BSEP 15-0070
Enclosure

Outline for Operating Test

(Enclosure only being sent to Chief Examiner)

Facility: Brunswick	Date of Examination: Nov/Dec 2015
Examination Level: RO ■ *SRO* ■	Operating Test Number: FINAL

Administrative Topic (see Note)	Type Code*	Describe activity to be performed
Conduct of Operations (COO-01) (RO, **then SRO**)	R, M	***Determine SRM/IRM Overlap*** 2.1.7 Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior, and instrument interpretation.
Conduct of Operations (COO-02) (**SRO only**)	R, D	***Determine Overtime Eligibility*** 2.1.3 Knowledge of shift or short-term relief turnover practices
Conduct of Operations (COO-03) (RO)	R, N	***Time-to-Boil Calculation*** 2.1.1 Knowledge of Conduct of Operations requirements.
Equipment Control (RO)	R, D	***Evaluate Core Spray Operability*** 2.2.37 Ability to determine operability and/or availability of safety related equipment.
Equipment Control (**SRO only**)	R, D	***Perform Safety Function Determination*** 2.2.22 Knowledge of Limiting Conditions for Operations and Safety Limits
Radiation Control (RO and **SRO**)	R, D	***Determine Stay Time in High Radiation Area*** 2.3.4 Knowledge of Radiation Exposure Limits under normal or emergency conditions.
Emergency Procedures/Plan (**SRO Only**)	R, N	**Determine a Protective Action Recommendation (PAR)** 2.4.44 Knowledge of the Emergency Plan Protective Action Recommendations.

NOTE: All items (5 total) are required for SROs. RO applicants require only 4 items unless they are retaking only the administrative topics, when all 5 are required.

* Type Codes & Criteria:

(C)ontrol room, (S)imulator, or Class(R)oom
(D)irect from bank (≤ 3 for ROs; ≤ 4 for SROs & RO retakes)
(N)ew or (M)odified from bank (≥ 1)
(P)revious 2 exams (≤ 1; randomly selected)

Conduct of Operations (COO-01) (RO, then SRO)

Calculate GAFs and Tech Spec Assessment R, M

K/A 2.1.7 Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior, and instrument interpretation.

This is a modified JPM that requires the Examinee to determine SRM/IRM overlap and then, for SRO only candidates, determine Technical Specification applicability. The readings were modified to make another IRM inoperable, and the SRO only portion was added, to provide a modification to the original Bank JPM.

Conduct of Operations (COO-02) (SRO only)

Evaluate Overtime Eligibility R, D

K/A 2.1.9 Ability to direct personnel activities inside the control room.

This is a Bank JPM that requires the SRO to determine overtime eligibility for several employees. Although the numbers of hours worked was not modified, the procedure governing working hours has changed to make the JPM different from the original.

Conduct of Operations (COO-03) (RO)

Time to Boil Calculation R, N

K/A 2.1.1 Knowledge of Conduct of Operations requirements

This is a new JPM that requires the Examinee to determine the time to boil IAW 2OI-03.4.1, Reactor Operator Daily Check Sheets.

Equipment Control (RO)

Evaluate Core Spray Operability R, D

K/A 2.2.37 Ability to determine operability and/or availability of safety related equipment.

This is a bank JPM that requires the Examinee to review Core Spray Operability test date and determine what parameters do not meet the Acceptance Criteria.

Radiation Control (RO and SRO)

Determine Stay Time in High Radiation Area R, P

K/A 2.3.4 Knowledge of Radiation Exposure Limits under normal or emergency conditions.

The is a bank JPM. It requires the Examinee to determine the stay time for workers in a high radiation area, and if they have exceeded administrative dose limits.

Emergency Procedures/Plan (SRO only)

Determine Protective Action Recommendations (PAR) R, N

GEN 2.4.44 Knowledge of Emergency Plan Protective Action Recommendations

This is a new JPM that requires the SRO Examinee to determine PARs during a General Emergency and evaluate whether a KI recommendation is warranted.

Facility: Brunswick		Date of Examination: NOV/DEC 2015
Exam Level: RO ■ SRO-I ■ ***SRO-U*** ■		Operating Test No.: DRAFT
Control Room Systems* (8 for RO); (7 for SRO-I) 2 or 5 for SRO-U		
System / JPM Title	Type Code*	Safety Function
a. Initiation of SLC System with RWCU Isolation Failure	A,S,P,EN	1
b. (RO ONLY) Start RCIC with steam line failure	A.S,P,L	2
c. Test the Main Steam Isolation Valves	N,S	3
d. Shifting Stator Cooling Pumps – Pump Trip	A,S,D	4
e. Isolate Recirc Pump IAW 0AOP-14.0 with THI	N,A,S	5
f. Bus E3 Normal feeder to DG3	S,D,P	6
g. Restoration of APRM Rod Block and Scram Setpoints from Single Loop Operation to Two Loop Operation	S,D	7
h. Restart RB HVAC with Failure to Isolate	A,S,D	9
In-Plant Systems* (3 for RO); (3 for SRO-I); (3 or 3 for SRO-U)		
i. Resetting RCIC Mechanical Overspeed	D,E,R	2
j. Unloaded Maintenance Start of the Supp DG	E,D	6
k. Local Deluge System Manual Operation for SBGT Train	D,R	8

* All RO and SRO-I control room (and in-plant) systems must be different and serve different safety functions; all five SRO-U systems must serve different safety functions; in-plant systems and functions may overlap those tested in the control room.

* Type Codes	Criteria for RO / SRO-I / SRO-U
(A)lternate path	4-6 / 4-6 / 2-3
(C)ontrol room	
(D)irect from bank	≤ 9 / ≤ 8 / ≤ 4
(E)mergency or abnormal in-plant	≥ 1 / ≥ 1 / ≥ 1
(EN)gineered safety feature	≥ 1 / ≥ 1 / ≥ 1 (control room system)
(L)ow-Power / Shutdown	≥ 1 / ≥ 1 / ≥ 1
(N)ew or (M)odified from bank including 1(A)	≥ 2 / ≥ 2 / ≥ 1
(P)revious 2 exams	≤ 3 / ≤ 3 / ≤ 2 (randomly selected)
(R)CA	≥ 1 / ≥ 1 / ≥ 1
(S)imulator	

a. Manual Initiation of SLC System with RWCU Isolation Failure

211000 A4.08 — Ability to operate and/or monitor in the control room: System Initiation

This is a simulator alternate path JPM that will have the examinees initiating SLC. When the system is started the RWCU Outboard Isolation Valve, G31-F004 does not close and the examinee is expected to take action to close this valve. This JPM was randomly selected from the 2014 NRC exam.

b. RCIC Start Per The Hard Card – Steam line break

217000 A4.08 — Ability to manually operate and/or monitor RCIC system flow

This is a simulator alternate path JPM that will require the examinee to start RCIC for injection per the Hard Card and restore RPV water level. As an alternate path the steam line breaks and RCIC does not auto isolate requiring manual isolation of RCIC. RCIC is an engineered safety feature.

c. Test the Main Steam Isolation Valves

239001 A4.01 — Ability to manually operate and/or monitor the MSIVs in the Control Room

This is a new JPM that will require the examinee to perform post-maintenance testing of a MSIV.

d. Shifting Stator Cooling Pumps – Pump Trip

245000 EA.21 — Ability to manually operate and/or monitor in the control room: Stator water cooling pumps

This is a banked alternate path simulator JPM that will require the examinee to swap stator cooling water pumps so that maintenance can be performed on the currently running pump. When the operating pump is secured, a malfunction on the alternate pump will require restarting the pump that was initially running.

e. Secure Recirculation Pump IAW AOP-14 - THI

295024 A4.04 — Ability to operate and/or monitor the following as they apply to HIGH DRYWELL PRESSURE: Recirculation System

This is a new alternate path simulator JPM that requires the examinee to secure and isolate a Recirculation Pump due to a failing Recirc Pump seal. Indications of Thermal Hydraulic Instability will require scramming the reactor.

f. Manual Transfer of Bus E3 from the Normal Feeder to the DG3

264000 A4.04 — Ability to manually operate and/or monitor in the control room: Manual start, loading, and stopping of emergency generator.

This is a banked simulator JPM that will require the examinee to perform the Control Operator actions associated with the manual transfer of E3 from the Normal Feeder to DG3 IAW 0OP-50.1, Diesel Generator Emergency Power System Operating Procedure

g. Restoration of APRM Setpoints from Single Loop to Two Loop Operation

201005 A1.04 — Ability to predict and/or monitor changes in Scram and Rod Block trip setpoints associated with operating APRM system controls.

This is a banked JPM that will require the examinee to perform the restoration of APRM rod block and scram setpoints following return to 2-loop Recirc Pump operation.

h. Restart RB HVAC with Failure to Isolate

288000 A3.01 — Ability to monitor Plant Ventilation System automatic isolation/initiation signals in the control room

This is a banked alternate path simulator JPM that will require the examinee to restart Reactor Building HVAC per SEP-04. After Reactor Building HVAC is restarted, high radiation levels will require the examinee to isolate the Reactor Building.

i. Resetting RCIC Mechanical Overspeed

295031 EA 1.05 — Ability to operate and/or monitor RCIC it applies to reactor low water level

This is a banked in-plant JPM that will require the examinee to manually reset the RCIC Mechanical Overspeed Trip device. This JPM is performed in the RCA.

j. Unloaded Maintenance Start of the Supp DG

264000 A3.03 — Ability to monitor automatic operations of the EMERGENCY GENERATORS (DIESEL/JET) including indicating lights, meters, and recorders.

This is a banked in-plant JPM that will require the examinee to simulate the actions associated with performing the field actions for starting the Supp DG, which is a recent plant modification.

k. Local Deluge System Manual Operation for SBGT Train

286000 A2.08 — Failure of Fire Protection System to Actuate When Required

This is a banked in-plant JPM that will require the examinee to simulate manually initiating the SBGT deluge system. This JPM is performed in the RCA.

DUKE ENERGY
BRUNSWICK TRAINING SECTION
JOB PERFORMANCE MEASURE

LESSON TITLE: Determine SRM/IRM Overlap Per GP-02

LESSON NUMBER: LOT-OJT-JP-307-A03

REVISION NO: 3

Lou Sosler 9/10/205
PREPARER / DATE

John Biggs 9/15/2015
TECHNICAL REVIEWER / DATE

Brian Moschet 9/10/2015
VALIDATOR / DATE

Jerry Pierce 9/23/2015
LINE SUPERVISOR / DATE

Jim Barry 9/25/2015
TRAINING SUPERVISION APPROVAL / DATE

RELATED TASKS:

215201B101, Verify Correct Overlap Between SRMs And IRMs Per GP-02

K/A REFERENCE AND IMPORTANCE RATING:

GEN 2.1.7 3.7/4.4
Ability to evaluate plant performance and make operational judgments based on operating characteristics / reactor behavior / and instrument interpretation.

REFERENCES:

0GP-02

TOOLS AND EQUIPMENT:

None

SAFETY FUNCTION (from NUREG 1123, Rev. 2, Supp. 1):

Generic (Administrative)

SETUP INSTRUCTIONS

None

SAFETY CONSIDERATIONS:

1. None

EVALUATOR NOTES: (Do not read to performer)

1. The applicable procedure section **WILL NOT** be provided to the trainee.
2. Prior to the first JPM of the JPM set, provide the JPM briefing contained in NUREG-1021, Appendix E, or similar briefing (for non-regulated exams) to the trainee(s).
3. Critical Step Basis
   a) Prevents Task Completion
   b) May Result in Equipment Damage
   c) Affects Public Health and Safety
   d) Could Result in Personal Injury

Read the following to the JPM performer.

TASK CONDITIONS:

1. Unit One startup is being performed per 0GP-02.
2. Initial (pre-startup) SRM and IRM readings were recorded as follows:

SRM Channel	Reading	IRM Channel	Reading*
A	100 CPS	A	3%
B	150 CPS	B	2%
C	150 CPS	C	4%
D	100 CPS	D	5%
		E	8%
		F	6%
		G	7%
		H	5%

3. Current SRM and IRM readings are as follows:

SRM Channel	Reading	IRM Channel	Reading*
A	2×10^5 CPS	A	11%
B	9×10^4 CPS	B	14%
C	5×10^5 CPS	C	16%
D	3×10^5 CPS	D	10%
		E	15%
		F	18%
		G	13%
		H	17%

*** All IRM Readings taken on Range One From the Bar Graph Recorder (0-125)**

It is NOT desired to use the highest reading IRM (pre-startup) for overlap criteria for all IRMs.

INITIATING CUE:

RO, and SRO candidates:

You are directed to determine if proper SRM/IRM overlap exists in accordance with GP-02.

For each IRM channel, state if overlap is met, and for each RPS trip system, state if a sufficient number of IRM inputs is available.

SRO ONLY:

Determine required actions, if any, for the given conditions.

PERFORMANCE CHECKLIST

NOTE: Sequence is assumed unless otherwise indicated, comments required for any step evaluated as UNSAT.

Step 1 - Perform take a minute at job site prior to beginning task.

Examinee should cover the following questions, as deemed necessary. What are the hazards in the area? What PPE is required? Tools/PPE inspected prior to use? Energy sources secured/isolated? Is Clearance/Tag Out sufficient? What's the worst that can happen? Any ALARA concerns? Will I affect plant status? HU Tools needed?

SAT/UNSAT

TIME START: ___________

Step 2 – Determine SRM/IRM overlap criteria is not met for IRM A and D based on not reading 10% of scale.

Determines that SRM/IRM overlap criteria is not met for IRM A and D based on not reading 10% of scale.

**** CRITICAL STEP **   SAT/UNSAT**

Step 3 – Determine SRM/IRM overlap criteria is not met for IRMs E and G based on not reading double the initial reading.

Determines that SRM/IRM overlap criteria is not met for IRM E and G based on not reading double the initial reading.

**** CRITICAL STEP **   SAT/UNSAT**

Step 4 – Determine RPS Trip System A does not have sufficient IRM inputs.

