" Hg Vac, before MSIVs close or any other auto actions).			
DISTRACTER A:	These scrams backup the turbine trip scram.			
DISTRACTER B:	This is part of the bases for the low level scram.			
DISTRACTER D:	This is a concern at all powers and is not the bases for this scram			
EXAM OUTLINE CROSS-REF:	LEVEL:	RO	SRO	
	TIER:	1	1	
	GROUP:	1	2	
K/A TEXT:	AK1.03 – Knowledge of the operational implications of the following concepts as they apply to LOSS OF CONDENSER VACUUM: loss of heat sink.			
QUESTION SOURCE:	BANK:			
	MODIFIED:			
	NEW:	X		
10CFR55:	41(b).5, 43(b).2			
COMMENTS:				

(O) REFERENCES: Technical Specifications Bases B 3.3.1.1.8

(P) POSITIONS: (check one or more boxes)	R - RO	S - SRO	A - ASO	N - NPO	T - STA
	X	X			

(Q) Prepared by ED BOWLES

(R) Reviewed by: R. E. C[illegible]

2

SSES 2001 NRC Initial Written Examination
Post-Examination Comments

Applicant Level: ☒ RO ☐ SRO

Applicant Name:

Question Type: ☒ Common ☐ RO only ☐ SRO only

Question #: RO 23 (enter number, if SRO only enter N/A)
SRO 21 (enter number, if RO only enter N/A)

Answer: (circle the answer key response) A B C D

Reference: (enter the answer key reference below)

Comment: (enter the comment below) *
"A" is also viable choice because all D/G's are running loaded. "C" could also be correct because U2 breakers are not addressed.

Recommendation: (The grader is encouraged to discuss the matter with the NRC Chief Examiner before proceeding with the grading)
☑ Change the correct answer. ☐ Do NOT change the correct answer.
☐ Accept two correct answers. ☐ Delete the question
☑ Make clarifications to the question.

Changes / clarifications made to examination: (provide a description)
Changed to make a. the correct answer, c cannot also be correct because Unit 2 breakers must be considered

Reference(s) to support change / clarification made to examination:
EO-100-030

Justification for rejection of an applicant's comment:

Proctor: ☒ Change made in INK on the master examination copy
Edwin W Bowles (master typed exam updated prior to grading)
Print Name
[signature] 8/10/2001
Signature Date

[signature] 8/14/01

* this comment was anticipated based on a review conducted by an instructor during exam administration.

OPERATIONS QUESTION AND ANSWER INPUT FORM

RO 23 SRO ~~33~~ 21

(A) SY017 M-1 (B)
Course Objective

(C) Question Type (check one)
[X] Multiple Choice
[ ] Matching
[ ] Free Format (Essay)

(D) **Bank**
Operations [X]
OP002 [ ]

(E)	1	2	3	4	5	6	7	8
Keywords :	Category	Topic 1	Topic 2	JTA	Setting	Other Objs.	Quiz Only	Retired
(≤9 characters)	OPS	ONXXXXXX						

(F) Point Value: 1 **(G)** Answer Time: (Minutes)

(I) Review Date (YYMM):

(H) Cognitive Level: (Check one)
[ ] 1 Memory
[ ] 2 Comprehension
[X] 3 Application
[ ] 4 Analysis
[ ] 5 Problem Solving

(J) QUESTION:

Unit 2 is shutdown with the following conditions:

- "2A" RBCCW and "2A" TBCCW are aligned to ESW.
- Loop "A" of ESW is isolated from the Diesel Generators (DGs).

A loss of off-site power occurs

- DG output breaker 1A20404 fails to close.
- "B" ESW Pump fails to start

Assuming NO operator actions, which one of the following is required?

a. Trip ALL the DGs in four and one half (4.5) minutes.

b. Trip DG "B" and DG "D" in four and one half (4.5) minutes

c. Trip DGs "A", "B" and "C" in four and one half (4.5) minutes and DG "D" in eight (8) minutes.

d. Trip DGs "A" and "C" in four and one half (4.5) minutes and DGs "B" and "D" in eight (8) minutes.

(K) ANSWER: ~~c.~~ a.

All four DGs are Loaded! - the DG "D" Breaker for Unit 2 closes 2A20404 - so that DG is Loaded! - A is the correct answer

R.E. Clin

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Rev. 3, (8/95)
Page 1 of 1

E. W. Bowler

File No. R11-2

OPERATIONS QUESTION AND ANSWER INPUT FORM

2A20464 closes on the loop

(L) REQUIRED MATERIALS: None

(M) K&A NUMBER/RATING: 295018, AK3.01/3.5

(N) NOTES:

JUSTIFICATION:	On a loss of cooling water the diesels ~~must~~ be tripped in 4.5 minutes if loaded and 8 minutes if unloaded. DG D does ~~not~~ load (its output breaker does not close. So the all the DGs are without cooling water, but D is unloaded. So A,B,C are tripped in 4.5 min. and D must be tripped in 8 min.		
DISTRACTER ~~A~~: B	DG ~~D is running~~ unloaded. All DGs are cooled and must be tripped		
DISTRACTER ~~B~~: C	A and ~~C must be~~ tripped and D may run 8 minutes. " " " " " "		
DISTRACTER ~~D~~: D	B must be ~~tripped in 4.5~~ minutes. " " " " " "		
EXAM OUTLINE CROSS-REF:	LEVEL:	RO	SRO
	TIER:	1	1
	GROUP:	2	2
K/A TEXT:	AK1.01 – Knowledge of the operational implications of the following concepts as they apply to PARTIAL OR COMPLETE LOSS OF CCW: Effects on component/system operation.		
QUESTION SOURCE:	BANK:		
	MODIFIED:		
	NEW:	X	
10CFR55:	41(b).7, 41(b).10, 43(b).5		
COMMENTS:			

(O) REFERENCES: EO-100-030, Caution on pages 2

(P) POSITIONS: (check one or more boxes)	R - RO	S - SRO	A - ASO	N - NPO	T - STA
	X	X			

(Q) Prepared by ED BOWLES

(R) Reviewed by: R. E. C[illegible]

PROCEDURE COVER SHEET

	NUCLEAR DEPARTMENT PROCEDURE	
SUSQUEHANNA STEAM ELECTRIC STATION PP&L	UNIT 1 RESPONSE TO STATION BLACKOUT	EO-100-030 Revision 17 Page 1 of 12

QUALITY CLASSIFICATION: (X) QA Program ( ) Non-QA Program	APPROVAL CLASSIFICATION: (X) Plant ( ) Non-Plant ( ) Instruction

EFFECTIVE DATE: 11/02/00

PERIODIC REVIEW FREQUENCY: 2 Year

PERIODIC REVIEW DUE DATE: 12/30/02

RECOMMENDED REVIEWS:

Training

Procedure Owner: David M. Kapuschinsky

Responsible Supervisor: Thomas R. Markowski

Responsible FUM: Manager-Nuclear Operations

Responsible Approver: Manager-Nuclear Operations

1. SYMPTOMS AND OBSERVATIONS

This procedure is entered concurrently with other Emergency Operating Procedures whenever following conditions are met:

1.1 All offsite power supplying Unit 1 Auxiliary Busses and Unit 1 ESS Busses is lost,

AND

1.2 All four Unit 1 ESS Buses remain de-energized.

2. OPERATOR ACTIONS

		CONFIRM
2.1	PERFORM following:	
2.1.1	CLASSIFY SBO event along with other existing plant conditions in accordance with EP-PS-100, Emergency Director (ED) - Control Room.	________
2.1.2	REFER to Attachment B to establish ES priorities.	________
2.1.3	OPERATE HPCI in accordance with EO-100-032.	________
2.1.4	OPERATE RCIC in accordance with EO-100-033.	________
2.2	MONITOR plant parameters on available instrumentation as identified on Attachment A.	

CAUTION

DIESEL GENERATOR FAILURE IS IMMINENT IF OPERATED WITHOUT ESW COOLING LONGER THAN 4.5 MINUTES LOADED OR 8 MINUTES UNLOADED.

2.3	MANUALLY ATTEMPT to start all Diesel Generators OG501A(B)(C)(D), as follows:	
2.3.1	At 0C653, PLACE D/G A(B)(C)(D) GOV MODE SEL HS-00055A(B)(C)(D) in ISOCH.	________
2.3.2	To start diesel at 0C653, DEPRESS DG A(B)(C)(D) START HS-00051A(B)(C)(D) push button.	________

#3

SSES 2001 NRC Initial Written Examination
Post-Examination Comments

Applicant Level: ☑ RO ☐ SRO

Applicant Name:

Question Type: ☑ Common ☐ RO only ☐ SRO only

Question #: RO 33 (enter number, if SRO only enter N/A)
SRO 29 (enter number, if RO only enter N/A)

Answer: (circle the answer key response) A B C D

Reference: (enter the answer key reference below) *
ON-149-001, Sect 3.4

Comment: (enter the comment below)
Bleed and feed with CRD & Recirc is another acceptable method

Recommendation: (The grader is encouraged to discuss the matter with the NRC Chief Examiner before proceeding with the grading)
☐ Change the correct answer. ☐ Do NOT change the correct answer.
☑ Accept two correct answers. ☐ Delete the question
☑ Make clarifications to the question.

Changes / clarifications made to examination: (provide a description)
Accept 2 answers a & d per ON-149-001 Discussion and Attachment B both methods are acceptable

Reference(s) to support change / clarification made to examination:
ON-149-001, Discussion and Att. B

Justification for rejection of an applicant's comment:

Proctor: ☑ Change made in INK on the master examination copy
Edwin W. Bowles
Print Name
Edwin W. Bowles 8/10/2001
Signature Date
08/14/01

* this method was also mentioned by a License class instructor during Exam Review.

OPERATIONS QUESTION AND ANSWER INPUT FORM

RO 33 SRO ~~30~~ 29

(A) AD045 — Course (B) — Objective

(C) Question Type (check one)
[X] Multiple Choice
[ ] Matching
[ ] Free Format (Essay)

(D) Bank
Operations [X]
OP002 [ ]

(E)	1	2	3	4	5	6	7	8
Keywords :	Category	Topic 1	Topic 2	JTA	Setting	Other Objs.	Quiz Only	Retired
(≤9 characters)	OPS	ONXXXXXX						

(F) Point Value: 1 (G) Answer Time: (Minutes)

(I) Review Date (YYMM):

(H) Cognitive Level: (Check one)
[X] 1 Memory
[ ] 2 Comprehension
[ ] 3 Application
[ ] 4 Analysis
[ ] 5 Problem Solving

(J) QUESTION:

Plant conditions are as follows:

- Reactor has been in Cold Shutdown for 2 days following power operation.
- Reactor water level is +87 inches.
- Both reactor recirc pumps are tagged out of service.
- Shutdown cooling has isolated and the shutdown cooling suction valves cannot be opened.

Which one of the following operator actions will reverse or prevent reactor vessel stratification AND provide alternate decay heat removal?

a. Place Reactor Water Cleanup in service in recirculation.

b. Insert a manual scram to maximize Control Rod Drive flow to the RPV.

c. Start a second Control Rod Drive pump and maximize cooling water D/P.

d. Begin rejecting water with Reactor Water Cleanup while injecting with CRD.

(K) ANSWER: a.

Accept two answers a. & d. per ON-149-001 Discussion and Attachment B a & d are both acceptable methods.

E. W. Bowler

OPERATIONS QUESTION AND ANSWER INPUT FORM

(L) REQUIRED MATERIALS: None

(M) K&A NUMBER/RATING: 295021, AK2.02/3.2

(N) NOTES:

JUSTIFICATION:	By recirculating water from the bottom vessel drain and discharging into the feed system the RWCU system will help circulate water in the RPV and cool by discharging heat to the Non-Regen HX.
DISTRACTER B:	Although this will add cool water it will add to stratification.
DISTRACTER C:	Although this will add cool water it will add to stratification.
DISTRACTER D:	This will remove water and inject cooler water but will not provide circulation and is NOT a method of cooling specified in Attachment B of ON-149-001.

EXAM OUTLINE CROSS-REF:	LEVEL:	RO	SRO
	TIER:	1	1
	GROUP:	3	2
K/A TEXT:	AK2.02 – Knowledge of the interrelations between LOSS OF SHUTDOWN COOLING and the following: Reactor water cleanup.		

QUESTION SOURCE:	BANK:	
	MODIFIED:	
	NEW:	X

10CFR55:	41(b).10, 43(b).5
COMMENTS:	

(O) REFERENCES: ON-149-001, Sect. 3.4 and App. B

(P) POSITIONS: (check one or more boxes)

R – RO	S - SRO	A - ASO	N - NPO	T - STA
X	X			

(Q) Prepared by ED BOWLES (R) Reviewed by: R.E. Cl

The most preferred methods for all reactor modes would be one or both of the following to maintain reactor coolant temperature at desired value:

- Simple feed and bleed operation using Reactor Water Cleanup, RHR Shutdown Cooling in level control letdown operation and Fuel Pool Cooling and Cleanup, as available and as required to letdown; and using CRD, Condensate System, Condensate Transfer, Core Spray and RHR to makeup. It should be noted when selecting a makeup source that although condensate is preferred, suppression pool water may be used without causing severe excursions to reactor coolant. Suppression Pool water is sampled monthly by Chemistry and high water quality is maintained.

- Use of available heat exchangers in Reactor Water Cleanup System and Fuel Pool Cooling and Cleanup System, as available and as required.

If these methods fail to maintain desired temperature, other methods must be used. Determination of method or methods employed is based on reactor decay heat load, plant status, availability of systems and nature of loss.

In Mode 3 with Primary and Secondary containment established, reactor pressure may be maintained greater than 20 psig (to clear column of water from SRV discharge downcomers if steam flow is routed or diverted there) and less than 98 psig (below reactor high pressure isolation). Steam may be routed to the main condenser or suppression pool and methods of makeup previously discussed may be used to maintain level. This method may also be used in Mode 4 with Primary and Secondary containments established. Reactor pressure and temperature may be allowed to rise until within pressure limits cooling by boiling as above. One factor to be considered in Mode 4 is that using this method will result in entering the Emergency Plan. The major factor that must be determined before using this method is that Primary and Secondary containments are established in order to prevent release of radioactive materials to the environment.

SYSTEM/EQUIPMENT AVAILABILITY DETERMINATION

SYSTEMS				STATUS (Circle One)		CHECKED
1.	Primary Containment (Mode 3 or 4 only)			avail	unavail	______
2.	Secondary Containment			avail	unavail	______
3.	Flowpath from reactor to Condenser w/vacuum maintained by SJAE			avail	unavail	______
4.	RPS Channel A1/A2			avail	unavail	______
5.	RPS Channel B1/B2			avail	unavail	______
6.	Methods to M/U to RX					
	a.	CRD		avail	unavail	______
	b.	Condensate		avail	unavail	______
	c.	Condensate Transfer				
		(1)	Keepfill	avail	unavail	______
		(2)	SDC Flush	avail	unavail	______
		(3)	*Skimmer Surge Tank	avail	unavail	______
	d.	RHR		avail	unavail	______
	e.	Core Spray		avail	unavail	______
7.	Methods of Letdown from RX					
	a.	RWCU				
		(1)	Main Condenser	avail	unavail	______
		(2)	Radwaste	avail	unavail	______
	b.	RHR		avail	unavail	______
	c.	SRV's to Supp Pool		avail	unavail	______

SYSTEM/EQUIPMENT AVAILABILITY DETERMINATION

	SYSTEMS		STATUS (Circle One)		CHECKED
8.	*Fuel Pool Gates		installed	not installed	______
9.	*Cask Storage Pit Gates		installed	not installed	______
10.	*Method of Cooling				
	a.	FPC and Cleanup	avail	unavail	______
	b.	RWCU Recirculation	avail	unavail	______
	c.	RHR in FPC Assist	avail	unavail	______
	d.	U-2 FPC and Cleanup	avail	unavail	______
	e.	U-2 RHR in FPC Assist	avail	unavail	______

* Applicable in Mode 5 and level >22 feet above flange.

#4

SSES 2001 NRC Initial Written Examination
Post-Examination Comments

* Identified during exam review by License Class Instructor

Applicant Level: ☐ RO ☐ SRO

Applicant Name:

Question Type: ☐ Common ☑ RO only ☐ SRO only

Question #: RO 50 (enter number, if SRO only enter N/A)
SRO ____ (enter number, if RO only enter N/A)

Answer: (circle the answer key response) A B C D

Reference: (enter the answer key reference below)
ON-164-001 Sect 3.0 And 5.0

Comment: (enter the comment below)
Placing the mode switch for the "A" Flow Unit in Zero (0) will cause a Rod Block

Recommendation: (The grader is encouraged to discuss the matter with the NRC Chief Examiner before proceeding with the grading)
☑ Change the correct answer. ☐ Do NOT change the correct answer.
☐ Accept two correct answers. ☐ Delete the question
☑ Make clarifications to the question.

Changes / clarifications made to examination: (provide a description)
Changed correct Answer from d to c.
Placing the mode switch out of operate will cause A Momentary APRM Hi Rod Block (Comparitor circuit incorporates break before make logic)

Reference(s) to support change / clarification made to examination: CWD 791E 41/8 E
AR-103-001, AR-104-001, Simulator, Prints MI-C51-19,

Justification for rejection of an applicant's comment:

Proctor: ☑ Change made in INK on the master examination copy And Typed in master copy prior to Exam Grading
Edwin W. Dowles
Print Name
Edwin W Dowles 8/10/2001
Signature Date
JMH 08/14/01

RO 50

(A) SY017 E-9 **(B)** 8 (337)

Course Objective

(C) Question Type (check one)

X	Multiple Choice
	Matching
	Free Format (Essay)

(D) Bank

Operations	X
OP002	

(E)

	1	2	3	4	5	6	7	8
Keywords :	Category	Topic 1	Topic 2	JTA	Setting	Other Objs.	Quiz Only	Retired
(≤9 characters)	PCS	PCSINST						

(F) Point Value: 1 **(G)** Answer Time: (Minutes)

(I) Review Date (YYMM):

(H) Cognitive Level: (Check one)

	1	Memory
X	2	Comprehension
	3	Application
	4	Analysis
	5	Problem Solving

(J) QUESTION:

Unit 1 is operating at 65% power while a surveillance test is being performed on the recirculation drive flow instruments. During the surveillance the Mode Switch for the A Flow Unit is placed in zero (0) without first bypassing the flow unit. Which one of the following will occur and what action is required?

a. Several control room annunciators alarm and a rod block occurs, NO half scrams occur, bypass the A Flow Unit.

b. Several control room annunciators alarm and a full scram occurs, enter ON-100-101, SCRAM and take the immediate actions.

c. Several control room annunciators alarm and a rod block and half scram occur, bypass the A Flow Unit and reset the half scram.

d. Control room annunciator APRM/RBM FLOW REFERENCE OFF NORMAL activates, NO rod block or trips occur, bypass the A Flow Unit.