Determines that RPS Trip System A does not have sufficient IRM inputs.

**** CRITICAL STEP **   SAT/UNSAT**

TERMINATING CUE: When SRM/IRM overlap determination has been made, and RPS is evaluated, this JPM is complete for RO candidates.

TIME COMPLETED: ___________

SRO Candidates ONLY:

NOTE:	GP-02 directs maintaining reactor power on the SRMs by inserting control rods. This is acceptable information, but not required for the JPM.

Step 5 – Determine that TS. 3.3.1.1 requires RPS Trip System be placed in tripped conditions within 12 hours.

Determines that RPS Trip System be placed in tripped conditions within 12 hours.

**** CRITICAL STEP ** SAT/UNSAT**

TERMINATING CUE:	When SRM/IRM overlap determination has been made, RPS is evaluated, and Tech Spec action statement is determined, this JPM is complete for SRO candidates.

TIME COMPLETED ______________

COLLECT AND CONTROL ALL JPM EXAM MATERIALS FOR EXAM SECURITY.

Step	Critical / Not Critical	Reason
1	Not Critical	Administrative
2	Critical	Required to complete JPM correctly.
3	Critical	Required to complete JPM correctly.
4	Critical	Required to complete JPM correctly.
5	Critical	Required to complete JPM correctly.

REVISION SUMMARY

3	Revise to new JPM format. Added Technical Specification determination as SRO only portion.

Validation Time: 20 Minutes (approximate).

Time Taken: ____ Minutes

APPLICABLE METHOD OF TESTING

Performance:	Simulate	____	Actual	X	Unit:	1
Setting:	In-Plant	____	Simulator	____	Admin	X
Time Critical:	Yes	____	No	X	Time Limit	N/A
Alternate Path:	Yes	____	No	X		

EVALUATION

Performer: ______________________________

JPM: Pass ______ Fail ______

Remedial Training Required: Yes ______ No ______

Comments: ______________________________

☐ Comments reviewed with Performer

Evaluator Signature: ______________________ Date: __________

Read the following to the JPM performer.

TASK CONDITIONS:

1. Unit One startup is being performed per 0GP-02.

2. Initial (pre-startup) SRM and IRM readings were recorded as follows:

SRM Channel	Reading	IRM Channel	Reading*
A	100 CPS	A	3%
B	150 CPS	B	2%
C	150 CPS	C	4%
D	100 CPS	D	5%
		E	8%
		F	6%
		G	7%
		H	5%

4. Current SRM and IRM readings are as follows:

SRM Channel	Reading	IRM Channel	Reading*
A	$2 X 10^5$ CPS	A	11%
B	$9 X 10^4$ CPS	B	14%
C	$5 X 10^5$ CPS	C	16%
D	$3 X 10^5$ CPS	D	10%
		E	15%
		F	18%
		G	13%
		H	17%

*** All IRM Readings taken on Range One From the Bar Graph Recorder (0-125)**

It is NOT desired to use the highest reading IRM (pre-startup) for overlap criteria for all IRMs.

INITIATING CUE:

RO, and SRO candidates:

You are directed to determine if proper SRM/IRM overlap exists in accordance with GP-02.

For each IRM channel, state if overlap is met, and for each RPS trip system, state if a sufficient number of IRM inputs is available.

SRO ONLY:

Determine required actions, if any, for the given conditions.

DUKE ENERGY

BRUNSWICK TRAINING SECTION

JOB PERFORMANCE MEASURE

LESSON TITLE: Evaluate Overtime Eligibility

LESSON NUMBER: LOT-ADM-JP-201-D01

REVISION NO: 1

Lou Sosler 9/10/2015

PREPARER / DATE

John Biggs 9/15/2015

TECHNICAL REVIEWER / DATE

Brian Moschet 9/10/2015

VALIDATOR / DATE

Jerry Pierce 9/23/2015

LINE SUPERVISOR / DATE

Jim Barry 9/25/2015

TRAINING SUPERVISION APPROVAL / DATE

RELATED TASKS:

Conduct Shift Turnover and Relief

K/A REFERENCE AND IMPORTANCE RATING:

Gen 2.1.9 2.9/4.5
Ability to direct personnel activities inside the control room

REFERENCES:

AD-SY-ALL-0460, Managing Fatigue and Work Hour Limits

TOOLS AND EQUIPMENT:

None

SAFETY FUNCTION (from NUREG 1123, Rev. 2, Supp. 1):

Admin – Conduct of Operations

SETUP INSTRUCTIONS

None

SAFETY CONSIDERATIONS:

1. None

EVALUATOR NOTES: (Do not read to performer)

1. The applicable procedure section **WILL** be provided to the trainee.
2. Prior to the first JPM of the JPM set, provide the JPM briefing contained in NUREG-1021, Appendix E, or similar briefing (for non-regulated exams) to the trainee(s).
3. Critical Step Basis
   a) Prevents Task Completion
   b) May Result in Equipment Damage
   c) Affects Public Health and Safety
   d) Could Result in Personal Injury

Read the following to the JPM performer.

TASK CONDITIONS:

1. A startup of Unit 1 is planned for the following shift. One Reactor Operator must be held over three hours for startup.

2. The following is the work history (excluding shift turnover time) of the available Reactor Operators on shift (hours reflect those worked PRIOR to the 3 hour holdover). A break of at least 8 hours occurred between all working periods. All operators began their shift schedule at the same time each day.

DAY	1	2	3	4	5	6	7	8
Operator #1	0	4	12	10	10	14	10	11
Operator #2	0	12	10	12	3	12	8	13
Operator #3	0	0	12	12	12	8	8	14
Operator #4	0	8	12	10	10	8	10	11
Operator #5	0	0	13	14	13	10	13	10

NOTE: A break of at least 8 hours has occurred between all work periods

INITIATING CUE:

Evaluate the work history for all 5 operators. Determine which operator(s), if any, can be held over for three hours without prior overtime approval, and determine which operators CANNOT be held over for three hours without prior overtime approval.

Also identify ALL deviations to AD-SY-ALL-0460 that may have already occurred between Day 1 and Day 7 (assume no authorization to exceed AD-SY-ALL-0460 limits)

PERFORMANCE CHECKLIST

NOTE: Sequence is assumed unless otherwise indicated, comments required for any step evaluated as UNSAT.

Step 1 - Perform take a minute at job site prior to beginning task.

Examinee should cover the following questions, as deemed necessary. What are the hazards in the area? What PPE is required? Tools/PPE inspected prior to use? Energy sources secured/isolated? Is Clearance/Tag Out sufficient? What's the worst that can happen? Any ALARA concerns? Will I affect plant status? HU Tools needed?

SAT/UNSAT

TIME START ___________

> **NOTE:** It is Critical that the examinee correctly determines which operators can be held over for three hours, and which operators cannot without overtime authorization per AD-SY-ALL-0460.
>
> It is not critical that the examinee identify the specific limit that would be exceeded for the operators who cannot be held over.

Step 2 – Determine Operator **#1** would exceed 72 hours in a 7 day period and would require overtime authorization.

*Determines Operator **#1** would require overtime authorization.*

****CRITICAL STEP** SAT/UNSAT**

Step 3 – Determine Operator **#2** would exceed 72 hours in a 7 day period and would require overtime authorization.

*Determines Operator **#2** would require overtime authorization.*

****CRITICAL STEP** SAT/UNSAT**

Step 4 – Determine Operator **#3** would exceed 16 hours straight and 16 hours in a 24 hour period (today) and 24 hours in a 48 hour period (days 7 and 8) and would require overtime authorization.

*Determines Operator **#3** would require overtime authorization.*

****CRITICAL STEP** SAT/UNSAT**

Step 5 – Determine Operator **#4** would not exceed any overtime restrictions and could be held over for the 3 hours.

*Determines that Operator **#4** would not exceed any overtime restrictions and could be held over for the 3 hours.*

SAT/UNSAT

NOTE: It is Critical that the examinee correctly determines that Operator #5 cannot be held over, and therefore, one of the next 2 steps is Critical.

Step 6 – Determine Operator **#5** would exceed 64 hours in a 48 hour period and would exceed 72 hours in a 7 day period and would require overtime authorization.

*Determines Operator **#5** would require overtime authorization.*

****CRITICAL STEP** SAT/UNSAT**

or

NOTE: If asked, inform examinee that no authorization to exceed AD-SY-ALL-0460 limits has been approved.

Step 7 – Determine Operator **#5** has exceeded 26 hours in a 48 hour period (day 3 and 4, and day 4 and 5).

*Determines that Operator **#5** exceeded 26 hours in a 48 hour period (day 3 and 4, and day 4 and 5).*

****CRITICAL STEP** SAT/UNSAT**

TERMINATING CUE: When the examinee has evaluated overtime restrictions, this JPM is complete.

TIME COMPLETED ____________

COLLECT AND CONTROL ALL JPM EXAM MATERIALS FOR EXAM SECURITY.

Step	Critical / Not Critical	Reason
1	Not Critical	Administrative
2	Critical	Personnel safety and federal regulation
3	Critical	Personnel safety and federal regulation
4	Critical	Personnel safety and federal regulation
5	Not Critical	No limits violated
6	Critical	Personnel safety and federal regulation
	or	
7	Critical	Personnel safety and federal regulation

REVISION SUMMARY

1	Updated to new JPM template.

Validation Time: 30 Minutes (approximate).

Time Taken: ____ Minutes

APPLICABLE METHOD OF TESTING

Performance:	Simulate	____	Actual	X	Unit:	1
Setting:	In-Plant	____	Simulator	____	Admin	X
Time Critical:	Yes	____	No	X	Time Limit	N/A
Alternate Path:	Yes	____	No	X		

EVALUATION

Performer: ____

JPM: Pass _____ Fail _____

Remedial Training Required: Yes _____ No _____

Comments: ____

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____

☐ Comments reviewed with Performer

Evaluator Signature: ______________________________ Date: ____________

TASK CONDITIONS:

1. A startup of Unit 1 is planned for the following shift. One Reactor Operator must be held over three hours for startup.

2. The following is the work history (excluding shift turnover time) of the available Reactor Operators on shift (hours reflect those worked PRIOR to the 3 hour holdover). A break of at least 8 hours occurred between all working periods. All operators began their shift schedule at the same time each day.

DAY	1	2	3	4	5	6	7	8
Operator #1	0	4	12	10	10	14	10	11
Operator #2	0	12	10	12	3	12	8	13
Operator #3	0	0	12	12	12	8	8	14
Operator #4	0	8	12	10	10	8	10	11
Operator #5	0	0	13	14	13	10	13	10

NOTE: A break of at least 8 hours has occurred between all work periods

INITIATING CUE:

Evaluate the work history for all 5 operators. Determine which operator(s), if any, can be held over for three hours without prior overtime approval, and determine which operators CANNOT be held over for three hours without prior overtime approval.

Also identify ALL deviations to AD-SY-ALL-0460 that may have already occurred between Day 1 and Day 7 (assume no authorization to exceed AD-SY-ALL-0460 limits)

DUKE ENERGY

BRUNSWICK TRAINING SECTION

JOB PERFORMANCE MEASURE

LESSON TITLE: Perform 'Time-To-200°F' Calculation

LESSON NUMBER: LOT-ADM-JP-201-D14

REVISION NO: 0

Lou Sosler 9/10/2015

PREPARER / DATE

John Biggs 9/15/2015

TECHNICAL REVIEWER / DATE

Brian Moschet 9/10/2015

Thomas Baker 9/10/205

VALIDATOR / DATE

Jerry Pierce 9/23/2015

LINE SUPERVISOR / DATE

Jim Barry 9/25/2015

TRAINING SUPERVISION APPROVAL / DATE

RELATED TASKS:

299202B201: Perform Daily Check Sheet Per OI-3.4.1

K/A REFERENCE AND IMPORTANCE RATING:

Generic 2.1.1 3.8/4.2
Knowledge of Conduct of Operations requirements

REFERENCES:

2OI-03.4.1, Unit 2 Reactor Operator Daily Check Sheets
0G41-0020, Refueling Outage Decay Heat Load Evaluation

TOOLS AND EQUIPMENT:

Calculator

SAFETY FUNCTION (from NUREG 1123, Rev. 2, Supp. 1):

Generic – Conduct of Operations

SETUP INSTRUCTIONS

None

SAFETY CONSIDERATIONS:

1. None

EVALUATOR NOTES: (Do not read to performer)

1. The applicable procedure section **WILL** be provided to the trainee.
2. Prior to the first JPM of the JPM set, provide the JPM briefing contained in NUREG-1021, Appendix E, or similar briefing (for non-regulated exams) to the trainee(s).
3. Critical Step Basis
   a) Prevents Task Completion
   b) May Result in Equipment Damage
   c) Affects Public Health and Safety
   d) Could Result in Personal Injury

Read the following to the JPM performer

TASK CONDITIONS:

1. Today is December 7, 2015.
2. Unit 2 is in Mode 2 preparing to perform a reactor startup.
3. The Reactor Operator is performing 2OI-03.4.1, Reactor Operator Daily Check Sheets
4. Current Spent Fuel Pool (SFP) temperature is 96.8°F

INITIATING CUE:

Calculate and record the SFP "Time to 200°F".

PERFORMANCE CHECKLIST

NOTE: Sequence is assumed unless otherwise indicated, comments required for any step evaluated as UNSAT.

Step 1 - Perform take a minute at job site prior to beginning task.

Examinee should cover the following questions, as deemed necessary. What are the hazards in the area? What PPE is required? Tools/PPE inspected prior to use? Energy sources secured/isolated? Is Clearance/Tag Out sufficient? What's the worst that can happen? Any ALARA concerns? Will I affect plant status? HU Tools needed?