(K) ANSWER: ~~d.~~ a

correct answer is a. Per alarm response procedures and logic diagrams, the effect of this action is a
(1) Rod Out Block and
(2) APRM/RBM Flow reference off Normal

R. E. Chin E. W. Dowd

File No. R11-2

OPERATIONS QUESTION AND ANSWER INPUT FORM

(L) REQUIRED MATERIALS:

(M) K&A NUMBER/RATING: 216000 / A2.05 / 2.8

(N) NOTES:

when taken out of operate, ① ROD OUT Block ② APRM/RBM FLOW REF OFF NORMAL

JUSTIFICATION:	During the surveillance test the mode switch for the flow unit is placed in zero (0) to simulate a high flow condition, which causes the auctionering circuit to shift to the other flow instrument, which in this condition is un-affected. No blocks or trips occur. (This same mode switch manipulation is an operator action on a flow unit failure.)			
DISTRACTER A:	~~No trips or blocks occur.~~ ROD Block AND APRM/RBM Flow Ref OFF NORM			
DISTRACTER B:	~~No trips or blocks occur.~~ ROD Block AND " " "			
DISTRACTER C:	~~No trips or blocks occur.~~ ROD Block AND " " "			
EXAM OUTLINE CROSS-REF:	LEVEL:	RO	-	
	TIER:	2	-	
	GROUP:	1	-	
K/A TEXT:	A2.05 – Ability to (a) predict the impacts of the following on the NUCLEAR BOILER INSTRUMENTATION SYSTEM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Surveillance testing.			
QUESTION SOURCE:	BANK:			
	MODIFIED:			
	NEW:	X		
10CFR55:	41(b)10			
COMMENTS:				

(O) REFERENCES: ON-164-001, Sect. 3.0 and 5.0

(P) POSITIONS: (check one or more boxes)

R – RO	S - SRO	A - ASO	N - NPO	T - STA
X	X			X

(Q) Prepared by Phil Ballard

(R) Reviewed by: R.E. Cl[illegible]

Form STCP-QA-325D
Rev. 3, (8/95)
Page 1 of 1

File No. R11-2

APRM/RBM FLOW/REFERENCE OFF NORMAL (C05)

SETPOINT: See Probable Cause

ORIGIN: Relays 1K1 or 1K3 for Flow Units A, B, C or D

1. PROBABLE CAUSE:

1.1 Flow signal from Reactor Recirc Loops ≥ 114/125% sensed by Flow Units A, B, C, or D.
1.2 Differential flow ≥ 10% between Flow Units A and C or C and D or A and B or B and D.
1.3 Inop condition in Flow Units A, B, C or D.
 1.3.1 Mode switch not in operate.
 1.3.2 Any internal module unplugged.
 1.3.3 Upscale trip.
1.4 Faulty or failed instrument.

2. OPERATOR ACTION:

2.1 If Flow Unit inoperative or out of calibration, ATTEMPT to bypass to remove Rod Block in accordance with ON-164-001 Recirc Drive Flow Instrument Failure.
2.2 If alarm caused by actual high Recirc loop flow, REDUCE flow.
2.3 If Flow Unit INOP, COMPLY with TS 3.3.1.1 and REVIEW TS Bases 3.3.1.1 section 2.b.

3. AUTOMATIC ACTION:

Rod Block to Reactor Manual Control System.

4. REFERENCE:

4.1 M1-C51-19(47)
4.2 E-323 SH 34
4.3 IOM 305
4.4 TS 3.3.1.1

ROD OUT BLOCK (H03)	SETPOINT: See Probable Cause ORIGIN: Rod Drive Control Cabinet

1.0 PROBABLE CAUSE:

1.1 Rod Block caused by any of following:

- 1.1.1 APRM Upscale - 0.58W + 50%
- 1.1.2 APRM Inop
- 1.1.3 Flow Unit Upscale-114/125 flow.
- 1.1.4 Flow Unit Inop
- 1.1.5 Flow Unit Comparitor Trip -10% between Flow Units
- 1.1.6 RBM Upscale $\leq 0.58W + 52\%$(1)
- 1.1.7 RBM Inop
- 1.1.8 Scram Disch High Level ≤ 35.9 gallons
- 1.1.9 Scram Disch Volume High Level Scram - Bypass
- 1.1.10 RWM Rod Block

1.2 Rod Block cause by any of following in Run position only:

- 1.2.1 APRM Downscale - 5%
- 1.2.2 RBM Downscale - 5%

1.3 Rod Block caused by any of following in Startup or Refuel position only:

- 1.3.1 SRM Upscale - $2x10^5$ cps
- 1.3.2 SRM Downscale - 3 cps
- 1.3.3 SRM Inop
- 1.3.4 SRM Detector Position Wrong
- 1.3.5 IRM Upscale-108/125% of scale.
- 1.3.6 IRM Inop
- 1.3.7 IRM Downscale-5/125% of scale
- 1.3.8 IRM Detector Position Wrong
- 1.3.9 Service Platform Hoist loaded.
- 1.3.10 Refueling Platform over core-Startup position.

1.4 Rod Block caused by any of following in Refuel position only:

- 1.4.1 Refuel Platform over core and Fuel Grapple not Full up.
- 1.4.2 Refuel Platform over core and Fuel Grapple loaded.
- 1.4.3 Refuel Platform over core and Frame Hoist loaded.
- 1.4.4 Refuel Platform over core and Trolley Hoist loaded.
- 1.4.5 No Rod selected.
- 1.4.6 One Control Rod withdrawn.

1.5 Rod Block caused by RSCS when in Run or Startup position.

2.0 OPERATOR ACTION:

2.1 DETERMINE cause of Rod Block observing appropriate annunciator.

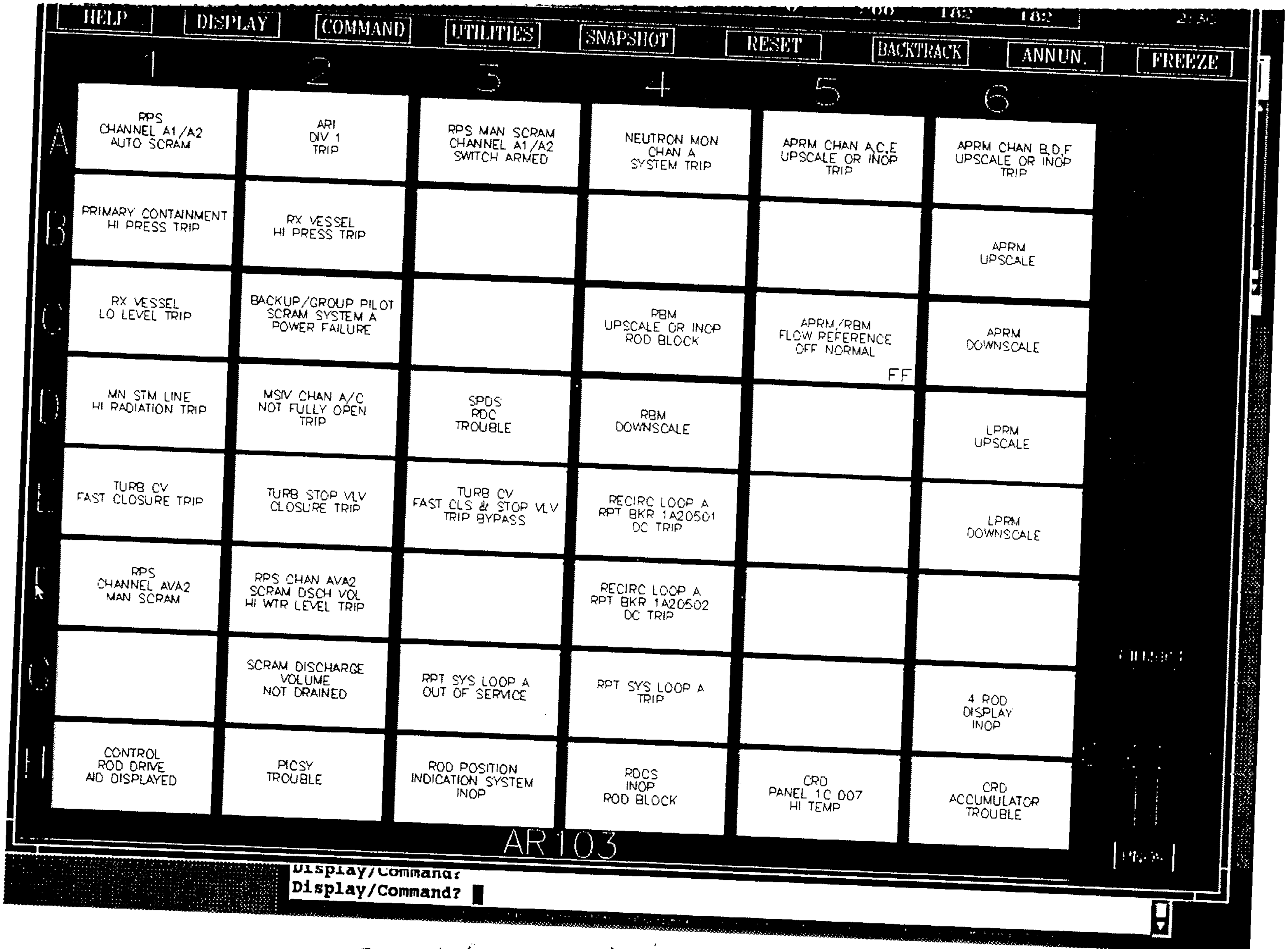
2.2 REFER to appropriate Alarm Response procedure.

3.0 AUTOMATIC ACTION:

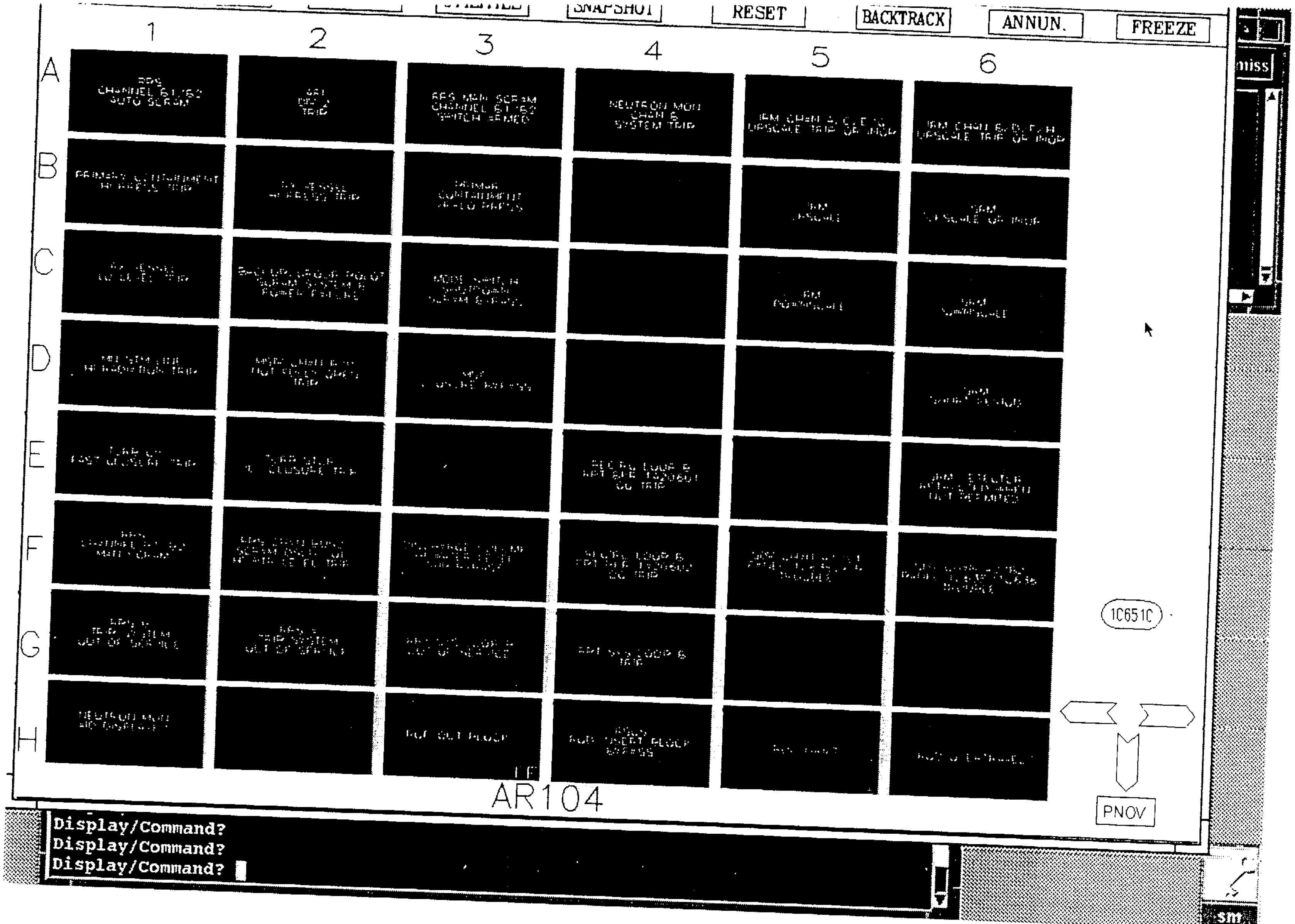
Stops control rod movement on receipt of annunciator.

4.0 REFERENCE:

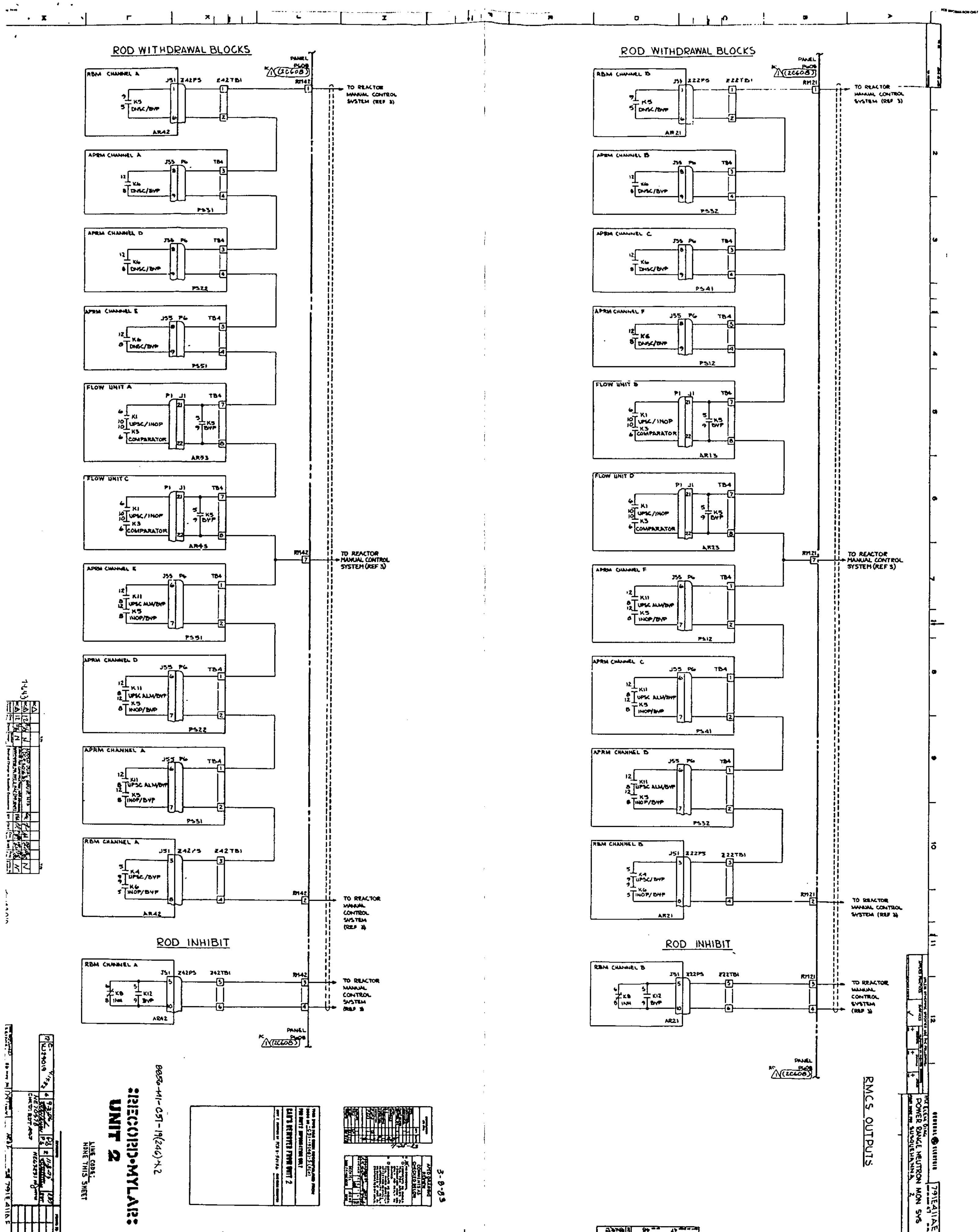
- 4.1 M1-C12-90(28)
- 4.2 M1-H12-778(1)
- 4.3 IOM 305
- 4.4 E-323 SH 35
- (1) 4.5 TS Amendment 176