SAT/UNSAT

TIME START: ___________

> **NOTE:** From 2OI-03.4.1, Note KK
> The "Time to 200°F" calculation is only required when the fuel pool gates are installed. The "Time to 200°F" is calculated by subtracting the current SFP Temperature from 200°F to get the delta°F, which is then divided by the Unit SFP H/U Rate for today's date from Attachment P of Calculation 0G41-0020.
> *(200°F – current SFP temp = delta temp)/Unit SFP HUR = Time to 200°F*
> Example: (200°F – 98.1°F = 101.9°F)/0.91 = 111.97 hrs until 200°F

Step 2 – Determine the Unit 2 SFP HUR using 0G41-0020, Attachment P.

Unit 2 SFP HUR determined to be 1.24

****CRITICAL STEP** SAT/UNSAT**

Step 3 – Calculate "Time to 200°F".

"Time to 200°F"calculated using 200°F – 96.8°F = 103.2°F)/1.24 = 83.2 hours until 200°F

****CRITICAL STEP** SAT/UNSAT**

> **TERMINATING CUE:** When "Time to 200°F" is calculated and recorded, this JPM is complete.

TIME COMPLETED: _____________

COLLECT AND CONTROL ALL JPM EXAM MATERIALS FOR EXAM SECURITY.

Step	Critical / Not Critical	Reason
1	Not Critical	Administrative
2	Critical	Required to complete task.
3	Critical	Required to complete task.

REVISION SUMMARY

0	New JPM

Validation Time: 10 Minutes (approximate).

Time Taken: ____ Minutes

APPLICABLE METHOD OF TESTING

Performance:	Simulate	____	Actual	X	Unit:	2
Setting:	In-Plant	____	Simulator	____	Admin	X
Time Critical:	Yes	____	No	X	Time Limit	N/A
Alternate Path:	Yes	____	No	X		

EVALUATION

Performer: ____

JPM: Pass ______ Fail ______

Remedial Training Required: Yes ______ No ______

Comments: ____

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☐ Comments reviewed with Performer

Evaluator Signature: ________________________________ Date: ______________

TASK CONDITIONS:

1. Today is December 7, 2015.
2. Unit 2 is in Mode 2 preparing to perform a reactor startup.
3. The Reactor Operator is performing 2OI-03.4.1, Reactor Operator Daily Check Sheets
4. Current Spent Fuel Pool (SFP) temperature is 96.8°F

INITIATING CUE:

Calculate and record the SFP "Time to 200°F".

Time to 200°F = ____________.

DUKE ENERGY
BRUNSWICK TRAINING SECTION
JOB PERFORMANCE MEASURE

LESSON TITLE: Determine Protective Action Recommendations (PARs) IAW PEP-02.6.28

LESSON NUMBER: SOT-ADM-301-A20

REVISION NO: 0

Lou Sosler 9/11/2015
PREPARER / DATE

John Biggs 9/15/2015
TECHNICAL REVIEWER / DATE

Kevin Kingston 9/11/2015

Brian Moschet 9/21/2015
VALIDATOR / DATE

Jerry Pierce 9/23/2015
LINE SUPERVISOR / DATE

Jim Barry 9/25/2015
TRAINING SUPERVISION APPROVAL / DATE

RELATED TASKS:

344236B502	Direct Emergency Response As Site Emergency Coordinator Following Declaration Of A General Emergency per PEP-2.1.1
344005B102	Recommend Protective Actions To States And Counties per PEP-02.6.28

K/A REFERENCE AND IMPORTANCE RATING:

GEN 2.4.44 2.4/4.4 Knowledge of Emergency Plan Protective Action Recommendations

REFERENCES:

0PEP-02.1 – Emergency Control – Notification of Unusual Event, Alert, Site Area Emergency and General Emergency
0PEP-02.6.28, Offsite Protective Action Recommendations (PAR)

TOOLS AND EQUIPMENT:

None

SAFETY FUNCTION (from NUREG 1123, Rev. 2, Supp. 1):

Admin – Emergency Procedures/ Plan

SETUP INSTRUCTIONS

None

SAFETY CONSIDERATIONS:

1. None

EVALUATOR NOTES: (Do not read to performer)

1. The examinee will have access to PEP procedures.
2. Prior to the first JPM of the JPM set, provide the JPM briefing contained in NUREG-1021, Appendix E, or similar briefing (for non-regulated exams) to the trainee(s).
3. Emphasize to candidates that this is a Time Critical JPM and that following cue sheet review the evaluator will designate the START TIME on the board and stop the JPM at the applicable critical time.
4. A clock must be available and visible to examiner and examinees.
5. Critical Step Basis
    a. Prevents Task Completion
    b. May Result in Equipment Damage
    c. Affects Public Health and Safety
    d. Could Result in Personal Injury
6. Explain to the examinees they should record their Protective Action Recommendation in the blocks provided beneath the Initiating Conditions.

Read the following to the JPM performer:

****This is a time critical JPM**.** Time begins when directed by the evaluator.

TASK CONDITIONS:

A General Emergency has been declared due to an on-going un-isolable RCIC steam line break, which began 12 hours ago, with indications of fuel failure. Weather data:

- Late fall evening
- Temperature 57°F
- Upper wind speed 16.7 mph
- Lower wind speed 15.8 mph
- Upper wind direction 36.4°
- Lower wind direction 40.8°
- Stability class D

Projected off-site dose per AD-EP-ALL-0202 is 450 mRem TEDE and 3500 mRem CDE. Off-site field survey readings are 280 mRem/hour. The release is expected to drop rapidly due to the emergency depressurization of the Reactor in progress. County and State governments have been notified, and have prepared local evacuation routes.

INITIATING CUE:

A hard copy Emergency Notification Form (ENF) is being completed. Determine what Protective Action Recommendations (PARs) should be made to Off-Site Agencies and determine if Potassium Iodide should be recommended to the General Public.

****This is a time critical JPM** Time now is: __________.**

TIME	PARs

Potassium Iodide (circle one) SHOULD / SHOULD NOT be recommended to the General Public.

PERFORMANCE CHECKLIST

NOTE: Sequence is assumed unless otherwise indicated, comments required for any step evaluated as UNSAT.

Step 1 - Perform take a minute at job site prior to beginning task.

Examinee should cover the following questions, as deemed necessary. What are the hazards in the area? What PPE is required? Tools/PPE inspected prior to use? Energy sources secured/isolated? Is Clearance/Tag Out sufficient? What's the worst that can happen? Any ALARA concerns? Will I affect plant status? HU Tools needed?

SAT/UNSAT

NOTE: Ensure a clock is visible for candidates. Announce and write the Start Time on the board. Start time is when the examinees have been given the initial conditions, initiating cue and state they understand the task (or state they have no questions)

The candidates will have 15 minutes to classify the event. The time, classification and EAL identifier should be recorded by the candidates in the blocks provided beneath the Initiating Conditions.

NOTE: PARs must be made in **15 minutes** from the Start Time.

START TIME __________

NOTE: IAW 0PEP-02.6.21, enter lower wind direction and wind speed if completing hard copy ENF.

Step 2 – Determine that Zones A, B, C, D, and E, should be evacuated.

Using 0PEP-2.6.28, Attachments 1 and 2, determines that Zones A, B, C, D, and E should be evacuated.

****CRITICAL STEP** SAT/UNSAT**

Step 3 – Determine that Zones F, G, H, J, K, L, M, and N, should be sheltered.

Using 0PEP-2.6.28, Attachments 1 and 2, determines that Zones F, G, H, J, K, L, M, and N, should be sheltered.

****CRITICAL STEP** SAT/UNSAT**

NOTE:	For actual or projected doses greater than 5 Rem CDE Thyroid, then recommend the consideration of KI use by the public.

Step 4 – Determine that KI should not be recommended for use by the General Public.
Determines that KI should NOT be recommended for use by the Public.

SAT/UNSAT

Step 5 – Determinations made within 15 minutes from start time.
Determinations made within 15 minutes.

**** CRITICAL STEP ** SAT/UNSAT**

TERMINATING CUE:	When final determination of PARs and KI recommendation, this JPM is complete.

TIME COMPLETED ______________

COLLECT AND CONTROL ALL JPM EXAM MATERIALS FOR EXAM SECURITY.

Step	Critical / Not Critical	Reason
1	Not Critical	Administrative
2	Critical	Evacuation Zones are critical for protection of the Public.
3	Critical	Shelter Zones are critical for protection of the Public.
4	Not Critical	Recommending KI will not harm the Public.
5	Critical	15- minute time limit is critical for protection of the Public.

REVISION SUMMARY

0	New JPM

Validation Time: _15_ Minutes (approximate).

Time Taken: ____ Minutes

APPLICABLE METHOD OF TESTING

Performance:	Simulate	____	Actual	_X_	Unit:	_2_
Setting:	In-Plant	____	Simulator	____	Admin	_X_
Time Critical:	Yes	_X_	No	____	Time Limit	15 min/15 min
Alternate Path:	Yes	____	No	_X_		

EVALUATION

Performer: ______________________________

JPM: Pass _____ Fail _____

Remedial Training Required: Yes _____ No _____

Comments: ______________________________

☐ Comments reviewed with Performer

Evaluator Signature: ______________________ Date: __________

Read the following to the JPM performer:

****This is a time critical JPM**.** Time begins when directed by the evaluator.

TASK CONDITIONS:

A General Emergency has been declared due to an on-going un-isolable RCIC steam line break, which began 12 hours ago, with indications of fuel failure. Weather data:

- Late fall evening
- Temperature 57°F
- Upper wind speed 16.7 mph
- Lower wind speed 15.8 mph
- Upper wind direction 36.4°
- Lower wind direction 40.8°
- Stability class D

Projected off-site dose per AD-EP-ALL-0202 is 450 mRem TEDE and 3500 mRem CDE. Off-site field survey readings are 280 mRem/hour. The release is expected to drop rapidly due to the emergency depressurization of the Reactor in progress. County and State governments have been notified, and have prepared local evacuation routes.

INITIATING CUE:

A hard copy Emergency Notification Form (ENF) is being completed. Determine what Protective Action Recommendations (PARs) should be made to Off-Site Agencies and determine if Potassium Iodide should be recommended to the General Public.

****This is a time critical JPM** Time now is: __________.**

TIME	PARs

Potassium Iodide (circle one) SHOULD / SHOULD NOT be recommended to the General Public.

DUKE ENERGY

BRUNSWICK TRAINING SECTION

JOB PERFORMANCE MEASURE

LESSON TITLE: Evaluate Core Spray System Operability Test Data

LESSON NUMBER: LOT-ADM-JP-018-01

REVISION NO: 5

Lou Sosler 9/11/2015
PREPARER / DATE

John Biggs 9/15/2015
TECHNICAL REVIEWER / DATE

Brian Moschet 9/11/2015
VALIDATOR / DATE

Jerry Pierce 9/23/2015
LINE SUPERVISOR / DATE

Jim Barry 9/25/2015
TRAINING SUPERVISION APPROVAL / DATE

RELATED TASKS:

209003B201 - Perform Core Spray System Operability Test Per PT-07.2.4A (07.2.4B)

K/A REFERENCE AND IMPORTANCE RATING:

GEN 2.2.12 3.7 / 4.1
Knowledge of surveillance procedures

REFERENCES:

0PT-07.2.4A, Core Spray System Operability Test - Loop A

TOOLS AND EQUIPMENT:

None

SAFETY FUNCTION (from NUREG 1123, Rev. 2, Supp. 1):

Admin – 2. Equipment Control

SETUP INSTRUCTIONS

None

SAFETY CONSIDERATIONS:

1. None

EVALUATOR NOTES: (Do not read to performer)

1. The applicable procedure section **WILL** be provided to the performer.
2. If this is the first JPM of the JPM set, read the JPM briefing contained NUREG 1021, Appendix E, or similar to the performer.
3. This JPM will be performed on Unit 1.
4. Critical Step Basis
   a) Prevents Task Completion
   b) May Result in Equipment Damage
   c) Affects Public Health and Safety
   d) Could Result in Personal Injury
5. This is an administrative JPM designed to be administered in any setting and may be administered to multiple candidates simultaneously in a classroom setting.
6. Obtain copy of 0PT-07.2.4A and fill out Attachment 1 up to second performed by review.
7. Fill out Attachment 1 for Unit 1 Core Spray Loop A Valve Test Information Sheet. All data filled out should fall within acceptance range with the exception of:

   1-E21-F005A stoke open time should be filled out as greater than maximum value, but less than limiting value.

   1-E21-F031A stroke close time should be filled out as less than minimum value.
8. Fill out Attachment 3 for Core Spray Pump 1A Test Information. Data should be within acceptance range with exception of the following

   Incorrectly determine pump DP and record that value. Use the wrong numbers that when correctly subtracted place pump DP in the acceptance range. Use 325 psig for discharge pressure, 4 psig for running suction pressure, and 6 psig for stopped suction pressure. Subtract the stopped suction pressure to get 319 psig (325 – **6** = 319) for pump DP

   Fill out one vibration (1W A) as greater than maximum acceptance value but less than required action – in alert range.
9. Provide the filled out Attachments 1 & 3 and Acceptance Criteria from 0PT-07.2.4A to performer. Provide performer an entire copy of 0PT-07.2.4A if requested (examiner should have available copy for each examinee).

Task standards (i.e. pass/fail criteria) for each JPM step are *ITALICIZED* below the step

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Read the following to the JPM performer.

TASK CONDITIONS:

1. 0PT-07.2.4A, Core Spray System Operability Test, has just been completed on Unit One for Core Spray Loop 1A by an operator.
2. The operator who completed the test has determined all acceptance criteria are met with no exceptions as certified on Attachments 1 and 3.
3. The operator who completed the test has requested a peer check of the data that was recorded on Attachments 1 and 3 to ensure all acceptance criteria are met.
4. Another operator is checking the remainder of the procedure, other than Attachments 1 and 3 for satisfactory completion.

INITIATING CUE:

You are directed to evaluate the data recorded in 0PT-07.2.4A, Attachments 1 and 3, against the acceptance criteria of the test. Inform the CRS of the results of your review.

PERFORMANCE CHECKLIST

NOTE: Sequence is assumed unless denoted in the **Comments**.