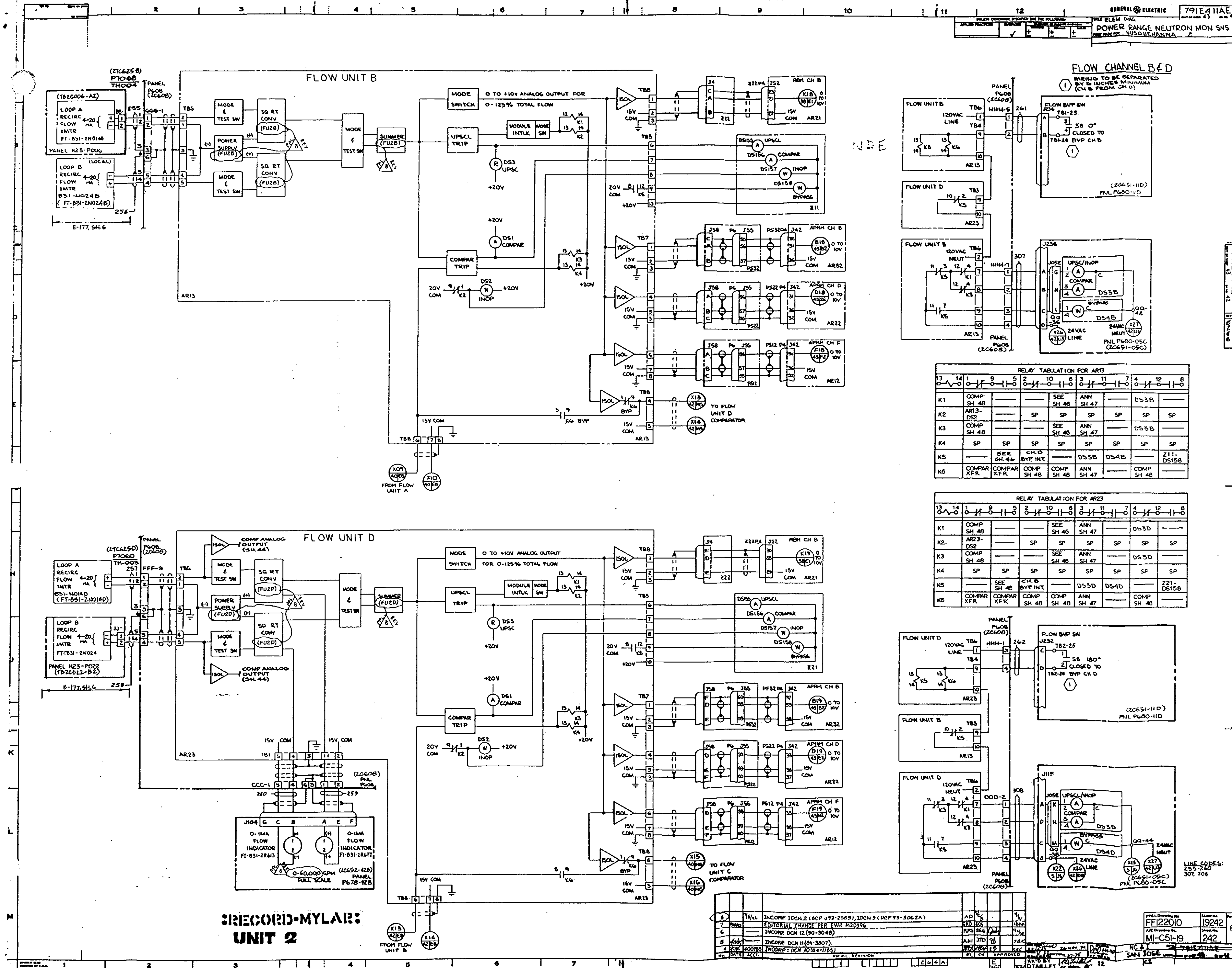
Simulator Verification



Simulator Verification



ROD WITHDRAWAL BLOCKS
RBM CHANNEL A
APRM CHANNEL A
APRM CHANNEL D
APRM CHANNEL E
FLOW UNIT A
FLOW UNIT C
RBM CHANNEL B
APRM CHANNEL B
APRM CHANNEL C
APRM CHANNEL F
FLOW UNIT B
FLOW UNIT D
DNSC/BYP
UPSC/INOP
COMPARATOR
UPSC ALM/BYP
INOP/BYP
UPSC/BYP
TO REACTOR MANUAL CONTROL SYSTEM (REF 3)
ROD INHIBIT
PANEL
RMCS OUTPUTS
RECORD-MYLAR
UNIT 2
POWER RANGE NEUTRON MON SYS
791E411AE
3-8-83



RELAY TABULATION FOR AR13	1–9	9–5	2–10	10–6	3–11	11–7	4–12	12–8
K1	COMP SH 48	—	—	SEE SH 46	ANN SH 47	—	DS3B	—
K2	AR13-DS2	—	SP	SP	SP	SP	SP	SP
K3	COMP SH 48	—	—	SEE SH 46	ANN SH 47	—	DS3B	—
K4	SP	SP	SP	SP	SP	SP	SP	SP
K5	—	SEE SH. 46	CH. D BYP. INT.	—	DS3B	DS4B	—	Z11-DS158
K6	COMPAR XFR	COMPAR XFR	COMP SH 48	COMP SH 48	ANN SH 47	—	COMP SH 48	—

RELAY TABULATION FOR AR23	1–9	9–5	2–10	10–6	3–11	11–7	4–12	12–8
K1	COMP SH 48	—	—	SEE SH 46	ANN SH 47	—	DS3D	—
K2	AR23-DS2	—	SP	SP	SP	SP	SP	SP
K3	COMP SH 48	—	—	SEE SH 46	ANN SH 47	—	DS3D	—
K4	SP	SP	SP	SP	SP	SP	SP	SP
K5	—	SEE SH 46	CH. B BYP. INT.	—	DS3D	DS4D	—	Z21-DS158
K6	COMPAR XFR	COMPAR XFR	COMP SH 48	COMP SH 48	ANN SH 47	—	COMP SH 48	—

No.	DATE	ACCT.	PP&L REVISION	BY	CK	APPROVED
8		7/5/94	INCORP. IDCN 2 (DCP J93-2085), IDCN 5 (DCP 93-3062A)	AD		
7			EDITORIAL CHANGE PER EWR M20396	GKD	DOS	
6		—	INCORP. DCN 12 (90-3048)	RPS	SEG	
5		—	INCORP. DCN 11 (84-3807).	AJH	JJD	
4		400783	INCORP: DCN 10 (84-1155)			

PP&L Drawing No. FF122010 Sheet No. 19242 Rev. 8
A/E Drawing No. MI-C51-19 Sheet No. 242
NC & I SAN JOSE

#5

SSES 2001 NRC Initial Written Examination
Post-Examination Comments

Applicant Level: ☑ RO ☐ SRO

Applicant Name:

Question Type: ☑ Common ☐ RO only ☐ SRO only

Question #: RO 52 (enter number, if SRO only enter N/A)
SRO 41 (enter number, if RO only enter N/A)

Answer: (circle the answer key response) A B C D

Reference: (enter the answer key reference below)
SY017 C-5, OP-150-001 3.2

Comment: (enter the comment below)
Stem Does not state that both channels have tripped on that rack. "B" could be correct if only one switch actuated.

Recommendation: (The grader is encouraged to discuss the matter with the NRC Chief Examiner before proceeding with the grading)
☐ Change the correct answer. ☑ Do NOT change the correct answer.
☐ Accept two correct answers. ☐ Delete the question
☐ Make clarifications to the question.

Changes / clarifications made to examination: (provide a description)

Reference(s) to support change / clarification made to examination:

Justification for rejection of an applicant's comment:
Question states 1 Division

Proctor: ☐ Change made in INK on the master examination copy
Edwin W. Bowles
Print Name
Edwin W. Bowles 8/10/2001
Signature Date
08/14/01

RO 52 SRO ~~54~~ 41

(A) SY017 C-5 — Course

(B) ______ — Objective

(C) Question Type (check one)
- [X] Multiple Choice
- [ ] Matching
- [ ] Free Format (Essay)

(D) Bank
- Operations [X]
- OP002 [ ]

(E)	1	2	3	4	5	6	7	8
Keywords	Category	Topic 1	Topic 2	JTA	Setting	Other Objs.	Quiz Only	Retired
(≤9 characters)	Systems	RCIC						

(F) Point Value: 1

(G) Answer Time: ____ (Minutes)

(I) Review Date (YYMM): ____

(H) Cognitive Level: (Check one)
- [ ] 1 Memory
- [X] 2 Comprehension
- [ ] 3 Application
- [ ] 4 Analysis
- [ ] 5 Problem Solving

(J) QUESTION:

Unit 1 was at 30% power when a reactor scram occurred on a loss of vacuum after circulating water was lost. After the initial scram actions were taken the following occurred:

- Reactor Core Isolation Cooling (RCIC) was placed in pressure control mode per OP-150-001.
- Workers in the Reactor Building bump Instrument Rack 1C004 causing a Division 1 Low RPV Level Trip (-30 inches).

Which one of the following is the effect on RCIC and the reasons for that effect?

a. No effects because RCIC will NOT realign after being manually placed in this line-up.

b. No effects, RCIC remains in pressure control mode, because only one division is effected.

c. RCIC automatically aligns for RPV injection because only one division is required for system initiation.

d. RCIC automatically aligns for RPV injection after RPV level lowers to actuate the Division 2 RPV Level Trip.

(K) ANSWER: c.

(L) REQUIRED MATERIALS: None

(M) K&A NUMBER/RATING: 217000, K1.02/3.5

(N) NOTES:

JUSTIFICATION:	Tripping one division of RCIC initiation will cause RCIC to automatically shift from pressure control mode to injection mode.		
DISTRACTER A:	RCIC will inject.		
DISTRACTER B:	RCIC will inject.		
DISTRACTER D:	RCIC does NOT have to wait for the second initiation signal it will inject with only one division actuated.		
EXAM OUTLINE CROSS-REF:	LEVEL:	RO	SRO
	TIER:	2	2
	GROUP:	1	1
K/A TEXT:	K1.02 - Knowledge of the physical connections and/or cause-effect relationships between REACTOR CORE ISOLATION COOLING (RCIC) SYSTEM and the following: Nuclear boiler system.		
QUESTION SOURCE:	BANK:		
	MODIFIED:	X	
	NEW:		
10CFR55:	41(b).7		
COMMENTS:			

(O) REFERENCES: ON-150-001, Sect. 3.2 and SY017, C-5

(P) POSITIONS: (check one or more boxes)	R – RO	S - SRO	A - ASO	N - NPO	T - STA
	X	X			

(Q) Prepared by ED BOWLES

(R) Reviewed by: R.E. Chi

#6

SSES 2001 NRC Initial Written Examination
Post-Examination Comments

Applicant Level: ☑ RO ☑ SRO

Applicant Name:

Question Type: ☐ Common ☐ RO only ☐ SRO only

Question #: RO 54 (enter number, if SRO only enter N/A)
SRO 43 (enter number, if RO only enter N/A)

Answer: (circle the answer key response) A B C D

Reference: (enter the answer key reference below)

Comment: (enter the comment below) *

C could be correct because RBCCW aligns to take over cooling to drywell on LOOP.

Recommendation: (The grader is encouraged to discuss the matter with the NRC Chief Examiner before proceeding with the grading)

☑ Change the correct answer. ☐ Do NOT change the correct answer.
☐ Accept two correct answers. ☐ Delete the question
☑ Make clarifications to the question.

Changes / clarifications made to examination: (provide a description)

Answer changed from D. to C on a LOOP, RBCCW Drywell loads shift to RBCCW

Reference(s) to support change / clarification made to examination:

ON-104-001

Justification for rejection of an applicant's comment:

Proctor: ☑ Change made in INK on the master examination copy And typed in Final Exam copy prior to Exam grading

Edwin W. Bowles
Print Name

[signature] 8/10/2001
Signature Date

[signature] 08/14/01

* This comment was anticipated based on Instructor review of the Exam

RO 54 SRO ~~56~~ 43

(A) SY017 E-6 (B)
Course Objective

(C) Question Type (check one)
[X] Multiple Choice
[ ] Matching
[ ] Free Format (Essay)

(D) Bank
Operations [X]
OP002 [ ]

(E)	1	2	3	4	5	6	7	8
Keywords :	Category	Topic 1	Topic 2	JTA	Setting	Other Objs.	Quiz Only	Retired
(≤9 characters)	PCS	ATMOSCTL						

(F) Point Value: 1 (G) Answer Time: (Minutes)

(I) Review Date (YYMM):

(H) Cognitive Level: (Check one)
[X] 1 Memory
[ ] 2 Comprehension
[ ] 3 Application
[ ] 4 Analysis
[ ] 5 Problem Solving

(J) QUESTION:

With Unit 1 at 85% power when a load reject and loss of off-site power occur. The diesel generators start and power their associated buses.

Which one of the following describes the effect on Drywell Cooling?

The operating drywell unit coolers trip, then...

a. remain shutdown until manually started.

b. restart when the diesels start and are cooled by RBCW.

c. restart when the diesels start and are cooled by RBCCW.

d. restart when the diesels start and run without cooling water.

(K) ANSWER: ~~d.~~ c.

per ON-104-001 Step 3.11

Drywell Cooling shifts to RBCCW when RBCW trips

Also discussion section pg 16, paragraph 2

E W B[illegible]

R.E. Cl[illegible]

OPERATIONS QUESTION AND ANSWER INPUT FORM

(L) **REQUIRED MATERIALS:** None

(M) **K&A NUMBER/RATING:** 223001 / K2.09 / 2.7 / 2.9

(N) **NOTES:**

JUSTIFICATION:	The power supplies to the drywell unit coolers trip but the fans restart and after cooling water is verified available cooling can be restored to the drywell.
DISTRACTER A:	The fans trip and restart.
DISTRACTER B:	The fans trip and restart.
DISTRACTER C:	The fans trip and restart.

EXAM OUTLINE CROSS-REF:	**LEVEL:**	RO	SRO
	TIER:	2	2
	GROUP:	1	1
K/A TEXT:	K2.09 – Knowledge of the electrical power supplies to the following: Drywell Cooling Fans.		

QUESTION SOURCE:	**BANK:**	
	MODIFIED:	
	NEW:	X

10CFR55:	41(b)(7)
COMMENTS:	

(O) **REFERENCES:** ON-104-001

(P) **POSITIONS:**
(check one or more boxes)

R – RO	S - SRO	A - ASO	N - NPO	T - STA
X	X			

(Q) Prepared by Phil Ballard

(R) Reviewed by: R.L. Chin

PROCEDURE COVER SHEET

PPL SUSQUEHANNA, LLC	NUCLEAR DEPARTMENT PROCEDURE	
UNIT 1 RESPONSE TO LOSS OF ALL OFFSITE POWER		ON-104-001 Revision 13 Page 1 of 17

QUALITY CLASSIFICATION:

(X) QA Program ( ) Non-QA Program

APPROVAL CLASSIFICATION:

(X) Plant ( ) Non-Plant

( ) Instruction

EFFECTIVE DATE: 06/15/01

PERIODIC REVIEW FREQUENCY: 2 Year

PERIODIC REVIEW DUE DATE: 06/30/03

RECOMMENDED REVIEWS:

Procedure Owner: Shift Technical Advisor-C Shift

Responsible Supervisor: Shift Supervisor-C Shift

Responsible FUM: Manager-Nuclear Operations

Responsible Approver: Manager-Nuclear Operations

CONTROL

3. OPERATOR ACTIONS

CHECKED

3.1 RECORD date and time of event.

______________________ Shift Supervision ____/____ Date Time

3.2 ENSURE proper Diesel Generator operation in accordance with OP-024-001. ________

(1) 3.3 If Diesel Generator A, B or C fail to start, PERFORM Attachment A approximately 30 minutes into LOOP EVENT to ensure 4 hour capacity of 250V DC batteries 1D650 and 1D660:

3.4 REFER to EP-PS-100, Emergency Director - Control Room, to classify situation. ________

3.5 ENSURE ESW Pumps 0P504A, B, C, D operating in accordance with OP-054-001 after Diesel Generators start and re-energize Emergency Busses. ________

3.6 On direction of Shift Supervision, TRANSFER following systems cooling water supply from Service Water to Emergency Service Water in accordance with OP-111-001:

3.6.1 Reactor Building Closed Cooling Water ________

3.6.2 Turbine Building Closed Cooling Water ________

3.7 RESTORE Reactor Building Closed Cooling Water in accordance with ON-114-001 Loss of RBCCW. ________

3.8 RESTORE Turbine Building Closed Cooling Water in accordance with ON-115-001 Loss of TBCCW. ________

3.9 RESTORE Instrument Air System in accordance with ON-118-001 Loss of Instrument Air. ________

3.10 RESTORE RPS in accordance with ON-158-001 Loss of RPS. ________

3.11 CHECK shift of Drywell Cooling from Reactor Building Chilled Water to Reactor Building Closed Cooling Water:

3.11.1 RBCCW Supply Vlv FV-18771D OPEN. ________

3.11.2 RBCCW Return Vlv FV-18771C OPEN. ________

			CHECKED
	3.11.3	Chilled Water Supply Vlv to Drywell Coolers FV-18771B CLOSE.	________
	3.11.4	Chilled Water Return Vlv from Drywell Coolers FV-18771A CLOSE.	________
	3.11.5	If desired, ALIGN control switch for Drywell Cooling Water Control Valves to the RBCCW position at panel 1C279.	________
3.12	ENSURE following Drywell Cooler Isolation Valves OPEN:		
	3.12.1	A Clrs Clg Wtr OB Iso Valves HV-18781A1&A2.	________
	3.12.2	A Clrs Clg Wtr IB Iso Valves HV-18782B1&B2.	________
	3.12.3	B Clrs Clg Wtr IB Iso Valves HV-18782A1&A2.	________
	3.12.4	B Clrs Clg Wtr OB Iso Valves HV-18781B1&B2.	________
3.13	ENSURE Containment Instrument Gas Isolation valves OPEN:		
	3.13.1	Instr Gas Cmp OB Suct Iso SV-12605.	________
	3.13.2	Instr Gas to Contn Iso SV-12651.	________
3.14	RESTART Control Rod Drive System by starting CRD Pump in accordance with ON-155-007 Loss of CRD System Flow.		________
3.15	RESTORE affected systems to desired status in accordance with ON-158-001 Loss of RPS.		________
3.16	ENSURE following running as necessary:		
	3.16.1	Turbine Generator Turning Gear Oil Pump (1P111).	________
	3.16.2	Turbine Bearing Lift Pumps (1P109A-J).	________
	3.16.3	Turbine Generator Emergency Bearing Oil Pump (1P112).	________
	3.16.4	Turbine Generator Turning Gear (when Main Turbine completed coastdown) (1S103).	________
	3.16.5	RFP Emergency Oil Pumps (1P125A,B,C).	________

After the main steam isolation valves close, decay heat raises reactor pressure to the lowest relief valves setting. Reactor water level decreases to the initiation setpoint for the RCIC and HPCI Systems, which actuate to restore reactor water level. If a Primary System boundary break should occur coincidentally with the loss of Offsite Power sources, the low pressure Core Spray, RHR and Automatic Depressurization Systems will initiate automatically to depressurize the RPV and restore reactor vessel level.

In the case where no break has occurred, immediate action must be taken to restore Drywell Cooling to prevent a high drywell pressure. With Unit Aux Buses 11A and 11B de-energized, Drywell Cooling shifts from RBCW to RBCCW. It is important to confirm that the Drywell Cooler isolation valves and containment instrument gas isolation valves are open once the ESS Buses have been energized by the diesel generators. ESW is aligned to supply the RBCCW and/or TBCCW heat exchangers if adequate cooling to the diesel generators and ECCS equipment is available. The loads of the RBCCW and TBCCW heat exchangers are large enough to permit operation of 2 ESW pumps per loop.

With a complete loss of offsite power, the fuel pool cooling pumps will trip and Reactor Building HVAC Zones 1, 2, and 3 will isolate and go into the recirculation mode. Moisture generated through evaporation and/or boiling of the fuel pool will spread to the secondary containment. Excessive moisture could cause adverse effects to safety related systems and components. ON-135-001 provides instructions for mitigating a Loss of Fuel Pool Cooling event.

Procedures ON-104-201, ON-104-202, ON-104-203 and ON-104-204 contain a list of the 4.16 KV Bus loads and the 480V Load Center loads that are energized on a loss of Offsite Power by the Standby Diesel Generators. ON-103-003 contains a list of all loads off the Unit Auxiliary Buses, and ON-003-001 contains a list of all loads off of Startup Bus 10.

FSAR 8.3.2.1.1.4 specifies each 250V battery has capacity without its charger to independently supply required loads for four (4) hours per FSAR Table 8.3-7. Table 8.3-7 shows various non-1E loads terminating at specified times to ensure four (4) hour capacity, however, plant design does not automatically shed these loads. This procedure sheds non-1E loads at 30 minutes to ensure a four (4) hour battery capacity, per design.

(#7)

SSES 2001 NRC Initial Written Examination
Post-Examination Comments

Applicant Level: ☑ RO ☑ SRO

Applicant Name:

Question Type: ☑ Common ☐ RO only ☐ SRO only

Question #: RO 55 (enter number, if SRO only enter N/A)
SRO 44 (enter number, if RO only enter N/A)

Answer: (circle the answer key response) A B C D

Reference: (enter the answer key reference below)

Comment: (enter the comment below)
"B" could also be correct because ON-184-001 has the I.B. MSIV's opened first. OP-184-001 directs opening the OB MSIV's First. B & D are correct.

Recommendation: (The grader is encouraged to discuss the matter with the NRC Chief Examiner before proceeding with the grading)
☐ Change the correct answer. ☐ Do NOT change the correct answer.
☑ Accept two correct answers. ☐ Delete the question
☑ Make clarifications to the question.

Changes / clarifications made to examination: (provide a description)
Accept 2 answers b & d - there are 2 procedures for re-opening MSIVs - ON-184-001 allows method b. the OP (OP-184-001) is response d.

Reference(s) to support change / clarification made to examination:
OP-184-001, Sect 3.2, ON-184-001

Justification for rejection of an applicant's comment:

Proctor: ☑ Change made in INK on the master examination copy
Edwin W Bowles
Print Name
Edwin W Bowles 8/10/2001
Signature Date

[signature] 08/14/01

OPERATIONS QUESTION AND ANSWER INPUT FORM

RO 55 SRO ~~57~~ 44

(A) SY017 H-2 (B)
Course Objective

(C) Question Type (check one)
[X] Multiple Choice
[ ] Matching
[ ] Free Format (Essay)

(D) Bank
Operations [X]
OP002 [ ]

(E)	1	2	3	4	5	6	7	8
Keywords :	Category	Topic 1	Topic 2	JTA	Setting	Other Objs	Quiz Only	Retired
(≤9 characters)	Systems	MSIVs						

(F) Point Value: 1 (G) Answer Time: (Minutes)

(H) Cognitive Level: (Check one)
[X] 1 Memory
[ ] 2 Comprehension
[ ] 3 Application
[ ] 4 Analysis
[ ] 5 Problem Solving

(I) Review Date (YYMM):

(J) QUESTION:

Unit 1 has scrammed and the MSIVs isolated. The cause of the isolation has been corrected and the MSIV isolation logic reset. With RPV pressure greater than 600 psig which one of the following is required to re-open the MSIVs?

a. Drain the steam lines, bypass the MSIVs with the steam drains, lower the D/P to less than 200 psid then open the inboard then the outboard MSIVs.

b. Open the inboard MSIVs, drain the steam lines, bypass the outboard valves with the steam drains, lower the D/P to less than 50 psid then open the outboard MSIVs.

c. Drain the steam lines, open the outboard MSIVs, bypass the inboard valves with the steam drains, lower the D/P to less than 200 psid then open the inboard MSIVs.

d. Open the outboard MSIVs, drain the steam lines, bypass the inboard valves with the steam drains, lower the D/P to less than 50 psid then open the inboard MSIVs.

(K) ANSWER: d.

there are two procedures for re-opening the MSIVs
the original ref OP-184-001, Sect 3.2
&
ON-184-001, In the ON the Inboard MSIVs are opened first.

Accept b & d.

E. W. [signature]

Form STCP-QA-325D
Rev. 3, (8/95)
Page 1 of 1

File No. R11-2

OPERATIONS QUESTION AND ANSWER INPUT FORM

(L) <u>REQUIRED MATERIALS:</u> None

(M) <u>K&A NUMBER/RATING:</u> 223002, A4.03/3.6

(N) <u>NOTES:</u>

JUSTIFICATION:	Per procedure and system knowledge the outboard MSIVs are opened first, then the lines must be drained, then the D/P lowered to <50 psid then the inboards opened.		
DISTRACTER A:	The lines are drained after the outboard valves are opened to allow steam drains downstream to drain the lines, once drained the steam line drains are used to bypass the inboard MSIVs. The D/P must be lowered to <50 psid.		
DISTRACTER B:	The outboard MSIVs are opened first to allow steam line drains to drain the lines downstream of the inboard MSIVs.		
DISTRACTER C:	The lines are drained after the outboard valves are opened to allow steam drains downstream to drain the lines. The D/P must be lowered to <50 psid.		
EXAM OUTLINE CROSS-REF:	LEVEL:	RO	SRO
	TIER:	2	2
	GROUP:	1	1
K/A TEXT:	A4.03 - Ability to manually operate and/or monitor in the control room: Reset system isolations.		
QUESTION SOURCE:	BANK:		
	MODIFIED:	X	
	NEW:		
10CFR55:	41(b).7, 41(b).10		
COMMENTS:			

(O) <u>REFERENCES:</u> OP-184-001, Sect. 3.2

(P) <u>POSITIONS:</u>
(check one or more boxes)

R – RO	S - SRO	A - ASO	N - NPO	T - STA
X	X			

(Q) Prepared by ED BOWLES

(R) Reviewed by: R.E. Ch

PROCEDURE COVER SHEET

SUSQUEHANNA STEAM ELECTRIC STATION PP&L	NUCLEAR DEPARTMENT PROCEDURE MAIN STEAM LINE ISOLATION AND QUICK RECOVERY	ON-184-001 Revision 6 Page 1 of 9

QUALITY CLASSIFICATION: (X) QA Program ( ) Non-QA Program	APPROVAL CLASSIFICATION: (X) Plant ( ) Non-Plant ( ) Instruction

EFFECTIVE DATE: 09/28/00

PERIODIC REVIEW FREQUENCY: 2 Year

PERIODIC REVIEW DUE DATE: 9-30-02

RECOMMENDED REVIEWS:

Procedure Owner: Shift Technical Advisor-C Shift

Responsible Supervisor: Shift Supervisor-C Shift

Responsible FUM: Manager-Nuclear Operations

Responsible Approver: Manager-Nuclear Operations

CONFIRM

3.16 CLOSE Mn Stm SJAE Iso HV-10107. ________

3.17 When directed by Shift Supervision AND initiating event is determined and cleared, RESET NSSSS Main Steam Line Isolation by depressing:

3.17.1 Mn Stm Line Div 1 Iso Reset HS-B21-1S32 Reset push button. ________

3.17.2 Mn Stm Line Div 2 Iso Reset HS-B21-1S33 Reset push button. ________

NOTE: If primary containment integrity in jeopardy, it is acceptable to open MSIV's (IB first) with ΔP > 200 PSID. This action will not damage MSIV's. If conditions permit, equalizing around MSIV's is preferred.

3.18 To OPEN IB MSIV's PLACE following control switches to AUTO:

3.18.1 Mn Stm Line A IB Iso HV-141-F022A. ________

3.18.2 Mn Stm Line B IB Iso HV-141-F022B. ________

3.18.3 Mn Stm Line C IB Iso HV-141-F022C. ________

3.18.4 Mn Stm Line D IB Iso HV-141-F022D. ________

3.19 ALIGN for steam line pressurization as follows:

3.19.1 PLACE AC MOV OL Byps HS-B21-1S37A to TEST. ________

3.19.2 PLACE DC MOV OL Byps HS-B21-1S37B to TEST. ________

3.19.3 OPEN Mn Stm Line IB Drain HV-141-F016. ________

3.19.4 OPEN Mn Stm Line OB Drain HV-141-F019. ________

3.19.5 ENSURE Mn Steam Line Warm Up HV-141-F020 OPEN. ________

3.19.6 After 2 minutes, PLACE AC MOV OL Byps HS-B21-1S37A to NORM. ________

CONFIRM

3.19.7 After 2 minutes, PLACE DC MOV OL Byps HS-B21-1S37B to NORM. ______

3.20 OBSERVE main steam line pressure INCREASING on Main Stm Press PR-10101C. ______

CAUTION

OPENING MSIV'S WITH LARGE DIFFERENTIAL PRESSURE WILL CAUSE RPV PRESSURE TO DROP RAPIDLY AND RPV LEVEL TO SWELL.

3.21 When differential pressure across MSIVs is between 50 psid and 200 psid, OPEN OB MSIV's by PLACING following control switches to AUTO:

3.21.1 Mn Stm Line A OB Iso HV-141-F028A ______

3.21.2 Mn Stm Line B OB Iso HV-141-F028B ______

3.21.3 Mn Stm Line C OB Iso HV-141-F028C ______

3.21.4 Mn Stm Line D OB Iso HV-141-F028D ______

3.22 If necessary, OPEN any or all of the following drain valves:

3.22.1 OPEN following by depressing Drip Leg Drn HS-10112 OPEN push button:

a. Drip Leg Drn HV-10112A1. ______

b. Drip Leg Drn HV-10112B1. ______

c. Drip Leg Drn HV-10112C1. ______

d. Drip Leg Drn HV-10112D1 ______

3.22.2 OPEN BPV Hdr Drip Leg Drn Byps HV-10108A by depressing HS-10108A OPEN push button. ______

3.22.3 OPEN MSV Bst Drn HV-10101 A,B,C,D by depressing common OPEN push button. ______

3.2 OPENING OF MAIN STEAM ISOLATION VALVES WITH DIFFERENTIAL PRESSURE ACROSS MAIN STEAM ISOLATION VALVES.

3.2.1 Prerequisites

a. Primary Containment Instrument Gas System in operation in accordance with OP-125-001.

b. Instrument Air System in operation in accordance with OP-118-001.

c. DC power available in accordance with OP-102-001 (125V).

d. DC power available in accordance with OP-188-001 (250V).

e. AC power available in accordance with OP-105-001 (480V).

f. No Main Steam Line Isolation signals present.

g. All Main Steam Isolation Valves closed.

h. Reactor Protection System in operation in accordance with OP-158-001.

3.2.2 Precautions

a. Do not open any Main Steam Isolation Valve with differential pressure of more than 200 psid across the valve. Under normal operating conditions, valve opening should be accomplished only after Main Steam pressure has equalized to less than 50 psid across valve.

b. If MSIV Isolation occurred, ensure both Inboard and Outboard MSIV's closed and all MSIV control switches in CLOSE position before MSIV isolation logic reset.

NOTE: Any following condition will cause Main Steam Line Isolation:

a. Main Steam Line pressure < 861 psig with Reactor Mode Switch in Run.

b. Main Condenser Vacuum < 9 inches HgV (20.9 HgA). (Can be bypassed with Reactor Mode Switch not in Run.)

c. Reactor Vessel Water level 1.

d. Main Steam Line High radiation (15 times normal).

e. Main Steam Tunnel temperature > 177°F.

f. Main Steam Tunnel differential temperature > 99°F.

g. Main Steam Line High flow (134%).

h. Logic Power Failure.

3.2.3 RESET Main Steam Line Isolation by depressing MN STM LINE DIV 1(2) ISO RESET HS-B21-1S32(S33) push button.

3.2.4 With differential pressure across Main Steam Isolation Valves, OPEN each Valve as follows:

a. PLACE each MN STM LINE A, B, C, D OB ISO HV-141-F028A, B, C, D control switch to AUTO.

b. CHECK each Outboard MSIV OPENS by observing Red indicating light ILLUMINATED.

c. CHECK OPEN MN STM LINE IB DRAIN HV-141-F016.

d. CHECK OPEN MAIN STM LINE OB DRAIN HV-141-F019.

e. JOG OPEN MN STM LINE DRAIN TO CDSR HV-141-F021.

f. ALLOW drain lines to heat up and drain any condensation to Main Condenser.

g. CLOSE MN STM LINE DRAIN TO CDSR HV-141-F021.

h. JOG OPEN MN STEAM LINE WARMUP HV-141-F020 to allow Main Steam Line pressure to increase.

i. CLOSE following by depressing DRP LEG DRN HS-10112:

(1) Drip Leg DRN HV-10112A1.

(2) Drip Leg DRN HV-10112B1.

(3) Drip Leg DRN HV-10112C1.

(4) Drip Leg DRN HV-10112D1.

j. CLOSE BPV HDR DRIP LEG DRN BYPS HV-10108A by depressing HS-10108A AUTO push button.

k. DEPRESS common CLOSE push button for MSV BST DRN HV-10101 A, B, C, D to close following:

(1) MSV-1 Before Seat Drain HV-10101A.

(2) MSV-2 Before Seat Drain HV-10101B.

(3) MSV-3 Before Seat Drain HV-10101C.

(4) MSV-4 Before Seat Drain HV-10101D.

l. If pressure not equalizing across Inboard MSIV's, CLOSE following:

(1) MN STM SJAE ISO HV-10107.

(2) SSE MN STM SUP HV-10109.

(3) RFPT MN STM SUP ISO HV-10111.

m. If Reactor Pressure higher than EHC pressure, RAISE EHC pressure by adjusting Main Turbine PRESSURE SETPOINT SELECTOR to prevent any rapid change in Reactor pressure.

n. When differential pressure across Inboard MSIV's equalized or < 50 psid, OPEN Inboard MSIV's by placing each MN STM LINE A, B, C, D IB ISO HV-141-F022A, B, C, D control switch to AUTO.

o. CHECK each Inboard MSIV OPENS by observing Red indicating light ILLUMINATED.

p. If closed, OPEN:

(1) MN STM SJAE ISO HV-10107.

(2) SSE MN STM SUP HV-10109.

(3) RFPT MN STM SUP ISO HV-10111.

q. OPEN any or all drain valves, as necessary, listed in step 3.2.4.i of this procedure.

BEFORE EXAM ADMIN.

#8

OPERATIONS QUESTION AND ANSWER INPUT FORM

RO 66 SRO 68

(A) SY017 K-6 Course (B) Objective

(C) Question Type (check one)
[X] Multiple Choice
[ ] Matching
[ ] Free Format (Essay)

(D) Bank
Operations [X]
OP002 [ ]

(E)	1	2	3	4	5	6	7	8
Keywords :	Category	Topic 1	Topic 2	JTA	Setting	Other Objs.	Quiz Only	Retired
(≤9 characters)	Systems	RWM						

(F) Point Value: 1 (G) Answer Time: (Minutes)

(I) Review Date (YYMM):

(H) Cognitive Level: (Check one)
[ ] 1 Memory
[ ] 2 Comprehension
[ ] 3 Application
[X] 4 Analysis
[ ] 5 Problem Solving

(J) QUESTION:

Control rods are being withdrawn during a plant startup. The following conditions exist:

- The Rod Worth Minimizer (RWM) is in operation
- Control rods are being withdrawn in group 4
- Only one control rod remains to be withdrawn in group 4
- The operator attempts to select and withdraw a control rod in group 5

Which one of the following describes the response of the control rod in group 5, the response of the RWM, and the required action?

The control rod...

a. will NOT withdraw, a control rod withdraw block will be applied to this rod.
Select the correct control rod in group 4.

b. will NOT withdraw, a select block will be applied to the control rod in group 5.
Bypass the RWM then select the correct control rod in group 4.

c. will withdraw to its withdraw limit, the last rod in group 4 will be identified as an insert error.
Promptly insert the control rod in group 5 to position 00.

d. will withdraw only one notch, then control rod withdrawal blocks will be applied to all other control rods.
Position the control rod in group 5 to its intended position.

(K) ANSWER: ~~d.~~ c

OPERATIONS QUESTION AND ANSWER INPUT FORM

(L) REQUIRED MATERIALS: None

(M) K&A NUMBER/RATING: 201006, A2.05/3.1

4(N) NOTES:

JUSTIFICATION:	A rod ~~block~~ will not be applied until the rod moves out ~~of its~~ current position, then blocks will be applied to all rods. This ~~control rod~~ is considered unintended rod motion because an ~~incorrect control~~ rod is selected and moved one notch. The correct action ~~is to~~ move the control rod ~~back to its intended~~ position.		
DISTRACTER A:	~~Blocks will be applied after the rod~~ moves out of its current position.		
DISTRACTER B:	A select block will not be applied		
DISTRACTER ~~C~~: D	This rod cannot be withdrawn to its withdraw limit, rod blocks will be applied after it leaves its initial ~~position~~. If a control rod ~~is mis~~positioned, the correct action is to promptly insert the control rod ~~to~~ position 00. This control rod is considered unintended rod motion ~~because~~ an ~~incorrect~~ control rod is selected and moved one notch.		
EXAM OUTLINE CROSS-REF:	LEVEL:	RO	SRO
	TIER:	2	2
	GROUP:	2	2
K/A TEXT:	A2.05 - Ability to predict the impacts of following on ROD WORTH MINIMIZER and (b) based on those predictions, use procedures to correct, control or mitigate the consequences of those abnormal conditions or operations: Out of sequence rod movement.		
QUESTION SOURCE:	BANK:		
	MODIFIED:		
	NEW:	X	
10CFR55:	41(b).6, 41(b).7		
COMMENTS:			

(O) REFERENCES: Op-131-001, SY017, K-6 NDAP-QA-0338

(P) POSITIONS: (check one or more boxes)

R – RO	S - SRO	A - ASO	N - NPO	T - STA
X	X			

(Q) Prepared by ED BOWLES

(R) Reviewed by: [signature] 8/14/01

The RWM automatically latches into a rod group according to a specific rule. It will select the group, which is the highest group having less than three insert errors and having at least one rod withdrawn past its insert limit. When this occurs, rods contained in the currently latched group or any lower group that are withdrawn past their withdraw limit will be classified as withdraw errors, as will those rods in groups above the currently latched group that are withdrawn past their insert limit.