Step 1 - Perform Take-A-Minute at job site prior to beginning task.

Examinee should cover the following questions, as deemed necessary. What are the hazards in the area? What PPE is required? Tools/PPE inspected prior to use? Energy sources secured/isolated? Is Clearance/Tag Out sufficient? What's the worst that can happen? Any ALARA concerns? Will I affect plant status? HU Tools needed?

SAT / UNSAT

TIME START: ___________

NOTE: If requested, provide copy of entire procedure 0PT-07.2.4A.

NOTE: The following steps are to evaluate data in the completed Attachments 1 and 3 and identify the following deficiencies and corrective actions:

Step 2 - Identify that Pump DP is incorrectly calculated and that actual DP is outside acceptance range and in required action range. (low)

Identified that Pump DP psid is outside acceptance range and in required action range

Declared Core Spray pump 1A INOPERABLE per procedure step 7.4.39.b.

****CRITICAL STEP**** **SAT / UNSAT**

NOTE: Testing is not required to be performed on inoperable equipment.

Step 3 - Identify that Pump vibration position 1W A is in the alert range.

Identified that Pump vibration position 1W A is in the alert range and less than the required action range / meets the acceptance criteria 5.1.3.

SAT / UNSAT

Step 4 - Identify that Valve 1-E21-F005A is outside acceptance range but within limiting time for opening.

Identified that Valve 1-E21-F005A is outside acceptance range but within limiting time for opening.

*Immediately re-test **OR** Declare 1-E21-F005A INOPERABLE per the guidance of Step 7.1.8.*

**** CRITICAL STEP **   SAT / UNSAT**

Step 5 - Identify that Valve 1-E21-F031A is outside (less than) minimum acceptance range for closing.

Identified that Valve 1-E21-F031A is outside (less than) minimum acceptance range for closing.

*Immediately re-test **OR** Declare 1-E21-F031A INOPERABLE per the guidance of Step 7.1.8.*

**** CRITICAL STEP **   SAT / UNSAT**

TERMINATING CUE: When the examinee has reviewed Attachments 1 & 3 of 0PT-07.2.4A and recommended corrective actions, this JPM is complete.

TIME COMPLETED: _________

NOTE: Comments required for any step evaluated as UNSAT.

COLLECT AND CONTROL ALL JPM EXAM MATERIALS FOR EXAM SECURITY.

Step	Critical / Not Critical	Reason
1	Note Critical	Administrative
2	Critical	Identification of out of specification pump D/P. Must declare the Pump INOP.
3	Not Critical	Identification of out of specification pump vibration
4	Critical	Valve is outside acceptance range. Re-test required or declared INOP
5	Critical	Identification of out of acceptance range for valve closing time. Re-test required or declare INOP.

REVISION SUMMARY

5	Revised to new JPM Template. Removed step to obtain a current revision of procedure as it is supplied now. Corrected faulted numbers to reflect changes to procedure acceptance criteria. Added critical step documentation table. Removed work practices criteria.
4	Revised to new JPM Template, Revision 3. Changed 0PT07.2.4A Attachments 1, 2, and 3, to Attachments 1 and 3.

Evaluate Core Spray System Operability Test Data

Validation Time: 30 Minutes (approximate)

Time Taken: ___________ Minutes

APPLICABLE METHOD OF TESTING

Performance:	Simulate	____	Actual	X	Unit:	1
Setting:	In-Plant	____	Simulator	____	Admin	X
Time Critical:	Yes	____	No	X	Time Limit	N/A
Alternate Path:	Yes	____	No	X		

EVALUATION

Performer: ____

JPM: Pass ______ Fail ______

Remedial Training Required: Yes ______ No ______

Comments: ____

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☐ Comments reviewed with Performer

Evaluator Signature: ______________________________ Date: ____________

TASK CONDITIONS:

1. 0PT-07.2.4A, Core Spray System Operability Test, has just been completed on Unit One for Core Spray Loop 1A by an operator.
2. The operator who completed the test has determined all acceptance criteria are met with no exceptions as certified on Attachments 1 and 3.
3. The operator who completed the test has requested a peer check of the data that was recorded on Attachments 1 and 3 to ensure all acceptance criteria are met.
4. Another operator is checking the remainder of the procedure, other than Attachments 1 and 3 for satisfactory completion.

INITIATING CUE:

You are directed to evaluate the data recorded in 0PT-07.2.4A, Attachments 1 and 3, against the acceptance criteria of the test. Inform the CRS of the results of your review.

ATTACHMENT 1
Page 1 of 2

Unit 1 Core Spray System (Loop A) Valve Test Information Sheet

Valve Number	Stroke Direction	Remote Position Indication, (Initials)		Stroke Time Test (sec)	Stroke Time Acceptance Criteria, (Seconds)				Fail-Safe Test, (Initials)	Full-Stroke Exercise (Initials)	Check Valve Exercise (Initials)	Valve SAT/UNSAT
					Acceptable Range							
		Stem	Indicating Lights		Minimum (≥)	Maximum (≤)	Limiting	Ref. Stroke Time				
1-E21-F015A	OPEN	RO	RO	NA	NA	NA	NA	NA	NA	NA	NA	Sat
1-E21-F015A	CLOSED	RO	RO	37.0	35.90	48.50	52.70	42.20	NA	RO	NA	Sat
1-E21-F004A	CLOSED	RO	RO	12.0	9.85	13.33	14.49	11.59	NA	NA	NA	Sat
1-E21-F005A	OPEN	RO	RO	13.7	10.01	13.55	14.73	11.78	NA	NA	NA	Sat
1-E21-F005A	CLOSED	RO	RO	12.2	9.88	13.36	14.53	11.62	NA	RO	NA	Sat
1-E21-F004A	OPEN	RO	RO	13.4	9.99	13.51	14.69	11.75	NA	RO	NA	Sat
1-E21-F001A	CLOSED	RO	RO	84.7	68.51	92.69	100.75	80.60	NA	NA	NA	Sat
1-E21-F001A	OPEN	RO	RO	81.9	72.73	98.39	106.95	85.56	NA	RO	NA	Sat
1-E21-F031A	CLOSED	RO	RO	10.8	11.00	14.80	16.10	12.90	NA	NA	NA	Sat
1-E21-F031A	OPEN	RO	RO	14.2	12.30	16.50	18.00	14.40	NA	RO	NA	Sat
1-E21-F003A	CLOSED	NA	NA	NA	NA	NA	NA	NA	NA	NA	RO	Sat
1-E21-F029A	PART OPEN	NA	NA	NA	NA	NA	NA	NA	NA	NA	RO	Sat
1-E21-F030A	PART OPEN	NA	NA	NA	NA	NA	NA	NA	NA	NA	RO	Sat
1-E21-F029A 1-E21-F030A	CLOSED	NA	NA	NA	NA	NA	NA	NA	NA	NA	RO	Sat
1-E21-F003A	OPEN	NA	NA	NA	NA	NA	NA	NA	NA	NA	RO	Sat

Unit 1 Core Spray Pump A Test Information Data Sheet

1. The lubricant level (pump running) is normal: AO

2. Calculate pump dP as follows:

   Pump discharge pressure - suction pressure (run) = pump dP

   325 - 6 = 319

NOTE

- Reference values for pump suction and discharge pressures are provided for determining the suitability of alternate test gauges, if used. ☒
- Pump stopped suction pressure should normally be between 4 and 8 psig. Values outside of this range may indicate air in the instrument line. ☒
- Should quarterly pump test data exceed the CPT limits, the pump remains OPERABLE and the test results evaluated as part of the BNP IST trending program. ☒

UNIT 1 Core Spray Pump A TEST DATA

TEST PARAMETER	ACTUAL VALUE	REFERENCE VALUE	ACCEPTANCE VALUE RANGE	ALERT RANGE		REQUIRED ACTION RANGE	
				LOW	HIGH	LOW	HIGH
Suction Press, (Stopped) psig	6.0	6.0	NA	NA	NA	NA	NA
Suction Press, (Running) psig	4.0	4.0	NA	NA	NA	NA	NA
Discharge Press. Psig	325	290.0	NA	NA	NA	NA	NA
Quarterly Pump DP psid	319	290.9	261.9 to 319.9	NA	NA	< 261.9	> 319.9
CPT Pump DP psid	N/A	290.9	270.6 to 299.6	261.9 to <270.6	NA	< 261.9	> 299.6
Flow Rate gpm	5100	4,700	NA	NA	NA	NA	NA
Vibration-vel, (in/s peak) Position 1S H	0.225	0.133	0 to 0.325	NA	> 0.325 to 0.700	NA	> 0.700
Vibration-vel, (in/s peak) Position 1W A	0.352	0.195	0 to 0.325	NA	> 0.325 to 0.700	NA	> 0.700
Vibration-vel(in/s peak) Position 1W H	0.235	0.144	0 to 0.325	NA	> 0.260 to 0.624	NA	> 0.700

Performed By, (Signature): R. Operator Date: Today Time: Now

Reviewed, IST Group, (Signature): ________ Date: ________

DUKE ENERGY
BRUNSWICK TRAINING SECTION
JOB PERFORMANCE MEASURE

LESSON TITLE: Safety Function Determination - Suppression Pool Cooling

LESSON NUMBER: SOT-OJT-JP-201-B01

REVISION NO: 1

Lou Sosler 9/11/2015
PREPARER / DATE

John Biggs 9/15/2015
TECHNICAL REVIEWER / DATE

Brian Moschet 9/11/2015
VALIDATOR / DATE

Jerry Pierce 9/23/2015
LINE SUPERVISOR / DATE

Jim Barry 9/25/2015
TRAINING SUPERVISION APPROVAL / DATE

RELATED TASKS:

341227B102,
Perform a Safety Function Determination per the Technical Requirements Manual (TRM)

K/A REFERENCE AND IMPORTANCE RATING:

Generic 2.2.223.4/4.1
Knowledge of limiting conditions for operations and safety limits

REFERENCES:

Unit Two Technical Specifications and Bases
TRM, Appendix F, Safety Function Determination Program

TOOLS AND EQUIPMENT:

None

SAFETY FUNCTION (from NUREG 1123, Rev. 2, Supp. 1):

Generic - Equipment Control

SETUP INSTRUCTIONS

None

SAFETY CONSIDERATIONS:

1. None

EVALUATOR NOTES: (Do not read to performer)

1. The applicable procedure section **WILL NOT** be provided to the trainee.
2. Prior to the first JPM of the JPM set, provide the JPM briefing contained in NUREG-1021, Appendix E, or similar briefing (for non-regulated exams) to the trainee(s).
3. Critical Step Basis
   a) Prevents Task Completion
   b) May Result in Equipment Damage
   c) Affects Public Health and Safety
   d) Could Result in Personal Injury

Read the following to the JPM performer.

TASK CONDITIONS:

1. Unit Two is operating at 100% power.
2. An Active 7 day LCO is in place for RHR Pump 2A being under clearance per Technical Specification 3.5.1 - Condition A and 3.6.2.3 - Condition A.
3. It has just been reported that valve E11-F068B (*RHR HX 2B SW DISCHARGE VALVE),* supply breaker at MCC 2XB has tripped on magnetics. The valve is currently closed.

INITIATING CUE:

The Shift Manager has directed you to perform a Safety Function Determination, and assess the Technical Specification requirements for the current plant conditions and inform him of the required Technical Specification actions.

PERFORMANCE CHECKLIST

NOTE: Sequence is assumed unless otherwise indicated, comments required for any step evaluated as UNSAT.

Step 1 - Perform take a minute at job site prior to beginning task.

Examinee should cover the following questions, as deemed necessary. What are the hazards in the area? What PPE is required? Tools/PPE inspected prior to use? Energy sources secured/isolated? Is Clearance/Tag Out sufficient? What's the worst that can happen? Any ALARA concerns? Will I affect plant status? HU Tools needed?

SAT/UNSAT

TIME START: ___________

Step 2 - Refer to LCO 3.7.1 and Bases. Determine one RHR SW Subsystem is Inoperable for reasons other than Condition A and Condition B requires restore the Inoperable RHR SW subsystem in 7 days and LCO 3.4.7 actions are required.

Determined that LCO 3.7.1 Condition B is required.

****CRITICAL STEP** SAT/UNSAT**

Step 3 - Refer to LCO 3.4.7 and determine that applicability conditions do not exist (tracking LCO condition)

Determined tracking LCO conditions exist for LCO 3.4.7.

SAT/UNSAT

Step 4 – Based on LCO 3.0.6, determine the RHR SW System supports RHR SDC (3.4.7) and RHR Suppression Pool Cooling (3.6.2.3). Refer to TRM Appendix F, Attachment 1.

Determined RHR SW Support system for RHR Suppression pool Cooling.

****CRITICAL STEP** SAT/UNSAT**

Step 5 - Refer to SFDP Attachment 4 for 3.6.2.3 and determine safety function is lost if two RHR SPC subsystems are inoperable.

Determined that loss of safety function exists for RHR SW.

****CRITICAL STEP** SAT/UNSAT**

Step 6 - Refer to Tech Spec 3.6.2.3 and Bases, determine Condition B now also applies and one loop of RHR SPC must be restored to operable within 8 hours or Condition C requires plant shutdown.

Actions entered for both RHR SPC subsystems inoperable, LCO 3.6.2.3 Condition B.

****CRITICAL STEP** SAT/UNSAT**

Step 7 - Inform SM that LCO 3.7.1, Condition B and LCO 3.6.2.3, Condition B applies.

SM notified that LCO 3.7.1, Condition B and LCO 3.6.2.3, Condition B applies

SAT/UNSAT

TERMINATING CUE: When notified of LCO Conditions, Required Actions, and Completion Times, this JPM is complete.

*** Comments required for any step evaluated as UNSAT.**

TIME STOP ____________

COLLECT AND CONTROL ALL JPM EXAM MATERIALS FOR EXAM SECURITY.

Step	Critical / Not Critical	Reason
1	Not Critical	Administrative
2	Critical	Required to complete task.
3	Not Critical	Tracking LCO only. Not required to complete task.
4	Critical	Required to complete task.
5	Critical	Required to complete task.
6	Critical	Required to complete task.
7	Critical	Required to complete task.
8	Not Critical	Communication only.

REVISION SUMMARY

1	Updated to new format. No technical information change.

Validation Time: 15 Minutes (approximate).

Time Taken: ____ Minutes

APPLICABLE METHOD OF TESTING

Performance:	Simulate	____	Actual	X	Unit:	2
Setting:	In-Plant	____	Simulator	____	Admin	X
Time Critical:	Yes	____	No	X	Time Limit	N/A
Alternate Path:	Yes	____	No	X		

EVALUATION

Performer: ______________________________

JPM: Pass ______ Fail ______

Remedial Training Required: Yes ______ No ______

Comments: ______________________________

☐ Comments reviewed with Performer

Evaluator Signature: ______________________________ Date: ______________

Read the following to the JPM performer.

TASK CONDITIONS:

1. Unit Two is operating at 100% power.

2. An Active 7 day LCO is in place for RHR Pump 2A being under clearance per Technical Specification 3.5.1 - Condition A and 3.6.2.3 - Condition A.

3. It has just been reported that valve E11-F068B (*RHR HX 2B SW DISCHARGE VALVE),* supply breaker at MCC 2XB has tripped on magnetics. The valve is currently closed.

INITIATING CUE:

The Shift Manager has directed you to perform a Safety Function Determination, and assess the Technical Specification requirements for the current plant conditions and inform him of the required Technical Specification actions.

Response: ____

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DUKE ENERGY

BRUNSWICK TRAINING SECTION

JOB PERFORMANCE MEASURE

LESSON TITLE: Determine Stay Time Limitations in High Radiation Areas

LESSON NUMBER: LOT-ADM-JP-102-A03

REVISION NO: 2

Lou Sosler 9/10/2015

PREPARER / DATE

John Biggs 9/15/2015

TECHNICAL REVIEWER / DATE

Brian Moschet 9/10/2015

VALIDATOR / DATE

Jerry Pierce 9/23/2015

LINE SUPERVISOR / DATE

Jim Barry 9/25/2015

TRAINING SUPERVISION APPROVAL / DATE

RELATED TASKS:

None

K/A REFERENCE AND IMPORTANCE RATING:

Generic 2.3.4 3.2/3.7
Knowledge of Radiation Exposure Limits under normal or emergency conditions
Generic 2.3.7 3.5/3/6
Ability to comply with radiation work permit requirements during normal and abnormal conditions

REFERENCES:

PD-RP-ALL-0001, Radiation Worker Responsibilities

TOOLS AND EQUIPMENT:

Calculator
Radiation Survey Map of 50' Reactor Building

SAFETY FUNCTION (from NUREG 1123, Rev. 2, Supp. 1):

A.3 Radiation Control

SETUP INSTRUCTIONS

None

SAFETY CONSIDERATIONS:

1. None

EVALUATOR NOTES: (Do not read to performer)

1. The applicable procedure section **WILL NOT** be provided to the trainee.
2. Prior to the first JPM of the JPM set, provide the JPM briefing contained in NUREG-1021, Appendix E, or similar briefing (for non-regulated exams) to the trainee(s).
3. This JPM may be performed on Unit 1 or Unit 2 as selected by the evaluator. Survey map must reflect correct unit.
4. Critical Step Basis
   a) Prevents Task Completion
   b) May Result in Equipment Damage
   c) Affects Public Health and Safety
   d) Could Result in Personal Injury

Read the following to the JPM performer.

TASK CONDITIONS:

Two workers will be performing a lube check and coupling alignment on the Unit 1(2) RWCU Pump 1(2)A.

Worker #1 has accumulated 800 mrem this year.
Worker #2 has accumulated 970 mrem this year.
The elevator is out of service
The following times for each worker have been estimated for performance of the job.

1.	Traversing Southeast stairwell 20' – 50' Rx Bldg:	6 minutes
2.	Staging time in access area directly outside the RWCU room:	45 minutes
3.	Staging time in area directly inside room access door:	20 minutes
4.	Work time at the "A" RWCU pump:	2.5 hours
5.	Following completion of the job, an additional 60 mrem per worker will be received during de-staging activities and transit back to the maintenance shop.	

INITIATING CUE:

Using the information above and the provided radiological survey using best ALARA practices:

1. Determine the total dose accumulated for both workers. (Assume the same task times for both workers).
2. Determine if any Brunswick administrative dose limitations will be exceeded.

PERFORMANCE CHECKLIST

NOTE: Sequence is assumed unless otherwise indicated, comments required for any step evaluated as UNSAT.

Step 1 - Perform take a minute at job site prior to beginning task.

Examinee should cover the following questions, as deemed necessary. What are the hazards in the area? What PPE is required? Tools/PPE inspected prior to use? Energy sources secured/isolated? Is Clearance/Tag Out sufficient? What's the worst that can happen? Any ALARA concerns? Will I affect plant status? HU Tools needed?

SAT/UNSAT

TIME START ___________

Step 2 - Determines dose for each worker as follows:

a. *Traversing SE stairwell 20' – 50' Rx Bldg (SE is the lowest dose stairwell)*
(6 min) 0.1 Hr X 5 mr/hr = ***0.5 mrem***
Estimate 0.5 mrem dose accumulation

****CRITICAL STEP** SAT/UNSAT**

b. *Staging time in access area directly outside the RWCU room*
(45 min) 0.75 Hr X 20 mr/hr = ***15 mrem***
Estimate 15 mrem dose accumulation

****CRITICAL STEP** SAT/UNSAT**

c. *Staging time in area directly inside room access door*
(20 min) 0.33 Hr X 80 mr/hr = ***26.7 mrem***
Estimate 26.7 mrem dose accumulation.

****CRITICAL STEP** SAT/UNSAT**

d. *Work time at the "A" RWCU pump*
2.5 Hrs X 200 mr/hr = ***500 mrem***
Estimate 500 millirem dose accumulation

****CRITICAL STEP** SAT/UNSAT**

NOTE: *An additional 60 mr will be accumulated once the job is done for de-staging activities.*

e. *Total = 0.5 + 15 + 26.7 + 500 + 60 = 602.2* ***mrem***

****CRITICAL STEP**   SAT/UNSAT**

Step 3 - Determines that neither worker would exceed the Brunswick administrative limit of 2 REM per calendar year if the estimated dose were accumulated.

Worker #1: 800 mr + 602.2 mr = 1402.2 mr (< 2R limit)
Worker #2: 970 mr + 602.2 mr = 1572.2 mr (< 2R limit)

****CRITICAL STEP**   SAT/UNSAT**

TERMINATING CUE: When the total dose for each worker has been determined and the administrative limits addressed, the JPM is complete.

TIME COMPLETED: __________

NOTE: Comments required for any step evaluated as UNSAT.

Step	Critical / Not Critical	Reason
1	Not Critical	Administrative
2a	Critical	Each calculation is critical to determine total dose for personnel safety.
2b	Critical	Each calculation is critical to determine total dose.
2c	Critical	Each calculation is critical to determine total dose.
2d	Critical	Each calculation is critical to determine total dose.
2e	Critical	Each calculation is critical to determine total dose.
3	Critical	Total calculation and knowledge of Admin Dose Limit is required to complete JPM.

REVISION SUMMARY

2	Revised to new JPM Template Revised times so that calculations are different than previous versions.
1	Revised to new JPM Template, Revision 3. No technical changes.

Validation Time: 15 Minutes (approximate)

Time Taken: ____ Minutes

APPLICABLE METHOD OF TESTING

Performance:	Simulate	X	Actual	X	Unit:	1/2
Setting:	In-Plant	____	Simulator	____	Admin	X
Time Critical:	Yes	____	No	X	Time Limit	N/A
Alternate Path:	Yes	____	No	X		

EVALUATION

Performer: ________________________________

JPM: Pass ______ Fail ______

Remedial Training Required: Yes ______ No ______

Comments: ________________________________

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☐ Comments reviewed with Performer

Evaluator Signature: ________________________ Date: ____________

TASK CONDITIONS:

Two workers will be performing a lube check and coupling alignment on the Unit 1(2) RWCU Pump 1(2)A.

Worker #1 has accumulated 800 mrem this year.
Worker #2 has accumulated 970 mrem this year.
The elevator is out of service
The following times for each worker have been estimated for performance of the job.

1. Traversing Southeast stairwell 20' – 50' Rx Bldg: 6 minutes
2. Staging time in access area directly outside the RWCU room: 45 minutes
3. Staging time in area directly inside room access door: 20 minutes
4. Work time at the "A" RWCU pump: 2.5 hours
5. Following completion of the job, an additional 60 mrem per worker will be received during de-staging activities and transit back to the maintenance shop.

INITIATING CUE:

Using the information above and the provided radiological survey using best ALARA practices:

1. Determine the total dose accumulated for both workers. (Assume the same task times for both workers).
2. Determine if any Brunswick administrative dose limitations will be exceeded.

Results: ______________________________

2RB50

BRUNSWICK NUCLEAR PLANT
RADIOLOGICAL SURVEY

SURVEY NO. 14 -175R

DATE: Today TIME: 0800	**DOSE RATE**	**CONTAMINATION**
SURVEYOR HP Tech	INTS. TYPE Teletector	β⁻ ☐ ✂☐ β⁻ ☐ ✂☐
SIGNATURE HP Tech	SERIAL NO. J117B	INST/SN
REVIEWED	β^- FACTOR	EFF.
	RX POWER 100 %	BKGD. CPM CPM
	H2 SCFM 38	

UNIT 2 BLDG RB EL 50 AREA RWCU

RWP NO. 11475-08

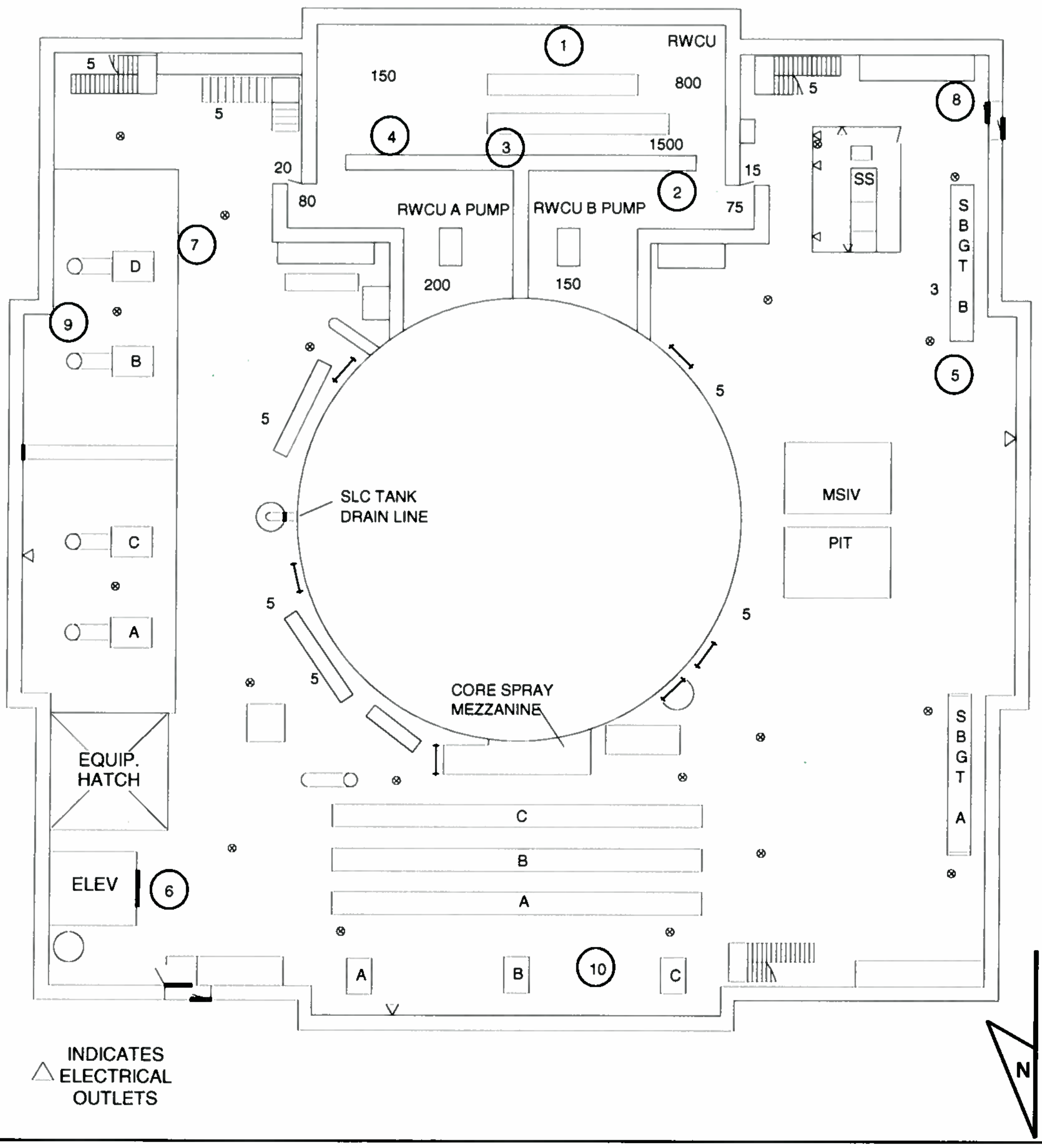
HP EXPOSURE mrem

COMPONENT OR TAG NO.