Upward (or downward) rod group latching also occurs when, upon completion of rod movements in the currently latched group, the operator selects the next in-sequence rod in the next higher (or lower) rod group. For upward latching, this occurs even with existing insert errors, as long as the maximum insert error rule, described above, is met. For downward latching, when this occurs, any rods in higher groups, which are not at the insert limit, are identified as withdraw errors.

Operation With Two Insert Errors

If, while withdrawing rods at power levels below the LPSP, it becomes necessary or desirable to leave one or two rods at positions lower than their withdraw limits, the next higher group can be latched, and the startup allowed to proceed. This may be necessary due to operational problems with the specific rods. It is accomplished as follows:

When the group is reached that contains this one or two rods, which are to be left at some intermediate position, the rod or rods, are withdrawn to their respective positions. (It is assumed here that both rods are in the same rod group, although it is possible for them to be in different rod groups. It is also assumed that the rod group(s) in question is/are below the 50 percent rod density point.)

All other rods in the group are withdrawn to the withdraw limit position for that group.

Any rod in the next higher group is then selected. The RWM program latches (or shifts groups) up to this group, the ROD GROUP display shows the higher group number, and the one (or two) rod(s) remaining inserted in the lower group are identified as insert error(s).

The withdrawal sequence can then proceed normally, so long as no more than two insert errors are allowed to occur.

There are times when the systems that interface with the RWM do not function properly. There is usually a warning that system status is not normal which enables the operator to perform actions to correct the situation prior to reaching off-normal conditions. The first line of defense is the alarm response procedures. These procedures provide guidance to various conditions in the form of probable causes, operator actions, automatic actions, and references, which are applicable to these conditions.

- ROD POSITION INDICATION SYSTEM INOP (AR-103-001-H03)
- ROD OUT BLOCK (AR-104-001-H03)

The details of these procedures can be found in the individual procedures.

There may be times when the RWM is required to be bypassed due to equipment failure or unforeseen circumstances. Procedure NDAP-QA-0338, Reactivity Management and Controls Program, Controls for Reactivity Control Systems (RWM, RSCS, and RBM), provides direction for bypassing the RWM. The procedure contains a flowcharted form to authorize bypassing of the RWM.

ON-155-004, RPIS Failure, provides direction for placing substitute data, bypassing, and re-initializing the RWM when RPIS failures occur. These failures range from a failed reed switch to complete failure of the RPIS system.

TEST MODE OPERATION

The RWM is tested in accordance with the following procedures:

SO-131-001, Startup Operability Demonstration Startup/Following System Failure

This procedure demonstrates the ability of the RWM to indicate a control rod selection error and block rod withdrawal of an out-of-sequence control rod in Mode 2 at $\leq$10 percent of rated thermal power.

The program will change up (latch the next higher group) during an up-power evolution, when all the rods in the currently latched group and in all lower groups have been withdrawn to their respective group withdraw limits, and a rod in the next higher group is selected. There is an exception to the requirement that all rods in previous groups be at their respective group's withdraw limit. The RWM makes provision for a maximum of two rods to be at other than their group withdraw limit without it impacting the system's upward latch (these would appear as "insert errors," as described below).

The number of the currently latched group is displayed in the LATCHED GROUP box on the Main Screen of the RWM Touchscreen Display (Figure 5). An evaluation of the current rod position distribution, to determine the group which should be latched, is performed by the RWM program during system initialization and at various other times during normal program operation. This ensures that the proper group is latched at all times.

Loaded Sequence
Rod movement sequence loaded within RWM System memory.

Low Power Set Point (LPSP)
The Low Power Set Point (LPSP) is the core average power level below which the RWM program enforces adherence to the operating sequence of rod withdrawals or insertions. When Reactor power is above the LPSP, the RWM program does not impose any constraints on operator requested rod movements. That is, the operating sequence ceases to be enforced by the RWM, above the LPSP. Rod blocks due to hardware failure, however, can occur at any power level.

Main Steam Line (MSL) flow is measured by the Feed Water Level Control (FWLC) System to determine when the plant is operating at 22 percent of Rated Thermal Power (RTP). This monitored parameter is inputted to the RDCS and PICSY to activate the LPSP. The setpoint can be adjusted by varying the trip value in the MSL flow sensor.

When the Reactor is operating below the LPSP, the BELOW LPSP box on the RWM Touchscreen Display changes color, from yellow to red, indicating that the RWM is enforcing the loaded sequence.

#9

SSES 2001 NRC Initial Written Examination
Post-Examination Comments

Applicant Level: ☐ RO ☐ SRO

Applicant Name:

Question Type: ☐ Common ☐ RO only ☐ SRO only

Question #: RO 67 (enter number, if SRO only enter N/A)
SRO ____ (enter number, if RO only enter N/A)

Answer: (circle the answer key response) A B C D

Reference: (enter the answer key reference below)

Comment: (enter the comment below) *
ON-178-002 allows insertion of C.R or Raising core Flow. A OR D correct.

Recommendation: (The grader is encouraged to discuss the matter with the NRC Chief Examiner before proceeding with the grading)
☐ Change the correct answer. ☐ Do NOT change the correct answer.
☑ Accept two correct answers. ☐ Delete the question
☑ Make clarifications to the question.

Changes / clarifications made to examination: (provide a description)
Procedure change 7/3/2001 - ON-178-002 Sect 3.4 Allows both methods. NDAP-QA-0338, Att F (pg 1) "CRAM Rods"

Reference(s) to support change / clarification made to examination:
ON-178-002, Sect 3.4 And NDAP-QA-0338, Att F (pg. 1)

Justification for rejection of an applicant's comment:

Proctor: ☑ Change made in INK on the master examination copy

Edwin W. Bowles
Print Name
Edwin W Bowles 8/10/2001
Signature Date
[signature] 08/14/01

* ALSO mentioned during Exam Review with a Licensing Class Instructor

OPERATIONS QUESTION AND ANSWER INPUT FORM

RO 67

(A) SY017 L-9 — Course

(B) ______ — Objective

(C) Question Type (check one)

X	Multiple Choice
	Matching
	Free Format (Essay)

(D) **Bank**

Operations	X
OP002	

(E) Keywords: (≤9 characters)	1 Category	2 Topic 1	3 Topic 2	4 JTA	5 Setting	6 Other Objs.	7 Quiz Only	8 Retired
	RC	RECCONT						

(F) Point Value: 1

(G) Answer Time: ____ (Minutes)

(I) Review Date (YYMM): ____

(H) Cognitive Level: (Check one)

	1	Memory
	2	Comprehension
X	3	Application
	4	Analysis
	5	Problem Solving

(J) **QUESTION:**

Unit 1 is operating at 80% power with 76 Mlbm/hr core flow when a spurious feedwater flow signal causes a recirculation flow control runback. After the runback the following conditions exist:

- APRMs oscillating between 44% and 48% power
- Core flow is 42 Mlbm/hr
- Green lights are illuminated above RX RECIRC LIMITER 1 RUNBK RESET pushbutton.
- One (1) center region C-level LPRM upscale alarm is sealed in.
- Two (2) peripheral A-level LPRM downscale alarms are sealed in.

In accordance with ON-164-002, Recirc Drive Flow Instrument Failure, and the Power/Flow Map, which one of the following actions is required?

a. Raise core flow to at least 44 Mlbm/hr.

b. Place the reactor mode switch in SHUTDOWN.

c. Monitor for power instabilities and wait for RE instructions.

d. Insert control rods in accordance with the cram array to less than 40% power.

(K) **ANSWER:** a.

Procedure Change 7/3/2001

ON-178-002, Rev 1D Step 3.4

"Perform either 3.4.3.a or 3.4.3.b

Accept a and d.

Also per NDAP-QA-0338, Attachment F page 1 Cram Rods

E. W. Bowler

File No. R11-2

OPERATIONS QUESTION AND ANSWER INPUT FORM

(L) **REQUIRED MATERIALS:** Power to Flow Map (NDAP-0338)

(M) **K&A NUMBER/RATING:** 202001, A3.04/ 3.2

4(N) **NOTES:**

JUSTIFICATION:	The Green lights are illuminated above RX RECIRC LIMITER 1 RUNBK RESET pushbutton, indicating that the runback (caused by the spurious feedwater flow signal) can be reset, allowing flow to be raised.		
DISTRACTER B:	The plant is in the immediate exit region, an immediate shutdown is NOT required.		
DISTRACTER C:	You can't wait for RE.		
DISTRACTER D:	The procedure's first step is to raise recirc flow ~~so flo~~w should be done first and cram arrays ~~are no~~ longer used. C-4.		
EXAM OUTLINE CROSS-REF:	**LEVEL:**	RO	SRO
	TIER:	2	-
	GROUP:	2	-
K/A TEXT:	A3.02 – Ability to monitor automatic operations of the RECIRCULATION SYSTEM including: Lights and alarms.		
QUESTION SOURCE:	**BANK:**		
	MODIFIED:		
	NEW:	X	
10CFR55:	41(b).5, 41(b).6, 43(b).5		
COMMENTS:			

(O) **REFERENCES:** ON-164-002, Sect. 3.4, NDAP-QA-0338-10

(P) **POSITIONS:**
(check one or more boxes)

R – RO	S - SRO	A - ASO	N - NPO	T - STA
X	X			

(Q) Prepared by ED BOWLES

(R) Reviewed by: R.E. Cl

PROCEDURE COVER SHEET

PPL SUSQUEHANNA, LLC	NUCLEAR DEPARTMENT PROCEDURE	
CORE FLUX OSCILLATIONS		ON-178-002 Revision 10 Page 1 of 5
QUALITY CLASSIFICATION: (X) QA Program ( ) Non-QA Program	APPROVAL CLASSIFICATION: (X) Plant ( ) Non-Plant ( ) Instruction	

EFFECTIVE DATE: 07/03/01

PERIODIC REVIEW FREQUENCY: 2 Year

PERIODIC REVIEW DUE DATE: 9-30-03

RECOMMENDED REVIEWS:

Procedure Owner: Shift Technical Advisor-F Shift

Responsible Supervisor: Shift Supervisor-F Shift

Responsible FUM: Manager-Nuclear Operations

Responsible Approver: Manager-Nuclear Operations

CHECKED

b. Peak to peak oscillations trending towards $10 w/cm^2$ on LPRM's.

c. Two (2) or more LPRM upscale lights flashing and clearing on a one to five second period.

d. Two (2) or more LPRM downscale lights flashing and clearing on a one to five second period.

3.4 If either:

- Region II of Power/Flow Map entered with ≥ 50% of required LPRM upscale alarms operable, ________

OR

- Abnormal flux oscillations determined to be associated with plant systems (FW, EHC, RECIRC, etc.) less than scram limits specified in step 3.3 observed ________

PERFORM the following:

3.4.1 INITIATE TRA.

3.4.2 MONITOR APRM's and LPRM's for oscillations. ________

3.4.3 PERFORM following, as required, to rapidly suppress oscillations:

a. PROMPTLY INSERT control rods IAW RE Instructions in CRC Book to exit Region II. ________

OR

CAUTION (1)

WITH ONE REACTOR RECIRCULATION PUMP IN OPERATION RATED PUMP SPEED IS LIMITED TO ≤ 80% PER TRO 3.4.4.

CHECKED

CAUTION (2)

EXCEEDING THE CORE FLOW VALUE SPECIFIED IN THE CRC BOOK MAY CAUSE FUEL PRECONDITIONING LIMIT VIOLATIONS.

b. INCREASE the speed of the Operating Recirc PP(s) to exit the instability region, without exceeding the Core Flow Value in the RE Instructions in the CRC book. ______

3.4.4 If flux oscillations continue after performance of preceding steps, SCRAM Reactor IAW ON-100-101, Scram. ______

3.5 When conditions permit, NOTIFY Duty Manager and Reactor Engineering. ______

3.6 FORWARD completed copy of this procedure to following for review and retention:

3.6.1 Shift Supervisor ______ / ______
Signature Date

3.6.2 Nuclear Operations Supervisor-Shift Operations ______ / ______
Signature Date

3.6.3 Manager-Nuclear Operations ______ / ______
Signature Date

3.6.4 DCS Supervisor

4. REFERENCES

4.1 GE SIL 380, Rev. 1

4.2 NRC Bulletin 88-07

4.3 INPO SER 14-88

4.4 TS 3.4.1

4.5 BWROG March 18, 1992 Letter Re: Implementation Guidance for Stability Interim Corrective Actions

4.6 TRO 3.4.4

Unit ____ Sequence ____

SURVEILLANCE INSTRUCTIONS	Acceptable for Use QRE	Acceptable for Use STA/QRE
CONTROL VALVE TESTING SHALL BE PERFORMED AT POWER LEVELS LESS THAN OR EQUAL TO ______% RATED POWER. 1. Load reduction to be done with recirc flow. 2. Perform Control Valve Testing. 3. Restore power at ______MWE/hr. NOTE: All other Turbine Valve Testing can be performed at current power levels. (N/A if a ramp is in progress.)		
~~WEEKLY~~ CONTROL ROD EXERCISING PERMITTED AT ≤ _____% RATED POWER. EACH CONTROL ROD SHOULD BE INSERTED ONE NOTCH FIRST, THEN WITHDRAWN TO PREVIOUS POSITION.		
POWERPLEX is programmed assuming the following plant conditions: Recirc Loops in Operation ☐ DUAL ☐ SINGLE Bypass Valves ☐ OP ☐ INOP EOC RPT Instrumentation ☐ OP ☐ INOP Scram Time Dep MCPROL ☐ REALISTIC ☐ INT ☐ MAX Exposure Dep MCPROL ☐ BOC to __________ MWD/MT ☐ BOC to EOC		
SHUTDOWN/CRAM ARRAY INSTRUCTIONS	QRE	STA/QRE
SHUTDOWN INSTRUCTIONS 1. Use Emergency Power Reduction/Shutdown Instructions (Form NDAP-QA-0338-7) with shutdown control rod sequence. RWM shutdown control rod sequence is controlling. OR __________ Initials 2. Use startup sequence sheets in reverse order. RWM startup control rod sequence is controlling. (Document in "NOTES" column of startup sheets.) __________ Initials Sequence ______ Exposure __________ RWM pointer is at ______		
CRAM RODS ARE INCLUDED IN THE EMERGENCY POWER REDUCTION/SHUTDOWN INSTRUCTIONS LOCATED BEHIND SHUTDOWN TAB UTILIZING: CRAM Array / Shutdown Sequence Sheets		
UNIT OPERATION INSTRUCTIONS	QRE	STA/QRE
IF POWER ≥ DESIGN APRM ROD BLOCK LINE 1. Contact Reactor Engineer. 2. If R.E. cannot be reached, insert the following rod(s), as necessary to reduce power to below the APRM Rod Block line.		