WR/JO

WORK PERFORMED Pump Lube Oil Check – Coupling Alignment

UNIT 2 REACTOR BLDG. 50'



△ INDICATES ELECTRICAL OUTLETS

SMEAR RESULTS IN DPM/100 CM β=BETA IN mRAD/Hr/100CM M=CPM/MASLIN	
NO.	RESULTS
1	3 k
2	12k
3	22k
4	18k
5	<1k
6	
7	↓
8	
9	
10	<1k

RA – RADIATION AREA
HRA – HIGH RADIATION AREA
LHRA – LOCKED HIGH RADIATION AREA
RCA – RADIATION CONTROL AREA

CA – CONTAMINATION AREA
RMA – RADIOACTIVE MATERIALS AREA
AA – AIRBORNE AREA
-x-x-x-x-x- ROPE BOUNDARY

DOSE RATES IN mREM/HR

n° Denotes NEUTRON (mREM/HR)

DUKE ENERGY

BRUNSWICK TRAINING SECTION

JOB PERFORMANCE MEASURE

NRC 2015 SIM 1

LESSON TITLE: Manual Initiation of SLC System with RWCU Isolation Failure

LESSON NUMBER: LOT-SIM-JP-005-01

REVISION NO: 3

Lou Sosler 9/10/2015
PREPARER / DATE

John Biggs 9/16/2015
TECHNICAL REVIEWER / DATE

Derek Lickett 9/10/2015
VALIDATOR / DATE

Jerry Pierce 9/24/2015
LINE SUPERVISOR / DATE

Jim Barry 9/25/2015
TRAINING SUPERVISION APPROVAL / DATE

RELATED TASKS:

211005B501
Manually Initiate Standby Liquid Control Per OP-05

K/A REFERENCE AND IMPORTANCE RATING:

211000 A4.08 4.2/4.2
Ability to operate and/or monitor in the control room: System Initiation

REFERENCES:

2EOP-01-LPC (Level Power Control)
2OP-05, Section 5.2

TOOLS AND EQUIPMENT:

None

SAFETY FUNCTION (from NUREG 1123, Rev. 2, Supp. 1):

1 Reactivity Control

SETUP INSTRUCTIONS

Recommended Initial Conditions

IC-11, 100% Power, BOC

Required Plant Conditions

Initiate malfunctions to:

1. Defeat auto scrams.
2. Insert an ATWS.
2. Fail open an SRV.

Activate malfunctions/overrides to Fail ARI and then Initiate ARI.
Activate malfunction for RWCU G31-F004 valve to fail to auto close.

Place Simulator in RUN and insert manual Reactor Scram when Suppression Pool temperature is approximately 95ºF. Carry out Immediate Operator Actions, and trip both Recirculation Pumps.

Triggers

Malfunctions

ES002F, ADS Valve E Fails Open, TRUE
RW016F, G31-F004 Failure to Auto Close, TRUE
RP005F, Auto Scram Defeat, TRUE
RP011F, ATWS 4

Overrides

ARI failed. Fail CS-5560 'AS IS' on P603.

Remotes

None

NOTE: When resetting simulator for multiple use, leave the ARI switch normal, use Switch Check Override to push through, then, after placing simulator in Run, place ARI to trip. (Otherwise a reactor scram will occur)

SAFETY CONSIDERATIONS:

1. Notify SM/CRS of JPM performance prior to commencing In-plant JPM.
2. Determine actual radiological conditions and potentially contaminated areas to achieve ALARA.
3. Ensure all electrical safety requirements are observed.
4. DO NOT OPERATE any plant equipment during performance of this JPM.

EVALUATOR NOTES: (Do not read to performer)

1. The applicable procedure section **WILL** be provided to the trainee.
2. Prior to the first JPM of the JPM set, provide the JPM briefing contained in NUREG-1021, Appendix E, or similar briefing (for non-regulated exams) to the trainee(s).
3. This JPM may be performed in the Simulator on Unit 2.
4. Critical Step Basis
   a) Prevents Task Completion
   b) May Result in Equipment Damage
   c) Affects Public Health and Safety
   d) Could Result in Personal Injury

Read the following to the JPM performer.

TASK CONDITIONS:

1. SRV E failed open and cannot be closed.
2. 0AOP-30, Safety/Relief Valve Failures, directs a Reactor Scram.
3. The Manual Reactor Scram pushbuttons have failed to initiate a scram.
4. The Unit CRS has entered ATWS Flowchart.
5. ARI has been initiated.
6. The Recirculation Pumps have been tripped.

INITIATING CUE:

You are directed to manually initiate the Standby Liquid Control (SLC) System, verify proper indications, and inform the Unit CRS when the actions are complete.

PERFORMANCE CHECKLIST

NOTE: Sequence is assumed unless denoted in the **Comments**.

Step 1 - Perform take a minute at job site prior to beginning task.

Examinee should cover the following questions, as deemed necessary. What are the hazards in the area? What PPE is required? Tools/PPE inspected prior to use? Energy sources secured/isolated? Is Clearance/Tag Out sufficient? What's the worst that can happen? Any ALARA concerns? Will I affect plant status? HU Tools needed?

SAT/UNSAT

TIME START: ___________

NOTE: The procedure may not be referenced for this task as it is considered skill of the craft, but the steps of the procedure are still required to be taken.

Step 2 - Unlock and place SLC Pumps A & B, C41-CS-S1, in the PUMP A & B RUN position.

Keylock switch in PUMP A & B RUN position.

****CRITICAL STEP**** **SAT/UNSAT**

NOTE: The following steps may be performed in any order.

Step 3 – **OBSERVE** the following indications:

Indication	**SAT**	**UNSAT**
SQUIB VALVE CONTINUITY LOSS Alarm		
SLC A/B SQUIB VALVE CONTINUITY Lights out		
SLC PUMP A red indicating light on		
SLC PUMP B red indicating light on		

SAT/UNSAT

NOTE:	The G31-F004, RWCU OUTBOARD ISOL VLV is expected to automatically close when SLC is initiated. Either RWCU Isolation Valve, G31-F004, RWCU OUTBOARD ISOL VLV, or G31-F001, RWCU INBOARD ISOL VLV, will close when the switch is taken to close.

Step 4 – Ensure RWCU Isolated.

Recognizes that RWCU did not isolate.

SAT/UNSAT

Step 5 - Closes the RWCU OUTBOARD ISOL VLV, G31-F004, or the RWCU INBOARD ISOL VLV, G31-F001, or BOTH.

RWCU OUTBOARD ISOL VLV, G31-F004, or RWCU INBOARD ISOL VLV, G31-F001, or BOTH, taken to close.

**** CRITICAL STEP **   SAT/UNSAT**

Step 6 - Ensure SLC Injection by:

Indication	**SAT**	**UNSAT**
SLC STORAGE TANK LEVEL indicating controller, *C41-LI-R601,* indicates level decreasing		
SLC PUMP DISCHARGE PRESSURE, C41-PI-R600, is greater than reactor vessel pressure.		

SAT/UNSAT

Step 7 - Informs Unit CRS that SLC has been initiated and that the RWCU Isolation valve failed to close and had to manually close the valve.

Unit CRS informed.

SAT/UNSAT

TERMINATING CUE: When the SLC pumps have been started and the RWCU Isolation valve has been closed then this JPM is complete.

TIME COMPLETED: __________

NOTE: Comments required for any step evaluated as UNSAT.

COLLECT AND CONTROL ALL JPM EXAM MATERIALS FOR EXAM SECURITY.

Step	Critical / Not Critical	Reason
1	Not Critical	Administrative
2	Critical	Required to complete task.
3	Not Critical	Observe step.
4	Not Critical	Ensure step.
5	Critical	Required to complete task.
6	Not Critical	Ensure step.
7	Not Critical	Communication to CRS.

REVISION SUMMARY

3	New JPM Format. Added Critical/Non Critical step explanation. Changed Critical Step 5 to allow closing G33-F001, RWCU INBOARD ISOL VLV.
2	Revised to new JPM Template, Revision 3. Changed from Manual Scram Failure to ATWS 4 malfunction.

Manual Initiation of SLC System with RWCU Isolation Failure.

Validation Time: 5 Minutes (approximate)

Time Taken: ________ Minutes

APPLICABLE METHOD OF TESTING

Performance:	Simulate:	____	Actual:	X	Unit:	2
Setting:	In-Plant	____	Simulator:	X	Admin:	____
Time Critical:	Yes	____	No	X	Time Limit:	____

(Ensure reference section on previous page identifies the regulation or procedure that mandates this time limit requirement)

Alternate Path:	Yes	X	No	____

EVALUATION

Performer: ______________________________

JPM Results: Pass ______ Fail ______

Remedial Training Required: Yes ______ No ______

Comments: ______________________________

☐ Comments reviewed with Performer

Evaluator Signature: ______________________________ Date: ______________

TASK CONDITIONS:

1. SRV E failed open and cannot be closed.
2. 0AOP-30, Safety/Relief Valve Failures, directs a Reactor Scram.
3. The Manual Reactor Scram pushbuttons have failed to initiate a scram.
4. The Unit CRS has entered ATWS Flowchart.
5. ARI has been initiated.
6. The Recirculation Pumps have been tripped.

INITIATING CUE:

You are directed to manually initiate the Standby Liquid Control (SLC) System, verify proper indications, and inform the Unit CRS when the actions are complete.

DUKE ENERGY

DUKE ENERGY
BRUNSWICK TRAINING SECTION
JOB PERFORMANCE MEASURE

NRC 2015 SIM 2

LESSON TITLE: RCIC Start – Steam Line Ruptures and RCIC Fails to Isolate

LESSON NUMBER: LOT-SIM-JP-016-A05

REVISION NO: 04

Lou Sosler 9/10/2015

PREPARER / DATE

John Biggs 9/15/2015

TECHNICAL REVIEWER / DATE

Derek Pickett 9/10/205

VALIDATOR / DATE

Jerry Pierce 9/24/2015

LINE SUPERVISOR / DATE

Jim Barry 9/25/2015

TRAINING SUPERVISION APPROVAL / DATE

RELATED TASKS:

217003B101, Manually Startup The RCIC System Per OP-16

K/A REFERENCE AND IMPORTANCE RATING:

217000A4.08 3.7 3.6

Ability to manually operate and/or monitor RCIC system flow

REFERENCES:

S/969 (RCIC Hard Card)

OP-16, Section 5.3

TOOLS AND EQUIPMENT:

None.

SAFETY FUNCTION (from NUREG 1123, Rev 2.):

2 - Inventory Control

SIMULATOR SETUP

Recommended Initial Conditions

Any 100% IC

Required Plant Conditions:

- RPV level <170 inches
- Inhibit ADS
- Place HPCI in PTL
- Trip RFPs

Triggers:

Auto: Q1619RRM, E51-F013 Red Light Equal to TRUE.

Malfunctions:

Event	System	Tag	Title	Value/ Ramp Rate	Activate Time (sec)	Deactivate Time (sec)
N/A	ES	ES055F	E51-F007, Failure to Auto Close	N/A	N/A	N/A
N/A	ES	ES056F	E51-F008, Failure to Auto Close	N/A	N/A	N/A
1	ES	ES025F	RCIC Stm Brk – S RHR Room	20%/ 0 sec.	40 sec	Trigger 1
N/A	ES	ES041F	RCIC Failure to Auto Start	N/A	N/A	N/A

Overrides:

None

Remotes

None

SAFETY CONSIDERATIONS:

None

EVALUATOR NOTES: (Do not read to performer)

1. The applicable procedure section **WILL** be provided to the trainee.
2. Prior to the first JPM of the JPM set, provide the JPM briefing contained in NUREG-1021, Appendix E, or similar briefing (for non-regulated exams) to the trainee(s).
3. This JPM may be performed on Unit 2.
4. Critical Step Basis
   a) Prevents Task Completion
   b) May Result in Equipment Damage
   c) Affects Public Health and Safety
   d) Could Result in Personal Injury

Read the following to the performer.

TASK CONDITIONS:

1. Both Reactor Feed Pumps have tripped and are not available.
2. Reactor level is below 170 inches.
3. HPCI is not available.

INITIATING CUE:

You are directed by the Unit CRS to place RCIC in service per the Hard Card and restore Reactor level to 170 to 200 inches. Notify the Unit CRS when all required actions are complete.

PERFORMANCE CHECKLIST

NOTE: Sequence is assumed unless otherwise indicated, comments required for any step evaluated as UNSAT.

Step 1 - Perform take a minute at job site prior to beginning task.
Examinee should cover the following questions, as deemed necessary. What are the hazards in the area? What PPE is required? Tools/PPE inspected prior to use? Energy sources secured/isolated? Is Clearance/Tag Out sufficient? What's the worst that can happen? Any ALARA concerns? Will I affect plant status? HU Tools needed?

SAT/UNSAT

TIME START: ___________

Step 2 - Ensure the following valves are open: Turbine Trip & Throttle Valve, E51-V8, and Turbine Trip & Throttle Valve Actuator, E51-V8, and Turbine Governor Valve, E51-V9.
E51-V8 (valve position) E51-V8 (actuator position) and E51-V9 are open.

SAT/UNSAT

Step 3 – Open Cooling Water Supply Valve, E51-F046.
E51-F046 is full open.

**** CRITICAL STEP ** SAT/UNSAT**

Step 4 - Start Vacuum Pump and leave switch in START.
Vacuum Pump running with switch in Start.

SAT/UNSAT

Step 5 - Open Turbine Steam Supply Valve, E51-F045.
E51-F045 is full open.

**** CRITICAL STEP ** SAT/UNSAT**

Step 6 – Open RCIC Injection Valve, E51-F013.
E51-F013 is full open.

**** CRITICAL STEP ** SAT/UNSAT**

Step 7 – Ensure that the RCIC turbine starts and comes up to speed as directed by RCIC FLOW CONTROL.

RCIC Turbine speed observed to come up to speed.

SAT/UNSAT

NOTE: When RCIC comes up to speed activate Trigger 1 to initiate steam line break.

Step 8 – Recognize the RCIC isolation and trip signal.

RCIC isolation and trip is recognized.

SAT/UNSAT

Step 9 – Recognize the failure of the RCIC Steam Supply Valves, E51-F007 and E51-F008, to close.

Failure of E51-F007 and E51-F008 to close is recognized. Operator refers to 2APP-A-03 (5-2 and 6-2).

SAT/UNSAT

Step 10 – Manually close RCIC Steam Supply Valve, E51-F007 OR RCIC Steam Supply Valve, E51-F008 OR both.

E51-F007 or E51-F008 or both are closed.

**** CRITICAL STEP ** SAT/UNSAT**

Step 11 – Notify the Unit CRS that the RCIC Steam Pipe has ruptured and that E51-F007 and E51-F008 were manually closed to isolate the leak.

Unit CRS is notified

SAT/UNSAT

TERMINATING CUE: When the RCIC Steam Line rupture is isolated and the Unit CRS is notified, this JPM is complete.

Time Completed: ____________

NOTE: Comments required for any step evaluated as UNSAT.

COLLECT AND CONTROL ALL JPM EXAM MATERIALS FOR EXAM SECURITY.

Step	Critical / Not Critical	Reason
1	Not Critical	Administrative
2	Not Critical	Not required to complete task.
3	Critical	Pump will be damaged without cooling water.
4	Not Critical	Not required to complete task.
5-6	Critical	Required to complete task.
7-9	Not Critical	Ensure and Recognize steps.
10	Critical	Actions required to complete task.
11	Not Critical	Informing CRS of results.

REVISION SUMMARY

6	New JPM format. Added Critical/Non Critical step explanation.
5	New JPM format.

Validation Time: 15 Minutes (approximate).

Time Taken: ____ Minutes

APPLICABLE METHOD OF TESTING

Performance:	Simulate	X	Actual	X	Unit:	2
Setting:	In-Plant	____	Simulator	X	Admin	___
Time Critical:	Yes	____	No	X	Time Limit	N/A
Alternate Path:	Yes	X	No	_____		

EVALUATION

Performer: ____

JPM: Pass _____ Fail _____

Remedial Training Required: Yes _____ No _____

Comments: ____

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☐ Comments reviewed with Performer

Evaluator Signature: ____________________________ Date: ____________

Read the following to the performer.

TASK CONDITIONS:

1. Both Reactor Feed Pumps have tripped and are not available.
2. Reactor level is below 170 inches.
3. HPCI is not available.

INITIATING CUE:

You are directed by the Unit CRS to place RCIC in service per the Hard Card and restore Reactor level to 170 to 200 inches. Notify the Unit CRS when all required actions are complete.

DUKE ENERGY
BRUNSWICK TRAINING SECTION
JOB PERFORMANCE MEASURE

NRC 2015 SIM 3

LESSON TITLE: Test the Main Steam Isolation Valves

LESSON NUMBER: LOT-SIM-JP-025-A04

REVISION NO: 0

Lou Sosler 9/11/2015
PREPARER / DATE

John Biggs 9/15/2015
TECHNICAL REVIEWER / DATE

Brian Moschet 9/11/2015

Derek Pickett 9/11/2015
VALIDATOR / DATE

Jerry Pierce 9/24/2015
LINE SUPERVISOR / DATE

Jim Barry 9/25/2015
TRAINING SUPERVISION APPROVAL / DATE

RELATED TASKS:

239201B201, Test Main Steam Isolation Valves per 0PT-40.2.7

K/A REFERENCE AND IMPORTANCE RATING:

239001 A4.01 4.2/4.0
Ability to manually operate and/or monitor the MSIVs in the Control Room

REFERENCES:

0PT-40.2.7, Testing of Main Steam Line Isolation Valves After Maintenance
0PT-40.2.8, Main Steam Isolation Valve Closure Test

TOOLS AND EQUIPMENT:

Stop Watch

SAFETY FUNCTION (from NUREG 1123):

3 – Pressure Control

SIMULATOR SETUP

Initial Conditions: Reactor power $\leq$50 RTP%

Place Feedwater Control Mode Select switch in 1-ELEM per 2OP-32.

SAFETY CONSIDERATIONS:

None

EVALUATOR NOTES: (Do not read to performer)

1. The applicable procedure section **WILL** be provided to the trainee.
2. Prior to the first JPM of the JPM set, provide the JPM briefing contained in NUREG-1021, Appendix E, or similar briefing (for non-regulated exams) to the trainee(s).
3. This JPM may be performed on Unit 2.
4. Critical Step Basis
   1. Prevents Task Completion
   2. May Result in Equipment Damage
   3. Affects Public Health and Safety
   4. Could Result in Personal Injury
5. **Provide copy of 0PT-40.2.7, Acceptance Criteria, Prerequisites, and Section 6.2, Post Maintenance Testing B21-F022A (Inboard MSIV A Vlv)**

Read the following to the JPM performer.

TASK CONDITIONS:

1. Unit Two startup is in progress following a forced outage to repair MSIV 2B21-F022A, Inboard MSIV A valve.
2. Conditions are such that steam flow can be stopped in the main steam line of the MSIVs being tested.
3. No other tests or maintenance activities are in progress that could provide a half scram signal to the RPS logic.
4. Another operator has placed Feedwater Control Mode Select switch in 1-ELEM per 2OP-32, Condensate and Feedwater System Operating Procedure.

INITIATING CUE:

You are directed by the Unit CRS to perform 0PT-40.2.7, Testing of Main Steam Isolation Valve after Maintenance, for MSIV 2B21-F022A, Inboard MSIV A Valve.

PERFORMANCE CHECKLIST

NOTE: Sequence is assumed unless otherwise indicated, comments required for any step evaluated as UNSAT.

Step 1 - Perform take a minute at job site prior to beginning task.

Examinee should cover the following questions, as deemed necessary. What are the hazards in the area? What PPE is required? Tools/PPE inspected prior to use? Energy sources secured/isolated? Is Clearance/Tag Out sufficient? What's the worst that can happen? Any ALARA concerns? Will I affect plant status? HU Tools needed?

SAT/UNSAT

TIME START: ___________

NOTE: The examinee should be provided a copy of 0PT-40.2.7, Testing of Main Steam Isolation Valve after Maintenance, and given time to review and pre-mark appropriate sections.

PROMPT If asked, a Reactivity Management Team is in place for this test.

Step 2 – Confirm Reactor power is less than 55% RTP

Confirmed power less than 55% RTP.

SAT/UNSAT

Step 3 – Confirm all MSIVs are open.

Confirmed all MSIVs are open.

SAT/UNSAT

Step 4 – Confirm Reactor Recirculation system is **NOT** in single loop operation (SLO)

Confirmed Reactor Recirculation system not in single loop.

SAT/UNSAT

NOTE: Have stop watch ready to give to Examinee.

Step 5 – Obtain a stopwatch and record calibration information.

Stop watch obtained and calibration information recorded.

SAT/UNSAT

Step 6 – Ensure the following annunciators are clear:

- A-05, 4-6, Main Steam Isol Vlv Not Full Open
- A-05, 1-7, Reactor Auto Scram Sys A
- A-05, 2-7, Reactor Auto Scram Sys B

Annunciators confirmed to be clear.

SAT/UNSAT

PROMPT It is NOT required to stop steam flow in Main Steam Line A.

PROMPT It IS required to perform slow closure (spring closure) test of B21-F022A.

Step 7 - **Depress** and **hold** B21-F022A (Inboard MSIV A Test) pushbutton until the valve goes CLOSED, approximately 45-60 seconds.

B21-F022A (Inboard MSIV A Test) pushbutton depressed and held until the valve is CLOSED, green light on, red light off.

****CRITICAL STEP** SAT/UNSAT**

Step 8 - **Release** B21-F022A (Inboard MSIV A Test) pushbutton and **confirm** the valve goes OPEN

Pushbutton for B21-F022A released and valve open confirmed.

****CRITICAL STEP** SAT/UNSAT**

PROMPT If asked, stroke time testing is required.

Step 9 - **Perform** stroke time test as follows:

a. **Ensure** B21-F022A (Inboard MSIV A Vlv) OPEN.

B21-F022A verified open.

SAT/UNSAT

b. **Close** B21-F022A (Inboard MSIV A Vlv) utilizing the pistol grip switch.

B21-F022A pistol grip switch taken to close.

****CRITICAL STEP** SAT/UNSAT**

c. **Record** stroke time:
Stroke time recorded.

SAT/UNSAT

NOTE: Section 6.2, Step 4.c is previous step.

d. **Enter** the measured stroke time from Section 6.2 Step 4.c and **calculate** the corrected stroke time (Stroke Time from Section 6.2, Step 4.c X 1.1 = Corrected Stroke Time)
Corrected stroke time calculated

****CRITICAL STEP** SAT/UNSAT**

e. **Record** corrected stroke time on Attachment 1 or Attachment 2
Corrected Stroke Time recorded on Attachment 2

SAT/UNSAT

PROMPT If asked, it is required by plant conditions to open B21-F022A.

Step 10 – **IF** required by plant conditions, **THEN open** B21-F022A (Inboard MSIV A Vlv).
B21-F022A pistol grip switch taken to open.

SAT/UNSAT

NOTE: Step 6.2.7 is N/A

PROMPT Inform Examinee that another operator will complete the Restoration section of the PT.

TERMINATING CUE: When the 2B21-F022A, Inboard MSIV A Valve, has been re-opened after testing, this JPM is complete.

TIME COMPLETED: ___________

COLLECT AND CONTROL ALL JPM EXAM MATERIALS FOR EXAM SECURITY.

Step	Critical / Not Critical	Reason
1	Not Critical	Administrative
2-6	Not Critical	Verification of initial conditions and pre-requisites.
7-8	Critical	Required actions to complete the test.
9a	Not Critical	Verification step.
9b	Critical	Action required to complete the test.
9c	Not Critical	Recording time not critical to test completion.
9d	Critical	Calculation of Corrected Stroke Time required to complete task.
9e	Not Critical	Recording required information.
10	Not Critical	Re-opening valve not required to obtain results.

REVISION SUMMARY

0	New JPM.

Validation Time: 10 Minutes (approximate).

Time Taken: ____ Minutes

APPLICABLE METHOD OF TESTING

Performance:	Simulate	____	Actual	X	Unit:	2
Setting:	In-Plant	____	Simulator	X	Admin	___
Time Critical:	Yes	____	No	X	Time Limit	N/A
Alternate Path:	Yes	____	No	X		

EVALUATION

Performer: ________________________________

JPM: Pass _____ Fail _____

Remedial Training Required: Yes _____ No _____

Comments: ________________________________

☐ Comments reviewed with Performer

Evaluator Signature: ________________________ Date: ____________

TASK CONDITIONS:

1. Unit Two startup is in progress following a forced outage to repair MSIV 2B21-F022A, Inboard MSIV A valve.
2. Conditions are such that steam flow can be stopped in the main steam line of the MSIVs being tested.
3. No other tests or maintenance activities are in progress that could provide a half scram signal to the RPS logic.
4. Another operator has placed Feedwater Control Mode Select switch in 1-ELEM per 2OP-32, Condensate and Feedwater System Operating Procedure.

INITIATING CUE:

You are directed by the Unit CRS to perform 2PT-40.2.7, Testing of Main Steam Isolation Valve after Maintenance, for MSIV 2B21-F022A, Inboard MSIV A Valve.

1.0 PURPOSE

This test demonstrates the OPERABILITY of the main steam isolation valves, MSIVs, after maintenance and provides direction for the slow closure test of MSIVs.

2.0 SCOPE

1. This test demonstrates each MSIVs ability to full stroke within the stroke times specified in Unit 1(Unit 2) Technical Specifications SR 3.6.1.3.5. This satisfies the IST requirement in Technical Specification 5.5.6.

2. This test checks the MSIV Slow Closure function described in UFSAR Sections 5.4.5 and 7.3.1.1.5.

3. This test does **NOT** provide instructions for stroke adjustments subsequent to testing.

3.0 PRECAUTIONS AND LIMITATIONS

1. When isolating and unisolating a steam line in MODE 1 or 2 a small pressure change may occur causing a reactivity change. This reactivity change is classified as a Reactivity Manipulation (R2) per OPS-NGGC-1306, Reactivity Management Program.

2. If this test is being performed in MODE 1, the RPS System will receive a partial trip signal that will **NOT** be annunciated as long as the remaining MSIVs are in the open position.

3. Annunciator A-05, 4-6, Main Steam Isol Vlv Not Full Open, may alarm when a main steam line is isolated. This annunciator is received only when two or more MSIVs are closed.

4. Operation with both MSIVs closed in a main steam line is minimized to reduce the severity of differential pressure transients when reopening the Outboard MSIV. The section of pipe between the inboard and outboard MSIV will depressurize as it cools down or if any steam leaks are present (such as stem packing leak).

3.0 PRECAUTIONS AND LIMITATIONS (continued)

5. An administrative band of 3.6 seconds to 4.4 seconds is applied when in MODE 2 or 3 due to temperature affects on stroke time. This is an administrative limit, and is **NOT** controlled by the IST program. More detail concerning these administrative limits is available in Section 8.7 Miscellaneous Document 3, EC# 86807, Evaluation of MSIV Stroke Time Criteria. If the corrected stroke time is satisfactory, but outside the Administrative range, it is to be adjusted to within the Administrative range per the applicable CHANNEL CALIBRATION.

4.0 ACCEPTANCE CRITERIA

This test may be considered satisfactory with the successful completion of this procedure.

NOTE

This test demonstrates the slow closure design base function of the MSIVs described in UFSAR Sections 5.4.5 and 7.3.1.1.5. This function requires MSIVs to close on spring pressure alone. Slow closure is **NOT** a safety function. .... ☐

1. Slow Closure Test

   a. When an MSIV is given a close signal from the Control Room test pushbutton, the valve goes to the CLOSED position.