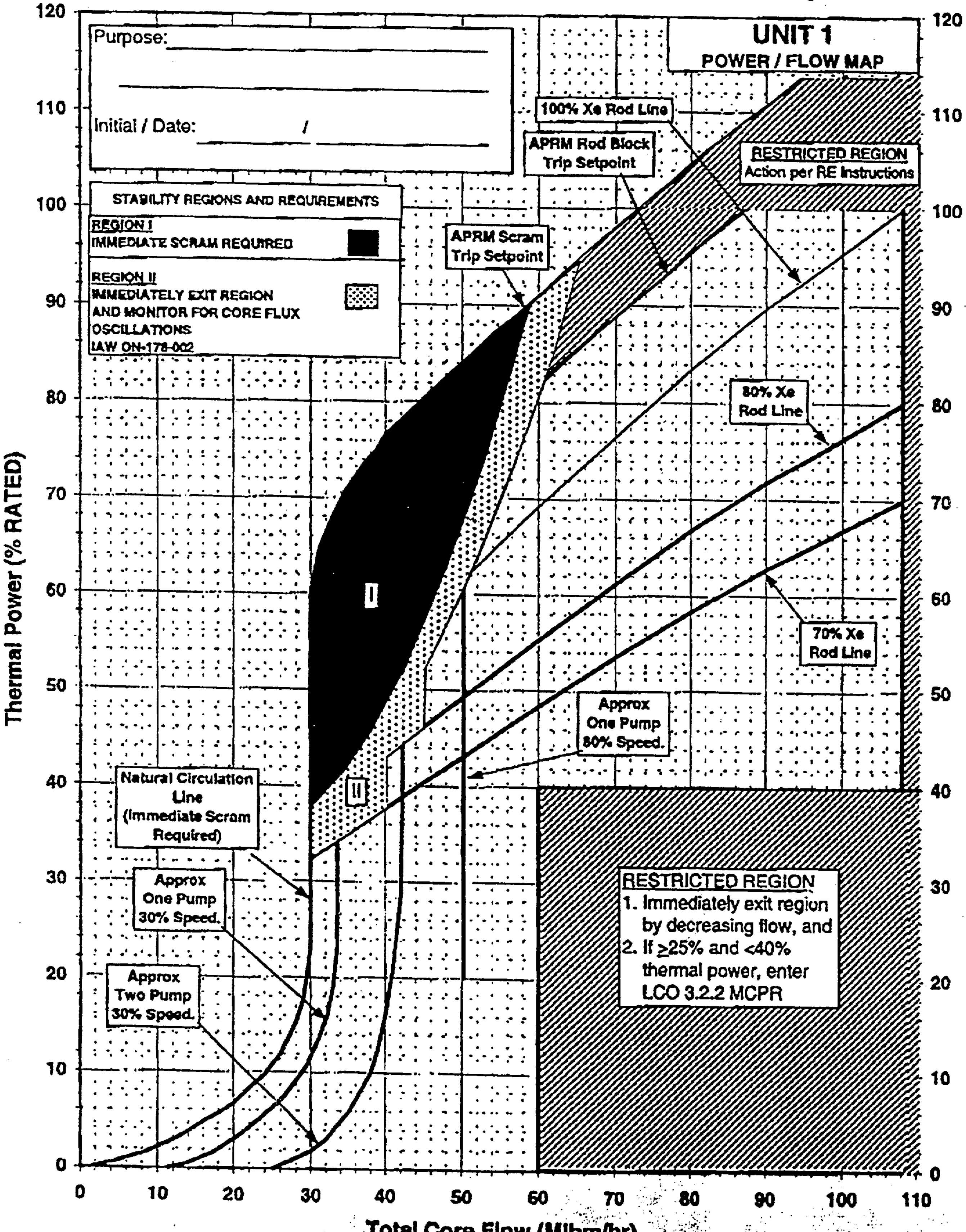
ROD	FROM	TO	PCO	PCO	DATE / TIME

Reselect and confirm previous moves: __________ / __________ / __________
PCO / PCO / DATE / TIME

PCAF # 2001-1213

PAGE 13 OF 16

UNIT ONE, POWER vs. FLOW MAP



PROCEDURE COVER SHEET

PPL SUSQUEHANNA, LLC	NUCLEAR DEPARTMENT PROCEDURE	
CORE FLUX OSCILLATIONS		ON-178-002 Revision 10 Page 1 of 5

QUALITY CLASSIFICATION:	APPROVAL CLASSIFICATION:
(X) QA Program ( ) Non-QA Program	(X) Plant ( ) Non-Plant ( ) Instruction

EFFECTIVE DATE: 07/03/01

PERIODIC REVIEW FREQUENCY: 2 Year

PERIODIC REVIEW DUE DATE: 9-30-03

RECOMMENDED REVIEWS:

Procedure Owner: Shift Technical Advisor-F Shift

Responsible Supervisor: Shift Supervisor-F Shift

Responsible FUM: Manager-Nuclear Operations

Responsible Approver: Manager-Nuclear Operations

1. SYMPTOMS AND OBSERVATIONS

1.1 DOWNSCALE LPRM alarms cycling on and off as indicated by annunciator or Full Core Display Status lights.

1.2 UPSCALE LPRM alarms cycling on and off as indicated by annunciator or Full Core Display Status lights.

1.3 APRM readings oscillating as indicated on SIP Panel 1C652.

1.4 LPRM readings in vicinity of a selected control rod oscillating as indicated on SIP Panel 1C652.

1.5 Inadvertent entry to Region II of Power/Flow Map.

2. AUTOMATIC ACTIONS

2.1 Possible intermittent APRM UPSCALE Rod Block Alarms.

2.2 Possible APRM UPSCALE Scram or Half-Scram.

3. OPERATOR ACTIONS

CHECKED

3.1 As time permits, RECORD date and time of event.

_______________________ Shift Supervision ____/____ Date Time

3.2 ENSURE non-peripheral rod selected to monitor LPRM's for oscillations. ________

3.3 IMMEDIATELY SCRAM Reactor IAW ON-100-101, Scram, per TS 3.4.1 if any following conditions observed: ________

3.3.1 Region I of Power/Flow Map entered.

3.3.2 Region II of Power/Flow Map entered, with less than 50% of required LPRM upscale alarms operable.

3.3.3 Any of following:

a. Peak to peak oscillations trending towards 10% on APRM's (oscillations measured from minimum peak to maximum peak).

CHECKED

b. Peak to peak oscillations trending towards $10 w/cm^2$ on LPRM's.

c. Two (2) or more LPRM upscale lights flashing and clearing on a one to five second period.

d. Two (2) or more LPRM downscale lights flashing and clearing on a one to five second period.

3.4 If either:

- Region II of Power/Flow Map entered with ≥ 50% of required LPRM upscale alarms operable, ________

OR

- Abnormal flux oscillations determined to be associated with plant systems (FW, EHC, RECIRC, etc.) less than scram limits specified in step 3.3 observed ________

PERFORM the following:

3.4.1 INITIATE TRA.

3.4.2 MONITOR APRM's and LPRM's for oscillations. ________

3.4.3 PERFORM following, as required, to rapidly suppress oscillations:

a. PROMPTLY INSERT control rods IAW RE Instructions in CRC Book to exit Region II. ________

OR

CAUTION (1)

WITH ONE REACTOR RECIRCULATION PUMP IN OPERATION RATED PUMP SPEED IS LIMITED TO ≤ 80% PER TRO 3.4.4.

CHECKED

CAUTION (2)

EXCEEDING THE CORE FLOW VALUE SPECIFIED IN THE CRC BOOK MAY CAUSE FUEL PRECONDITIONING LIMIT VIOLATIONS.

b. INCREASE the speed of the Operating Recirc PP(s) to exit the instability region, without exceeding the Core Flow Value in the RE Instructions in the CRC book. ________

3.4.4 If flux oscillations continue after performance of preceding steps, SCRAM Reactor IAW ON-100-101, Scram. ________

3.5 When conditions permit, NOTIFY Duty Manager and Reactor Engineering. ________

3.6 FORWARD completed copy of this procedure to following for review and retention:

3.6.1 Shift Supervisor ________ / ________
Signature Date

3.6.2 Nuclear Operations Supervisor-Shift Operations ________ / ________
Signature Date

3.6.3 Manager-Nuclear Operations ________ / ________
Signature Date

3.6.4 DCS Supervisor

4. REFERENCES

4.1 GE SIL 380, Rev. 1

4.2 NRC Bulletin 88-07

4.3 INPO SER 14-88

4.4 TS 3.4.1

4.5 BWROG March 18, 1992 Letter Re: Implementation Guidance for Stability Interim Corrective Actions

4.6 TRO 3.4.4

5. DISCUSSION

This procedure specifies actions required to reduce potential for fuel damage resulting from uncontrolled power oscillations, and is in compliance with Limiting Conditions of Operation as specified in TS 3.4.1.

If power oscillations occur and are not suppressed immediately, the MCPR Safety Limit may be violated.

The Reactor is most susceptible to power (flux) oscillations when operating in Regions I and II as identified on the Power/Flow Map. It is important to note that Regions I and II are not absolute with regards to preventing instabilities. As operating conditions approach Region II of the Power/Flow Map heightened awareness must be employed to ensure unstable operation does not occur or is mitigated. The instability regions may be approached during startup; shutdown; sequence exchanges; recirculation pump(s) trip(s) or runbacks; loss of feedwater heating events; inadvertent HPCI/RCIC injection etc. Thermal hydraulic instabilities may be occurring if any of the following is observed:

1. PEAK to PEAK oscillations are TRENDING TOWARDS, 10% on APRM's.
2. PEAK to PEAK oscillations are TRENDING TOWARDS, 10 w/cm^2 on LPRM's.
3. Two (2) or more LPRM UPSCALE lights cycling with one to five second period.
4. Two (2) or more LPRM DOWNSCALE lights cycling with one to five second period.

A Reactor scram must be initiated if any thermal hydraulic instability is confirmed, regardless if a limit (i.e., 10 w/cm^2 or 10% APRM) is actually exceeded.

Immediate Operator actions are required to suppress power oscillations which may lead to high local neutron flux levels without an automatic scram.

Region I, if entered, has a high probability of thermal hydraulic instabilities occurring. This region requires an immediate manual scram to prevent safety limits from being violated.

Region II, has a lower probability of thermal hydraulic instabilities occurring than Region I, although if entered still requires immediate action. This region must be immediately exited if entered.

Operation near the Region II boundary should be minimized to provide the largest margin to potential core instabilities.

PROCEDURE CHANGE PROCESS FORM

1. PCAF NO. 2000-5980 | 2. PAGE 1 OF 4 | 3. PROC. NO. ON-164-002 REV. 18

4. FORMS REVISED - ___ R___ , - ___ R___ , - ___ R___ , - ___ R___ , - ___ R___ , - ___ R___

5. PROCEDURE TITLE
Loss of Reactor Recirculation Flow

6. REQUESTED CHANGE
PERIODIC REVIEW ☒ NO ☐ YES
INCORPORATE PCAFS ☐ NO ☒ YES # 2000-3365 # ______ # ______ # ______
REVISION ☐ PCAF ☒ DELETION ☐ (CHECK ONE ONLY)

7. SUMMARY OF / REASON FOR CHANGE

Deleted the restriction on single loop operation imposed by PCAF 2000-3365. See attached pages 3 and 4.

Siemens Power Company has corrected analysis errors that have required the restriction on single loop operation. Revised Core Operating Limits Reports (COLRs) for both units have been issued.

Continued ☐

8. DETERMINE COMMITTEE REVIEW REQUIREMENTS
PORC REVIEW? (REQ'D FOR PLANT NDAP'S, SICT/E'S, AND FUP'S W/SAFETY EVALUATIONS) ☐ NO ☒ YES
ERC REVIEW? (REQ'D FOR QUALITY NON-PLANT NDAP'S) ☒ NO ☐ YES

9. PORC MTG# 00-12-15

10. ERC MTG# N/A

BLOCKS 11 THRU 14 ARE ON PAGE 2 OF FORM

15. David M. Kapuchinski 3529 / 12/15/00
PREPARER (Print or Type) ETN DATE

16 TRAINING REQUIRED? ☐ NO ☒ YES (TYPE) Hot Box 00-169

17. Grant Gensler 12/15/00
RESPONSIBLE SUPERVISOR DATE
SIGNATURE ATTESTS THAT RESPONSIBLE SUPERVISOR HAS CONDUCTED QADR AND TECHNICAL REVIEW UNLESS OTHERWISE DOCUMENTED IN BLOCK 14 OR ATTACHED REVIEW FORMS. CROSS DISCIPLINE REVIEW (IF REQUIRED) HAS BEEN COMPLETED BY SIGNATURE IN BLOCK 14 OR ATTACHED REVIEW FORMS.

18. [signature] 12/15/00
FUM APPROVAL DATE

19. RESPONSIBLE APPROVER ENTER N/A IF FUM HAS APPROVAL AUTHORITY
~~N/A~~ DMK 12/15/00 12/15/00
INITIALS DATE

RECEIVED
DEC 15 2000
DOCUMENT CONTROL SERVICES SSES

CONTROLLED

PROCEDURE CHANGE PROCESS FORM

1. PCAF NO. 2000-5980	2. PAGE 2 OF 4	3. PROC. NO. ON-164-002 REV. 18

11. A 50.59 and 72.48 Evaluation per NDAP-QA-0726 is required to be attached or referenced for all procedure changes except Expedited Reviews and Administrative Corrections. Either 11a, b, or c must be checked "YES" and the appropriate form attached or referenced.

a. 50.59 and 72.48 Screening Determination (Form NDAP-QA-0726-5) ☒ YES ☐ N/A

b. 50.59 or 72.48 Safety Evaluation (Note: 50.59 Safety Evaluations prepared on Form NDAP-QA-0726-1 Rev. 5 or earlier also require a 50.59 & 72.48 Screening Determination) ☐ YES ☒ N/A

Safety Evaluation No. ______________________

c. Expedited Review/Administrative Correction- 50.59 and 72.48 Evaluation not Required ☐ YES ☒ N/A

12. Is a Surveillance Procedure Review Checklist required per NDAP-QA-0722? ☐ YES ☒ NO

13. Is a Special, Infrequent or Complex Test/Evolution Analysis Form required per NDAP-QA-0320? (SICT/E form does not need to be attached.) ☐ YES ☒ NO

14. Reviews may be documented below or by attaching Document Review Forms NDAP-QA-0101-1.

REVIEW	REVIEWED BY WITH NO COMMENTS	DATE
QADR TECHNICAL REVIEW	[signature]	12/15/00
REACTOR ENGINEERING/NUCLEAR FUELS *	[signature]	12/14/00
IST **		
OPERATIONS		
NUCLEAR SYSTEMS ENGINEERING		
NUCLEAR MODIFICATIONS		
MAINTENANCE		
HEALTH PHYSICS		
NUCLEAR TECHNOLOGY		
CHEMISTRY		
OTHER ______________		

* Required for changes that affect, or have potential for affecting core reactivity, nuclear fuel, core power level indication or impact the thermal power heat balance. (58)

** Required for changes to Section XI Inservice Test Acceptance Criteria.

PROCEDURE CHANGE PROCESS FORM

1. PCAF NO. 2000- 5503	2. PAGE 1 OF 3	3. PROC. NO. ON-164-002 REV. 18

4. FORMS REVISED - ___ R___ , - ___ R___ , - ___ R___ , - ___ R___ , - ___ R___ , - ___ R___

5. PROCEDURE TITLE
LOSS OF REACTOR RECIRCULATION FLOW

6. REQUESTED CHANGE

PERIODIC REVIEW ☒ NO ☐ YES

INCORPORATE PCAFS ☒ NO ☐ YES # ________ # ________ # ________ # ________

REVISION ☐ PCAF ☒ DELETION ☐ (CHECK ONE ONLY)

7. SUMMARY OF / REASON FOR CHANGE
Admin change. This change is made in fulfillment of CRA 264438 and provides information on pump operating characteristics only. This information is already incorporated as an attachment to the recirc pump operating procedure and is being restated at the appropriate times within the body of this procedure. The PCAF incorporates an NSE recommendation to alternatively state that operating a reactor recirc pump near the 30% limiter equates to an approximate pump speed of 500 RPM. The pump "high oscillation" range (460 to 485 RPM) should be avoided when operating near the 30% limiter.

Continued ☐

8. DETERMINE COMMITTEE REVIEW REQUIREMENTS

PORC REVIEW? (REQ'D FOR PLANT NDAP'S, SICT/E'S, AND FUP'S W/SAFETY EVALUATIONS) ☒ NO ☐ *YES*

ERC REVIEW? (REQ'D FOR NON-PLANT NDAP'S) ☒ NO ☐ *YES*

9. PORC MTG# NA

10. ERC MTG# NA

BLOCKS 11 THRU 14 ARE ON PAGE 2 OF FORM

15. Eric Miller — 3321 / 7/21/00
PREPARER (Print or Type) — ETN — DATE

16 TRAINING REQUIRED? ☒ NO ☐ YES (TYPE) NA

17. [signature] 7/21/00
RESPONSIBLE SUPERVISOR — DATE

SIGNATURE ATTESTS THAT RESPONSIBLE SUPERVISOR HAS CONDUCTED QADR AND TECHNICAL REVIEW UNLESS OTHERWISE DOCUMENTED IN BLOCK 14 OR ATTACHED REVIEW FORMS. CROSS DISCIPLINE REVIEW (IF REQUIRED) HAS BEEN COMPLETED BY SIGNATURE IN BLOCK 14 OR ATTACHED REVIEW FORMS.

18. NA
FUM APPROVAL — DATE

19. RESPONSIBLE APPROVER — ENTER N/A IF FUM HAS APPROVAL AUTHORITY

NA
INITIALS — DATE

CONTROLLED

RECEIVED
JUL 24 2000
DOCUMENT CONTROL SERVICES - SSES

PROCEDURE CHANGE PROCESS FORM

1. PCAF NO. 2000-5503	2. PAGE 2 OF 3	3. PROC. NO. ON-164-002 REV. 18

11. A 50.59 and 72.48 Evaluation per NDAP-QA-0726 is required to be attached or referenced for all procedure changes except Expedited Reviews and Administrative Corrections. Either 11a, b, or c must be checked "YES" and the appropriate form attached or referenced.

   a. 50.59 and 72.48 Screening Determination (Form NDAP-QA-0726-5) ☐ YES ☒ N/A

   b. 50.59 or 72.48 Safety Evaluation (Note: 50.59 Safety Evaluations prepared on Form NDAP-QA-0726-1 Rev. 5 or earlier also require a 50.59 & 72.48 Screening Determination) ☐ YES ☒ N/A

   Safety Evaluation No. ____________

   c. Expedited Review/Administrative Correction- 50.59 and 72.48 Evaluation not Required ☒ YES ☐ N/A

12. Is a Surveillance Procedure Review Checklist required per NDAP-QA-0722? ☐ YES ☒ NO

13. Is a Special, Infrequent or Complex Test/Evolution Analysis Form required per NDAP-QA-0320? (SICT/E form does not need to be attached.) ☐ YES ☒ NO

14. Reviews may be documented below or by attaching Document Review Forms NDAP-QA-0101-1.

REVIEW	REVIEWED BY WITH NO COMMENTS	DATE
QADR		
TECHNICAL REVIEW	J Rinehimer	07-21-00
REACTOR ENGINEERING/NUCLEAR FUELS *		
IST **		
OPERATIONS		
NUCLEAR SYSTEMS ENGINEERING		
NUCLEAR MODIFICATIONS		
MAINTENANCE		
HEALTH PHYSICS		
NUCLEAR TECHNOLOGY		
CHEMISTRY		
OTHER ____________		

* Required for changes that affect, or have potential for affecting core reactivity, nuclear fuel, core power level indication or impact the thermal power heat balance. ([58])

** Required for changes to Section XI Inservice Test Acceptance Criteria.

PROCEDURE COVER SHEET

	NUCLEAR DEPARTMENT PROCEDURE	
SUSQUEHANNA STEAM ELECTRIC STATION PP&L	LOSS OF REACTOR RECIRCULATION FLOW	ON-164-002 Revision 18 Page 1 of 9

QUALITY CLASSIFICATION:
(X) QA Program ( ) Non-QA Program

APPROVAL CLASSIFICATION:
(X) Plant ( ) Non-Plant
( ) Instruction

EFFECTIVE DATE: 10/27/99

PERIODIC REVIEW FREQUENCY: 2 Years

PERIODIC REVIEW DUE DATE: 9-30-01

RECOMMENDED REVIEWS:

Procedure Owner: Dayne R. Brophy

Responsible Supervisor: Grant Fernsler

Responsible FUM: Manager-Nuclear Operations

Responsible Approver: General Manager-SSES

1. SYMPTOMS AND OBSERVATIONS

1.1 Reactor power begins to decrease immediately.

1.2 Reactor vessel water level increases.

1.3 Any following indication on Standby Information Panel 1C652:

1.3.1 Recirculation loop flow(s) decreases in affected loop(s).

1.3.2 Jet pump flow(s) for affected loop(s) and total jet pump flow decreases.

1.4 Any following annunciator on Operating Unit Benchboard 1C651:

1.4.1 RECIRC M-G GEN A(B) LOCKOUT TRIP.

1.4.2 RECIRC M-G A(B) DRIVE MTR TRIP.

1.4.3 RECIRC PUMP A(B) HI PRESS/LO LEVEL TRIP.

1.4.4 RPT SYS LOOP A(B) TRIP.

1.4.5 RECIRC A(B) FLOW LIMIT RUNBACK.

2. AUTOMATIC ACTIONS

2.1 Possible main turbine trip and reactor scram because of RPV level swell caused by the trip of both Reactor Recirculation Pumps, depending on plant operating conditions.

2.2 Reduction in Reactor power, core flow, steam flow, feedwater flow, and generator output corresponding to recirc runback flow decrease.

3. OPERATOR ACTIONS

CHECKED

3.1 RECORD date and time of event.

____________________________ ____/____
Shift Supervision Date Time

3.2 If <u>both</u> Recirculation Pumps trip, IMMEDIATELY SCRAM reactor. ________

CHECKED

3.3 If <u>one</u> Reactor Recirculation Pump trips:

NOTE: If jet pump flow in operating loop is < 38 $\times 10^6$ lbm/hr computer generated core flow is not correct.

3.3.1 If jet pump flow in operating loop is < 38 $\times 10^6$ lbm/hr, ADD idle and operating loop flows together to determine actual core flow. ______

3.3.2 PLOT position on Power/Flow Map, Form NDAP-QA-0338-10. ______

3.3.3 PERFORM appropriate action as specified on Power/Flow Map. ______

3.3.4 ENSURE thermal power REDUCED to < 70% rod line. ______

3.3.5 REDUCE operating pump speed to 80% rated pump speed (80% = 1344 rpm) in accordance with OP-164-001. ______

3.3.6 COMPLY with COLR Section 8.0 Limits in TRM. ______

3.3.7 COMPLY with Tech Spec LCOs 3.4.1. ______

<u>CAUTION</u>

THERMAL BINDING AND/OR PRESSURE LOCKING OF RECIRCULATION ISOLATION VALVES MIGHT OCCUR IF CLOSED FOR MORE THAN APPROXIMATELY 5 MINUTES.

3.3.8 For stopped pump, PLACE RECIRC A(B) MOV OL BYPS HV-143-F031A(B)/F032A(B) key switch to TEST position. ______

3.3.9 ENSURE RECIRC PUMP A(B) DSCH BYPS HV-143-F032A(B) OPEN. ______

3.3.10 CLOSE RECIRC PUMP A(B) DSCH HV-143-F031A(B). ______

3.3.11 Within 5 minutes, OPEN RECIRC PUMP A(B) DSCH HV-143-F031A(B). ______

3.3.12 After 2 minutes, PLACE RECIRC A(B) MOV OL BYPS HV-143-F031A(B)/F032A(B) key switch to NORM position. ______

2000 - 5980
1 of 4

CHECKED

3.3.13 Prior to restart of pump, NOTIFY Duty Reactor Engineer to perform an evaluation of core thermal limits and preconditioning. ________

CAUTION

DO NOT ATTEMPT TO RESTART RECIRC PUMP IF OPERATING ABOVE 70% ROD LINE OR IF ANY FLUX OSCILLATIONS ARE OBSERVED.

3.3.14 When cause of trip corrected, RESTART Reactor Recirculation Pump in accordance with OP-164-001 Reactor Recirculation System. ________

3.3.15 If pump will be out of service > 1 hour, COMPLY with GO-100-009 Single Recirculation Loop Operation. ________

3.4 In the event of Reactor Recirculation Pump runback:

3.4.1 PLOT position on Power/Flow Map, Form NDAP-QA-0338-10. ________

3.4.2 PERFORM appropriate action specified on Power/Flow Map. ________

3.4.3 DETERMINE which limiter initiated runback:

a. Limiter #1 (30%) limiting by Green light illuminated above RX RECIRC LIMITER 1 RUNBK RESET HS-B31-1S15A(B) pushbutton. ________

b. Limiter #2 (45%) limiting by Green light illuminated above LOSS OF FW PP RUNBK RESET HS-B31-1S12A(B) pushbutton. ________

3.4.4 ENSURE both pumps run back to value associated with controlling limiter. ________

CHECKED

3.4.5 OBSERVE following plant parameters WITHIN LIMITS corresponding to new power level: ______

a. Power to flow limits

b. Condenser vacuum

c. Feedwater flow/steam flow

d. RPV water level

3.4.6 DETERMINE signal that initiated runback from following:

a. Limiter #1 (30%) runback initiated by any following condition: ______

(1) Total feedwater flow ≤ 20% for > 15 seconds. ______

(2) RECIRC PUMP A(B) DSCH HV-143-F031A(B) not fully open. ______

(3) RPV water level < level 3. ______

b. Limiter #2 (45%) runback initiated by: ______

(1) Any Circulating Water Pump protective trip. ______

(2) RPV low water level (+ 30") and any of following:

(a) Feedwater flow A, B, or C decrease to ≤ 20%. ______

(b) Any Condensate Pump discharge pressure ≤ 100 psig. ______

CHECKED

(c) Auto isolation of Feedwater Heaters String A, B or C due to high level in Feedwater Heaters 1 or 2. ______

3.4.7 ENSURE REACTOR RECIRC PUMP A(B) SPEEDS SY-B31-1R621A(B) IN MANUAL.

3.4.8 For Limiter #1 runback PERFORM following for one or both pumps as required:

CAUTION(1)

WHEN ESTABLISHING CONTROL WITH THE RECIRC PUMP SPEED CONTROLLERS, MINIMIZE LOWERING CORE FLOW TO AVOID INADVERTENT ENTRY INTO REGIONS I OR II OF THE POWER/FLOW MAP.

CAUTION (2)

WHEN ESTABLISHING CONTROL WITH THE RECIRC PUMP SPEED CONTROLLERS, PUMP SPEED SHOULD BE MAINTAINED AT APPX. 500 RPM. SPEED OSCILLATIONS ARE POSSIBLE WHEN THE PUMP IS OPERATED BETWEEN 460 TO 485 RPM.

a. To prevent pump speed from changing when Limiter #1 reset, ENSURE GEN 1A(1B) DEMAND adjusted such that GEN 1A(1B) SPEED decreases when controller DEMAND is decreased. ______

b. DEPRESS RX RECIRC LIMITER 1 RUNBK RESET HS-B31-1S15A(B) pushbutton. ______

c. MONITOR GEN 1A(1B) SPEED SI-14032A(B). ______

d. If speed increases rapidly, TRIP scoop tube on affected generator by depressing SCOOP TUBE A(B) LOCK OR RESET HS-B31-1S03A(B) TRIP pushbutton. ______

e. If previously illuminated, OBSERVE Green light above RX RECIRC LIMITER 1 RUNBK RESET HS-B31-1S15A(B) pushbutton EXTINGUISHED. ______

CHECKED

3.4.9 For Limiter #2 runback PERFORM following for one or both pumps as required:

CAUTION

WHEN ESTABLISHING CONTROL WITH THE RECIRC PUMP SPEED CONTROLLERS, MINIMIZE LOWERING CORE FLOW TO AVOID INADVERTENT ENTRY INTO REGIONS I OR II OF THE POWER/FLOW MAP.

a. To prevent pump speed from changing when Limiter #2 reset, ENSURE GEN 1A(1B) DEMAND adjusted such that GEN 1A(1B) SPEED decreases when controller DEMAND is decreased. ________

b. DEPRESS RECIRC A(B) LOSS OF FW PP RUNBK RESET HS-B31-1S12A(B) pushbutton. ________

c. MONITOR GEN 1A(1B) SPEED SI-14032A(B). ________

d. If speed increases rapidly, TRIP scoop tube on affected generator by depressing SCOOP TUBE A(B) LOCK OR RESET HS-B31-1S03A(B) TRIP pushbutton. ________

e. OBSERVE Green light above RECIRC A(B) LOSS OF FW PP RUNBK RESET HS-B31-1S12A(B) pushbutton EXTINGUISHED. ________

3.4.10 CHECK RECIRC A(B) FLOW LIMIT RUNBACK annunciator CLEARED. ________

3.4.11 NOTIFY Reactor Engineering. ________

3.5 FORWARD completed copy of this procedure to following for review and retention:

3.5.1 Shift Supervisor ______________ / ________
Signature Date

3.5.2 Operations Supervisor-Nuclear ______________ / ________
Signature Date

3.5.3 Manager-Nuclear Operations ______________ / ________
Signature Date

3.5.4 DCS Supervisor

4. REFERENCES

4.1 FSAR Section 5.4.1 Reactor Recirculation Pumps

4.2 FSAR Section 15.3 Decrease in Reactor Coolant Flowrate

4.3 GE SIL No. 380

4.4 M-143 Reactor Recirculation

4.5 Memo PLI-42281, S.A. Somma to A.M. Price, "Indicated Core Flow Anomaly," October 1, 1985

4.6 NRC Bulletin 88-07 Supplement 1

4.7 OP-164-001, Reactor Recirculation System

5. DISCUSSION

Loss of Reactor recirculation flow can be caused by the unexpected tripping of one or both Reactor Recirculation Pumps. Tripping of both pumps has the greatest impact on plant operation. At high power levels, the Main Turbine may trip automatically because of RPV water level swell and result in a Reactor scram. If the Reactor does not scram automatically following trip of both Reactor Recirculation Pumps, the reactor is immediately scrammed manually to avoid potential for core flux oscillations.

The Reactor may be operated at reduced power with one Reactor Recirculation Pump out of service and the other driving half of the jet pumps. The idle Reactor Recirculation Pump is kept hot by reverse flow through the loop and is not started unless the idle loop temperature is within 50°F of the operating loop temperature, and the Reactor is operating below the 70% rod line.

Loss of one Reactor Recirculation Pump during Reactor operation does not initiate any Reactor Protection System or safeguards systems actuation because fuel thermal margins are maintained. Flow in the idle jet pumps reverses in approximately 6 seconds, and flow in the operating jet pumps increases to about 143% of its normal flow if the operating pump is at 84% pump speed. If the jet pump flow in the operating loop is less than 38×10^6 lbm/hr, then flow through the idle jet pumps is not reverse flow. In this case the total core flow logic of automatically subtracting loop flows is not correct.

If total core flow decreases into Region I of Power/Flow Map, the Reactor is manually scrammed to avoid potential for core flux oscillations. The operator reduces flow in the operating loop to single pump flow criteria. Single pump flow criteria is based on reducing the operating pump speed to less than 80% rated pump speed (80% = 1344 rpm), in accordance with OP-164-001.

Automatic trips of Reactor Recirculation Pumps can come from EOC-RPT trip (CV fast closure, ≥ 500 psig; SV closure, ≤ 5.5% closed) or ATWS (RPV low level 2, Reactor high pressure). The ATWS and EOC RPT trips open the RPT breakers which in turn cause the MG set drive motor to trip. A LPCI Initiation signal with low Reactor pressure (< 236 psig) trips the Reactor Recirculation Pumps by closing the discharge and discharge bypass valves.

Two speed limiters limit recirc flow by limiting generator speed on the Reactor Recirc MG sets. Limiter #1 limits speed to 30% and provides NPSH protection for the recirc pumps. Limiter #2 limits speed to 45% and prevents spurious scrams due to transients on the Condensate/Feedwater System and Circulating Water System.

(18)

SSES 2001 NRC Initial Written Examination
Post-Examination Comments

Applicant Level: ☑ RO ☐ SRO

Applicant Name:

Question Type: ☐ Common ☐ RO only ☐ SRO only

Question #: RO 69 (enter number, if SRO only enter N/A)
SRO 53 (enter number, if RO only enter N/A)

Answer: (circle the answer key response) A B C D

Reference: (enter the answer key reference below)

Comment: (enter the comment below)
"New Band" of +90 to +100" is also the "established band" prior to event. ∴ B and C both correct.

Recommendation: (The grader is encouraged to discuss the matter with the NRC Chief Examiner before proceeding with the grading)
☐ Change the correct answer. ☑ Do NOT change the correct answer.
☐ Accept two correct answers. ☐ Delete the question
☑ Make clarifications to the question.

Changes / clarifications made to examination: (provide a description)

Reference(s) to support change / clarification made to examination:

Justification for rejection of an applicant's comment:
this is NOT a new band.

Proctor: ☐ Change made in INK on the master examination copy
Edwin W. Bowles
Print Name
Edwin W. Bowles 8/10/2001
Signature Date

[signature] 08/14/01

OPERATIONS QUESTION AND ANSWER INPUT FORM

RO 69 SRO ~~70~~ 53

(A) SY017 C-1 (B) ??

Course Objective

(C) Question Type (check one)

X	Multiple Choice
	Matching
	Free Format (Essay)

(D) Bank

Operations	X
OP002	

(E)	1	2	3	4	5	6	7	8
Keywords :	Category	Topic 1	Topic 2	JTA	Setting	Other Objs.	Quiz Only	Retired
(≤9 characters)	RHR SDC	SY-017 C-1			C			

(F) Point Value: 1 (G)Answer Time: (Minutes)

(I) Review Date (YYMM):

(H) Cognitive Level: (Check one)

X	1	Memory
	2	Comprehension
	3	Application
	4	Analysis
	5	Problem Solving

(J) QUESTION:

A shutdown and cooldown is in progress on Unit 1. Per OP-149-002, RHR Shutdown Cooling, the required level band is established then Shutdown Cooling (SDC) is placed into service with one Reactor Recirculation (RR) loop shutdown.

One (1) hour after establishing SDC, the operating RR pump trips. Which one of the following describes the action to be taken for reactor water level?

Adjust reactor water level...

a. to a new band of +35 to +50 inches.

b. to a new band of +90 to +100 inches.

c. maintaining level within the band established before the event.

d. maintaining level above the band established before the event.

(K) ANSWER: c.

Two correct Answers were Not accepted recommend changing response b to +60 to +80 inches to avoid any confusion

Form STCP-QA-325D
Rev. 3, (8/95)
Page 1 of 1

File No. R11-2

(L) <u>REQUIRED MATERIALS:</u> None

(M) <u>K&A NUMBER/RATING:</u> 205000 / K6.03 / 3.1 / 3.2

4(N) <u>NOTES:</u>

JUSTIFICATION:	A level of +90 to +100 inches is established before starting SDC and is maintained throughout SDC operations – with RR flow and without RR flow.			
DISTRACTER A:	+90 to +100 inches is established.			
DISTRACTER B:	Once +90 to +100 inches is established before starting SDC, this level is maintained throughout SDC operations – with RR flow and without RR flow.			
DISTRACTER D:	+90 to +100 inches is established before establishing SDC and is maintained throughout SDC operations.			
EXAM OUTLINE CROSS-REF:	LEVEL:	RO	SRO	
	TIER:	2	2	
	GROUP:	2	2	
	K6.03 - Knowledge of the effect that a loss or malfunction of the following will have on the SHUTDOWN COOLING SYSTEM: Recirculation System.			
QUESTION SOURCE:	BANK:			
	MODIFIED:	X		
	NEW:			
10CFR55:	41(b)(10)			
COMMENTS:	The original question asked for the required reactor water level to secure all RR flow and why. The question was changed to the action to be taken for reactor water level if the operating RR pump trips once the conditions of OP-149-002 are established. The answer and all distracters changed.			

(O) <u>REFERENCES:</u>
OP-149-002, 3.1.1, 3.1.2

(P) <u>POSITIONS:</u>
(check one or more boxes)

R – RO	S - SRO	A - ASO	N - NPO	T - STA
X	X			X

(Q) Prepared by Phil Ballard

(R) Reviewed by: R.E. Cl...

#11

SSES 2001 NRC Initial Written Examination
Post-Examination Comments

* Identified during exam review by licensed class Instructor.

Post Examination Review

Applicant Level: ☐ RO ☐ SRO

Applicant Name:

Question Type: ☑ Common ☐ RO only ☐ SRO only

Question #: RO 92 (enter number, if SRO only enter N/A)
SRO 68 (enter number, if RO only enter N/A)

Answer: (circle the answer key response) A B C D

Reference: (enter the answer key reference below)
GO-100-002, Sect 617 And SO-100-011

Comment: (enter the comment below)
SO-100-011 Cautions operators to limit heat up rate to 25°F/15min. the question asks for a 45 min change so the limit would be 75°F - the correct ans 314°F is less than the lowest temperature given as a response

Recommendation: (The grader is encouraged to discuss the matter with the NRC Chief Examiner before proceeding with the grading)
☐ Change the correct answer. ☐ Do NOT change the correct answer.
☐ Accept two correct answers. ☑ Delete the question
☐ Make clarifications to the question.

Changes / clarifications made to examination: (provide a description)
Delete Question RO 92 And SRO 68

Reference(s) to support change / clarification made to examination:
SO-100-011, step 6.1.3 And Att A

Justification for rejection of an applicant's comment:

Proctor: ☑ Change made in INK on the master examination copy
Edwin W. Bowles
Print Name
Edwin W Bowles 8/10/2001
Signature Date
08/14/01

RO 92 SRO ~~89~~ 68

(A) SY017 J-1 (B) 10.e
Course Objective

(C) Question Type (check one)
[X] Multiple Choice
[ ] Matching
[ ] Free Format (Essay)

(D) Bank
Operations [X]
OP002 [ ]

(E)

	1	2	3	4	5	6	7	8
Keywords :	Category	Topic 1	Topic 2	JTA	Setting	Other Objs.	Quiz Only	Retired
(≤9 characters)	Systems	RPV						

(F) Point Value: 1 (G)Answer Time: (Minutes)

(H) Cognitive Level: (Check one)
[ ] 1 Memory
[ ] 2 Comprehension
[ ] 3 Application
[X] 4 Analysis
[ ] 5 Problem Solving

(I) Review Date (YYMM):

(J) QUESTION:

During a reactor heatup the following temperature readings are recorded on Attachment A of SO-100-011, Reactor Vessel Temperature and Pressure Recording:

- 0800 - 242°F
- 0815 - 263°F
- 0817 - Startup was temporarily halted
- 0830 - 239°F
- 0845 - 268°F
- 0900 - 311°F

Per GO-100-002, which one of the following is the maximum allowable temperature at 0915?

a 329°F

b 339°F

c. 353°F

d 363°F

(K) ANSWER: a.

Question should be deleted
- No Correct Answer -
Per SO-100-011, step 6.1.3
Caution and Attachment A (pg 16)
All ΔTs should be maintained < 25°
in any 25°f, 15 min period
Correct answer would be 314°F
239°f + 75°f = 314°f

E. W. Bank

(L) REQUIRED MATERIALS: None

(M) K&A NUMBER/RATING: PWG, 2.1.32/3.4

(N) NOTES:

JUSTIFICATION:	Temperature cannot raise more than 90°F per hour. Because heatup rate was allowed to lower (0830) and temperature to lower to 239°F the temperature cannot exceed 329°F during the next hour.		
DISTRACTER B:	This would exceed 90°F/hr from 0830.		
DISTRACTER C:	This would exceed 90°F/hr from 0830.		
DISTRACTER D:	This would exceed 90°F/hr from 0830.		
EXAM OUTLINE CROSS-REF:	LEVEL:	RO	SRO
	TIER:	3	3
	GROUP:	-	-
K/A TEXT:	2.1.32 – Ability to explain and apply system limits and precautions.		
QUESTION SOURCE:	BANK:		
	MODIFIED:	X	
	NEW:		
10CFR55:	41(b).10, 43(b).5		
COMMENTS:			

(O) REFERENCES: GO-100-002, Sect ~~6.42~~ 6.17 and SO-100-011, pg. 6

(P) POSITIONS:
(check one or more boxes)

R – RO	S - SRO	A - ASO	N - NPO	T - STA
X	X			

(Q) Prepared by ED BOWLES

(R) Reviewed by: R.E. Cli

PROCEDURE COVER SHEET

SUSQUEHANNA STEAM ELECTRIC STATION PP&L	NUCLEAR DEPARTMENT PROCEDURE REACTOR VESSEL TEMPERATURE AND PRESSURE RECORDING	SO-100-011 Revision 12 Page 1 of 17

QUALITY CLASSIFICATION:

(X) QA Program ( ) Non-QA Program

APPROVAL CLASSIFICATION:

(X) Plant ( ) Non-Plant

( ) Instruction

EFFECTIVE DATE: 12-31-98

PERIODIC REVIEW FREQUENCY: N/A

PERIODIC REVIEW DUE DATE: N/A

RECOMMENDED REVIEWS:

Procedure Owner: Jay Barnes

Responsible Supervisor: David T. Walsh

Responsible FUM: Manager-Nuclear Operations

Responsible Approver: General Manager-SSES

6.1.2 On Reactor Coolant System Temperature and Pressure Log (Attachment A), RECORD following information every 15 minutes:

* a. Recirc loop A temperature
* b. Recirc loop B temperature
* c. Reactor Vessel Bottom Head Drain Temperature
* d. Reactor Vessel Pressure
* e. Rx Steam Dome Temperature (Applicable only when Rx Coolant Temperature > 212°F)

CAUTION

IF 15 MINUTE TEMPERATURE CHANGE > 25°F, ACTION SHOULD BE TAKEN TO REDUCE HEATUP/COOLDOWN RATE. CONTINUING HEATUP OR COOLDOWN AT THIS RATE WILL LEAD TO A TS VIOLATION.

NOTE: Calculated temperature change is change in temperature that occurred in previous 15 minutes.

6.1.3 CALCULATE temperature changes for following and RECORD on Reactor Coolant System Temperature and Pressure Log (Attachment A).

* a. Recirc loop A
* b. Recirc loop B
* c. Reactor Vessel Bottom Head Drain
* d. Rx Steam Dome Temperature (Applicable only when Rx Coolant Temperature > 212°F)

* (1) 6.1.4 On Reactor Coolant System Temperature and Pressure Log (Attachment A), CONFIRM compliance with SR 3.4.10.1 once every 30 minutes (refer to TS Figure 3.4.10-1 Minimum Reactor Vessel Metal Temperature vs. Reactor Vessel Pressure, Attachment B) by verifying following applicable statements for Heatup and Cooldown Events:

* a. Reactor Vessel Pressure and Temperature are to right of curve C.

REACTOR COOLANT SYSTEM TEMPERATURE AND PRESSURE LOG

All Rx Coolant Temp and Press Data shall be recorded until Heatup, Cooldown or Inservice Leak and Hydrostatic testing is complete. TS Required Actions should only be entered if Rx Steam Dome Temperature ΔT's are > 100°F in any one hour. However, ALL ΔT's should be maintained <25° in any 15 minute period.

CIRCLE DATA SOURCE WHERE APPLICABLE	PI-C32-1R605 NFP02 NFP03	TR-B31-1R650 NRT01 NRT51 NRT52	NA	TR-B31-1R650 NRT02 NRT53 NRT54	NA	TR-B21-1R006 (PT. 8) NLT01	NA	NFA05	NA	
TIME/DATE AS NECESSARY	REACTOR PRESSURE PSIG	RECIRC LOOP A TEMP °F	RECIRC LOOP A DELTA TEMP °F	RECIRC LOOP B TEMP °F	RECIRC LOOP B DELTA TEMP °F	BOTTOM HEAD DRAIN TEMP °F	BTM HD DRAIN DELTA TEMP °F	RX STM DOME TEMP °F	RX STM DOME DELTA TEMP °F	CONFIRM COMPLIANCE W/SR 3 4 10 1

Shiftly Review and Confirmation above recorded data is accurate, compliant, and complete.

SHIFT SUPERVISION DATE Time

REACTOR VESSEL FLANGE AND TOP HEAD FLANGE TEMP LOG

Rx Vessel Flange and Rx Vessel Top Head Flange Temp need only be recorded when in Mode 4 with Rx Vessel head bolting studs under tension and coolant temp ≤100°F or at least 30 minutes during tensioning c Rx Vessel head bolting studs.

	TR-B21-1R006 (POINT #4)	TR-B21-1R006 (POINT #2)	
TIME/DATE AS NECESSARY	RX VESSEL FLANGE TEMP LOCAL 1C007	RX VESSEL TOP HEAD FLANGE TEMP LOCAL 1C007	CONFIRM COMPLIANC W/SR 3 4 10 or 3 4 10 8, o 3 4 10 9

#13

SSES 2001 NRC Initial Written Examination
Post-Examination Comments

Applicant Level: ☑ RO ☑ SRO

Applicant Name:

Question Type: ☑ Common ☐ RO only ☐ SRO only

Question #: RO 95 (enter number, if SRO only enter N/A)
SRO 70 (enter number, if RO only enter N/A)

Answer: (circle the answer key response) A B C D

Reference: (enter the answer key reference below)

Comment: (enter the comment below) *
"A" could also be correct because Steam flow to HPCI initially lowers MWE. STEM does not ask final Steady-State conditions

Recommendation: (The grader is encouraged to discuss the matter with the NRC Chief Examiner before proceeding with the grading)
☐ Change the correct answer. ☐ Do NOT change the correct answer.
☑ Accept two correct answers. ☐ Delete the question
☑ Make clarifications to the question.

Changes / clarifications made to examination: (provide a description)
Generator MWE will lower initially - accept 2 answers A & C

Reference(s) to support change / clarification made to examination:
Transient Analysis SC007 And Simulator Traces

Justification for rejection of an applicant's comment:

Proctor: ☑ Change made in INK on the master examination copy

Edwin W. Bowles
Print Name
Edwin W. Bowles 8/10/2001
Signature Date

[signature] 08/14/01

* this possibility was also mentioned by a Licensed Instructor during the exam review.

RO 95 SRO ~~93~~ 70

(A) AD045 — Course (B) — Objective

(C) Question Type (check one)
- [X] Multiple Choice
- [ ] Matching
- [ ] Free Format (Essay)

(D) Bank
- Operations [X]
- OP002 [ ]

(E)	1	2	3	4	5	6	7	8
Keywords :	Category	Topic 1	Topic 2	JTA	Setting	Other Objs.	Quiz Only	Retired
(≤9 characters)	OPS	ONXXXXXX						

(F) Point Value: 1 (G) Answer Time: (Minutes)

(I) Review Date (YYMM):

(H) Cognitive Level: (Check one)
- [ ] 1 Memory
- [ ] 2 Comprehension
- [ ] 3 Application
- [X] 4 Analysis
- [ ] 5 Problem Solving

(J) QUESTION:

Unit 1 is operating at 95% power when the High Pressure Coolant Injection (HPCI) system initiates on a spurious high drywell pressure signal. Which one of the following sets of parameters would result from this transient?

	APRM Power	Total Core Flow	Generator MWe	Feedwater Flow
a.	RISE	NO CHANGE	LOWER	LOWER
b.	NO CHANGE	LOWER	RISE	NO CHANGE
c.	RISE	NO CHANGE	RISE	LOWER
d.	NO CHANGE	LOWER	LOWER	NO CHANGE

(K) ANSWER: c.

Accept a. & C

Two candidates assumed this question referred to these parameters when the transient first occurred, i.e. steam flow is diverted from the turbine/generator to feed HPCI, this results in generator MWe lowering. Per the attached curve this does occur

E. W. Bowler
R.E. Chin

OPERATIONS QUESTION AND ANSWER INPUT FORM

(L) REQUIRED MATERIALS: None

(M) K&A NUMBER/RATING: 4(N) NOTES: 2.2.34/2.8

JUSTIFICATION:	An inadvertent HPCI injection will cause a rise in power from the cooler feedwater, a reduction in feedwater flow to maintain RPV water level with the additional HPCI flow, increased main generator output from the rise in power, and no change in core flow.		
DISTRACTER A:	Generator output will rise because of the rise in power.		
DISTRACTER B:	Reactor power will rise, core flow will not lower and feedwate flow will lower		
DISTRACTER D:	Reactor power will rise, core flow will not lower generator output will rise and feedwate flow will lower		
EXAM OUTLINE CROSS-REF:	LEVEL:	RO	SRO
	TIER:	3	3
	GROUP:	-	-
K/A TEXT:	2.2.34 – Knowledge of the process of for determining the internal and external effects on core reactivity.		
QUESTION SOURCE:	BANK:		
	MODIFIED:	X	
	NEW:		
10CFR55:	55.41(b).1, 55.41(b).5		
COMMENTS:	Used on 12/91 NRC Exam		

(O) REFERENCES: Chapter 13, FSAR

(P) POSITIONS:
(check one or more boxes)

R – RO	S - SRO	A - ASO	N - NPO	T - STA
X	X			

(Q) Prepared by ED BOWLES

(R) Reviewed by: R.E. Cli

REAL TIME DATE [redacted]
17:28:00 Aug 10 [redacted]

Environment		
REMFS	OVRDS	TRIGS
0	0:0	0

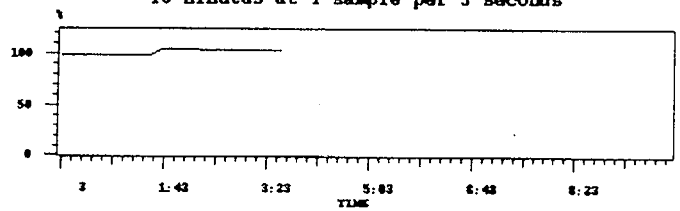
Initial Conditions		
SNAP	RESET	CURR
200	182	182

RUN TIME
3:52

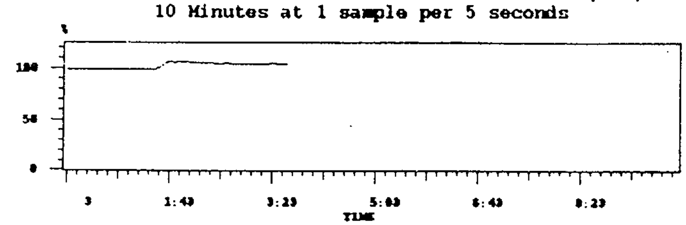
HELP | DISPLAY | COMMAND | UTILITIES | SNAPSHOT | RESET | BACKTRACK | ANNUN. | RUN

MONITORED PARAMETER TRENDS MPT

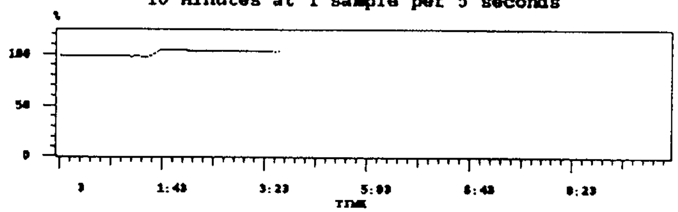
line 1: ZAONRC511R603AA APRM FLUX CHAN A (RED)
10 Minutes at 1 sample per 5 seconds



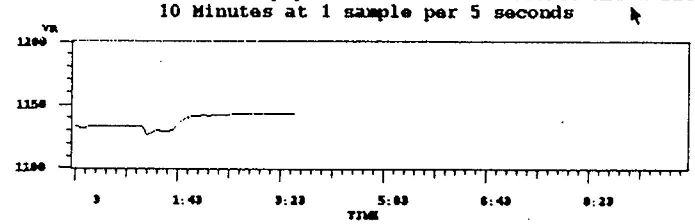
line 2: ZAONRC511R603CA APRM FLUX CHAN E (RED)
10 Minutes at 1 sample per 5 seconds



line 3: ZAONRC511R603BA APRM FLUX CHAN B (RED)
10 Minutes at 1 sample per 5 seconds



line 4: ZAOXR10001[2] GENERATOR GROSS OUTPUT RED=W BLU
10 Minutes at 1 sample per 5 seconds



MALFUNCTIONS

HP152004

List Files
Display Format
CHART
SUMMARY

TOP | PAGE UP | LINE UP
BOTTOM | PAGE DOWN | LINE DOWN

Display/Command? Transition from Run mode to Freeze mode complete.

Display/Command?

The opening of a relief is a relatively minor transient for the reactor and core. Variations in plant parameters are shown on Figure 6. The sudden increase in steam flow leaving the reactor vessel causes a mild depressurization transient. The pressure regulator senses the system pressure decrease and within a few seconds, closes the turbine control valves far enough to stabilize reactor pressure at a slightly lower pressure. Reactor power settles out at nearly the initial power level.

Turbine megawatts will decrease from diversion of steam through the SRV to the suppression pool (a 720,000 lbm/hr). The decrease in MW output will be proportional to the percent change in turbine steam flow. Since the steam flow through the SRV is not measured at the steam line flow restrictors, the feedwater control system will see a steam/feed flow mismatch. Final steady-state reactor vessel water level will be slightly lower than its initial value, such that the level error signal present will offset the steam/feed flow mismatch signal.

This transient is classified as a decrease in moderator temperature due to the small decrease in pressure and the corresponding P_{sat}/T_{sat} relationship. However this, is not the critical parameter during this transient. The major concern with a stuck open relief is the resultant heating of the suppression pool to its design temperature and resultant decrease of the pool heat capacity. If the temperature of the pool gets too high, condensation of steam during a LOCA or an ADS blowdown will not be complete, possibly causing containment pressure to exceed its design limits. If a stuck open relief cannot be closed, the reactor will have to be shut down and cooled down. Any of the following conditions will require a scram;

- It becomes evident that the SRV will not close.
- The SRV has not closed after two minutes have elapsed.
- Suppression pool temperature has reached 105 °F.

Shutting down the reactor will ensure the plant is not operating without sufficient heat capacity in the pool. After the scram, the MSIVs should remain open and the cooldown performed by steaming through the bypass valves to the main condenser. It is possible that with the relief open, the cooldown may exceed the limit of 100 °F in an hour period.

VI. Inadvertent Initiation of HPCI

As stated earlier, this transient is actually classified as an increase in reactor coolant inventory in the FSAR. Because it is the single event analyzed in that category, and because it does result in an increase in core inlet subcooling, it is included here. The inadvertent initiation of HPCI is assumed to occur as a result of operator error.

Figure 7 illustrates the net effect of this transient in terms of steam flow, feedwater flow, and water level. In Figure 7a, plant is operating at steady-state. In Figure 7b, HPCI has inadvertently been initiated and HPCI has reached rated flow. Though there is no measured steam flow feed flow mismatch, there is actually a greater rate of feed into the vessel, since the HPCI injection is greater than its steam usage. Reactor vessel water level will rise as a result. This creates a water level error signal causing the feedwater flow to be reduced. The indicated steam/feed flow mismatch signal will offset the higher water level signal to maintain a constant water level in the vessel. See Figure 7.

A plot of major plant parameters for this transient is shown on Figure 8. When HPCI is initiated, steam flow out of the vessel increases. HPCI steam flow is only about 180,000 lbm/hr or about 1.3 percent of rated steam flow. Thus, no significant pressure drop will be seen in the reactor. In about 25 seconds, HPCI will begin injecting into the core. The combination of HPCI and feedwater will be close to 120 percent of rated feedwater flow. Reactor water level will begin to increase.

Core inlet subcooling increases due to extra injection and cooler injection of HPCI flow into the downcomer. Reactor power increases. Steam flow to the turbine does not increase significantly with the power, since the increase in power is needed to heat the cooler water and create the extra steam required for the HPCI turbine.

The rising water level will cause the feedwater control system to reduce feedwater flow. Feedwater flow will decrease until feed and HPCI flows combined equals the steam flow. The final steady-state conditions will be:

1. Higher reactor power (Turbine power remains about the same)
2. Higher vessel water level
3. Lower feedwater flow rate
4. Indicated steam/feed flow mismatch
5. Lower MCPR (due to higher heat flux)

VII. Inadvertent RHR Shutdown Cooling Operation

Since the RHR system is a low pressure system, it could not be operated in the shutdown cooling mode while the reactor is at operating conditions. If the reactor were critical or near critical on a startup or shutdown, misoperation of the shutdown cooling mode of RHR could result in a moderator temperature decrease. This would cause a slow insertion of positive reactivity into the core, causing a slow increase in the fission rate. This flux increase would be controlled by the operator in the same manner normally used to control the fission rate in the source or intermediate range. If for some reason no operator action is taken, the fission rate increase will be terminated by a scram before any fuel damage could occur.

VIII. Summary

A loss of feedwater heating due to a loss of extraction steam to one or more feedwater heaters will result in colder feedwater entering the reactor. The analysis is shown on Figure 9. The increase in core inlet subcooling will cause reactor power to increase. An APRM reactor scram is possible under the most severe conditions. In addition to the higher MW_{th}, generator MW_e will increase due to the increased steam flow through the turbine. Key parameters of interest in the transient include:

1. Reactor power - Increases due to a rise in core inlet subcooling
2. Reactor pressure – Increases because of greater steam generation
3. MCPR - Decrease