NOTE

- The Valve Stroke time test satisfies Unit 1(Unit 2) Technical Specifications SR 3.6.1.3.5 and partially satisfies the IST requirement in Technical Specification 5.5.6. .... ☐
- Stroke time is measured from the time the control switch is repositioned to the time the valve is fully stroked by light indication.. .... ☐
- For MSIVs, the measured stroke times are multiplied by a correction factor of 1.1 to compensate for the position settings of the indicating light sensors of 10% and 100% open. .... ☐

2. Valve Stroke Time

   a. Corrected stroke times are within the Limiting range as specified by the minimum and maximum stroke times shown on Attachment 1, Unit 1 Nuclear Steam Supply System Valves Data or Attachment 2, Unit 2 Nuclear Steam Supply System Valves Data

4.0 ACCEPTANCE CRITERIA (continued)

b. For tests where the corrected stroke time of the valve is less than the minimum or greater than the maximum Limiting stroke time or the valve disc or stem fail to exhibit the required change of position, the valve shall immediately be declared INOPERABLE.

NOTE

This test partially satisfies the IST requirement in Technical Specification 5.5.6. .... ☐

3. Valve Fail-Safe Testing

   a. The fail-safe test is considered satisfactory when the control switch is placed in the CLOSED position for fail-closed valves or OPEN position for fail-open valves, and the valve changes position in response to control switch movement.

5.0 PREREQUISITES

1. **Confirm** Reactor power is less than 55% RTP. .... _______

2. **Confirm** conditions are such that steam flow can be stopped in the main steam line of the MSIV being tested or <u>**NO**</u> steam flow exists. .... _______

3. <u>**IF**</u> unit is in MODE 1, <u>**THEN**</u> **confirm** the following:

   - <u>**NO**</u> other tests or maintenance activities are in progress that could provide a half scram signal to the RPS logic. .... _______

   - All main steam isolation valves are OPEN. .... _______

4. **Confirm** the Reactor Recirculation system is <u>**NOT**</u> in single loop operation (SLO). .... _______

5. **Obtain** a stopwatch and **record** information:.. _______

TEST EQUIPMENT			
Item	**ID No.**	**Cal Date**	**Cal Due Date**
Stopwatch			

6.0 INSTRUCTIONS

6.1 <u>General</u>

1. **Request** permission from the Unit CRS to perform this test. .... _______

2. **Ensure** all prerequisites in Section 5.0 are met. .... _______

3. **Ensure** Feedwater Control Mode Select switch, in 1-ELEM per <u>1OP-32</u> (<u>2OP-32</u>) Condensate and Feedwater System Operating Procedure. .... _______

4. <u>**IF AT ANY TIME**</u> while performing this test in MODE 1, annunciator A-05, 4-6, Main Steam Isol Vlv Not Full Open, is received, <u>**THEN**</u> **suspend** this test and **determine** its cause. .... _______

NOTE

When isolating and unisolating a steam line in MODE 1 or 2 a small pressure change may occur causing a reactivity change. This reactivity change is classified as a Reactivity Manipulation (R2) per OPS-NGGC-1306, Reactivity Management Program. .... ☐

6.2 Post Maintenance Testing B21-F022A (Inboard MSIV A Vlv)

1. **IF** unit is in MODE 1,
THEN ensure the following annunciators CLEAR:

   - A-05, 4-6, Main Steam Isol Vlv Not Full Open .... _______
   - A-05, 1-7, Reactor Auto Scram Sys A .... _______
   - A-05, 2-7, Reactor Auto Scram Sys B .... _______

CAUTION

When this test is performed in MODE 1, reactor pressure, power level, and steam flow are monitored while closing the MSIVs. Any deviation from expected plant response is cause for suspension of this test and notification of the Unit CRS prior to proceeding. .... ☐

BEGIN R.M. LEVEL R2 REACTIVITY EVOLUTION

2. **IF** it is required to stop steam flow in Main Steam Line A,
THEN perform the following:

   a. **Depress** and **hold** B21-F028A (Outboard MSIV A Test) pushbutton until the valve is CLOSED. .... _______

   b. **Place** pistol grip switch for B21-F028A (Outboard MSIV A Vlv) in CLOSE. .... _______

3. **IF** performing slow closure (spring closure) test of B21-F022A (Inboard MSIV A Vlv),
THEN perform the following:

   a. **Depress** and **hold** B21-F022A (Inboard MSIV A Test) pushbutton until the valve goes CLOSED, approximately 45-60 seconds.. _______

   b. **Release** B21-F022A (Inboard MSIV A Test) pushbutton and **confirm** the valve goes OPEN... _______

6.2 Post Maintenance Testing B21-F022A (Inboard MSIV A Vlv) (continued)

CAUTION

Operation with both MSIVs closed in a main steam line is minimized to reduce the severity of differential pressure transients when reopening the Outboard MSIV. .... ☐

4. **Perform** stroke time test as follows:

   a. **Ensure** B21-F022A (Inboard MSIV A Vlv) OPEN. ....______

   b. **Close** B21-F022A (Inboard MSIV A Vlv) utilizing the pistol grip switch. ....______

   c. **Record** stroke time: ....______

   Stroke Time ________ Seconds

   d. **Enter** the measured stroke time from Section 6.2 Step 4.c and **calculate** the corrected stroke time. ....______

   ______ IV

   ________ seconds X 1.1 = ________ seconds

   Stroke Time from Section 6.2 Step 4.c — Corrected Stroke Time

   e. **Record** corrected stroke time on Attachment 1 or Attachment 2. ...______

5. **IF** B21-F028A (Outboard MSIV A Vlv) pistol grip switch was placed in CLOSE in Section 6.2 Step 2,
   THEN **place** pistol grip switch in OPEN and **confirm** the valve goes OPEN...______

6. **IF** required by plant conditions,
   THEN **open** B21-F022A (Inboard MSIV A Vlv). ....______

END R.M. LEVEL R2 REACTIVITY EVOLUTION

6.2 Post Maintenance Testing B21-F022A (Inboard MSIV A Vlv) (continued)

7. IF all the following conditions are met:

- Corrected stroke time is within the Limiting range
- Corrected stroke time is outside the Administrative range
- The unit is in MODE 2 or 3 with the Drywell/MSIV Pit **NOT** accessible,

THEN generate an CR to adjust the valve stroke time to within the Administrative range during the next outage... ______

DUKE ENERGY
BRUNSWICK TRAINING SECTION
JOB PERFORMANCE MEASURE

NRC 2015 SIM 4

LESSON TITLE: Shifting Stator Cooling Pumps – Pump Trip

LESSON NUMBER: LOT-SIM-JP-027.2-01

REVISION NO: 3

Lou Sosler 9/18/2015
PREPARER / DATE

John Biggs 9/18/2015
TECHNICAL REVIEWER / DATE

Brian Moschet 9/21/2015
VALIDATOR / DATE

Jerry Pierce 9/24/2015
LINE SUPERVISOR / DATE

Jim Barry 9/25/2015
TRAINING SUPERVISION APPROVAL / DATE

RELATED TASKS:

253 002 B1 01, Startup The Generator Stator Cooling System Per OP-27.2

K/A REFERENCE AND IMPORTANCE RATING:

245000 A4.03 2.7/2.8
Ability to manually operate and/or monitor in the control room: Stator water cooling pumps

REFERENCES:

2OP-27.2 Section 8.6
2APP UA-02 4-9

TOOLS AND EQUIPMENT:

None

SAFETY FUNCTION (from NUREG 1123, Rev 2.):

4 – Heat Removal from Reactor Core (Main Turbine Generator and Auxiliary Systems)

SIMULATOR SETUP:

A. Initial Conditions:

Any power IC
Rx. Pwr. Any
Core Age Any

Required Plant Conditions:

2A Stator Cooling Pump running and 2B Stator Cooling Pump in standby

B. Triggers

None

C. Malfunctions

XY008F Stator Cooling Pump B Sheared Shaft (Active)

D. Overrides

Annunciator UA-02 1-9, Loss of Stat Coolant Trip Ckt Ener, OFF

E. Special Instructions

Load Malfunctions

SAFETY CONSIDERATIONS:

None

EVALUATOR NOTES: (Do not read to performer)

1. The applicable procedure section **WILL** be provided to the trainee.
2. Prior to the first JPM of the JPM set, provide the JPM briefing contained in NUREG-1021, Appendix E, or similar briefing (for non-regulated exams) to the trainee(s).
3. This JPM may be performed on Unit 2.
4. Critical Step Basis
   a) Prevents Task Completion
   b) May Result in Equipment Damage
   c) Affects Public Health and Safety
   d) Could Result in Personal Injury

Read the following to the JPM performer.

TASK CONDITIONS:

1. 2A Stator Cooling Pump is in operation.
2. All applicable prerequisites in OP-27.2, Section 5.0 are met.

INITIATING CUE:

You are directed to start 2B Stator Cooling pump and secure 2A Stator Cooling pump so that routine maintenance may be performed on 2A Stator Cooling Pump.

Inform the Unit CRS when the pump swap is complete.

PERFORMANCE CHECKLIST

NOTE: Sequence is assumed unless otherwise indicated, comments required for any step evaluated as UNSAT.

Step 1 - Perform take a minute at job site prior to beginning task.

Examinee should cover the following questions, as deemed necessary. What are the hazards in the area? What PPE is required? Tools/PPE inspected prior to use? Energy sources secured/isolated? Is Clearance/Tag Out sufficient? What's the worst that can happen? Any ALARA concerns? Will I affect plant status? HU Tools needed?

SAT/UNSAT

TIME START: ___________

PROMPT: When AO is directed to close GSC-Y-34 to isolate GSC-63-P79, report that the valve has been closed.

Step 2 – Direct AO to close GSC-63-P79 Instrument Isolation Valve, GSC-Y-34.

AO directed to close GSC-Y-34.

SAT/UNSAT

PROMPT: When AO is directed to monitor Stator Cooling System pressure on GSC-PI-YGA-2 report that the AO is monitoring system pressure. Pressure is currently 46 psig.

Step 3 – Direct AO to monitor Stator Cooling System pressure on GSA-PI-YGA-2.

AO directed to monitor Stator Cooling System pressure on GSA-PI-YGA-2.

SAT/UNSAT

Step 4 – Start 2B Stator Cooling pump.

Rotates 2B Stator Cooling pump switch to ON without pausing in OFF and observes the red light illuminates and the green light goes off.

****CRITICAL STEP** SAT/UNSAT**

Step 5 – Acknowledges STATOR COOL RESERVE PUMP RUNNING (UA-02 4-9) alarm.

Silences and reports to CRS Stator Cool Reserve Pump Running in alarm.

SAT/UNSAT

PROMPT: After 2A Stator Cooling Pump is off, inform performer (as AO) that system pressure indicated on GSA-PI-YGA-2 is 30 psig and stable.

Step 6 – Stop 2AStator Cooling pump.

Places 2A Stator Cooling pump control switch in OFF and observes green light illuminates and red light goes off.

****CRITICAL STEP** SAT/UNSAT**

Step 7 – IMMEDIATELY START 2A Stator Cooling pump.

Places 2A Stator Cooling pump switch to ON and observes the red light illuminates and the green light goes off.

****CRITICAL STEP** SAT/UNSAT**

PROMPT: When asked, system pressure indicated on GSC-PI-YGA-2 is 47 psig.

Step 8 – Direct AO to check Stator Cooling System pressure is between 42 and 50 psig as indicated on GSC-PI-YGA-2.

Directs AO to check Stator Cooling System pressure is between 42 and 50 psig as indicated on GSC-PI-YGA-2.

SAT/UNSAT

PROMPT: Inform performer:
(1) 2B Stator Cooling Pump has been evaluated to have failed.
2) Place 2B pump in OFF, WCC will take care of the status control for the pump.
3) Restore the rest of the system back into standby alignment.

NOTE: Status of 2B Stator Cooling Pump is not critical to completion of the JPM. 2B Stator Cooling Pump may be left in OFF or AUTO, but should be secured.

Step 9 – Stops 2B Stator Cooling Pump and then places the control switch in Auto.

2B Stator Cooling Pump is stopped by placing control switch in OFF.

SAT/UNSAT

Step 10 – Confirm STATOR COOL RESERVE PUMP RUNNING (UA-02 4-9) clears.
Reports to CRS, Stator Cool Reserve Pump Running alarm is clear.

SAT/UNSAT

PROMPT: When directed to open GSC-Y-34 report valve is open. This step is critical for restoring protection to the generator in case of Stator Cooling System failure.

Step 11 – Direct AO to open GSC-Y-34, GSC-63-P79 Instrument Isolation Valve.
AO directed to open GSC-Y-34.

****CRITICAL STEP** SAT/UNSAT**

TERMINATING CUE: When 2A Stator Cooling pump is restarted and the GSC0Y-34 valve is directed to be opened, this JPM is complete.

TIME COMPLETE: ____________

COLLECT AND CONTROL ALL JPM EXAM MATERIALS FOR EXAM SECURITY.

Step	Critical / Not Critical	Reason
1	Not Critical	Administrative
2-3	Not Critical	Task can be accomplished without these steps.
4	Critical	Required to complete task.
5	Not Critical	Acknowledge alarm
6-7	Critical	Required to complete task.
8-10	Not Critical	Task can be accomplished without these steps.
11	Critical	Restores protection to the generator in the event of Stator Cooling System failure.

REVISION SUMMARY

3	New JPM format. Added Critical/Non Critical step explanation.
2	Updated to new JPM template.

Validation Time: 10 Minutes (approximate).

Time Taken: ____ Minutes

APPLICABLE METHOD OF TESTING

Performance:	Simulate	X	Actual	X	Unit:	2
Setting:	In-Plant	____	Simulator	X	Admin	____
Time Critical:	Yes	____	No	X	Time Limit	N/A
Alternate Path:	Yes	X	No	____		

EVALUATION

Performer: ____

JPM: Pass ______ Fail ______

Remedial Training Required: Yes ______ No ______

Comments: ____

____

____

____

____

____

____

☐ Comments reviewed with Performer

Evaluator Signature: ______________________________ Date: ______________

TASK CONDITIONS:

1. 2A Stator Cooling Pump is in operation.
2. All applicable prerequisites in OP-27.2, Section 5.0 are met.

INITIATING CUE:

You are directed to start 2B Stator Cooling pump and secure 2A Stator Cooling pump so that routine maintenance may be performed on 2A Stator Cooling Pump.

Inform the Unit CRS when the pump swap is complete.

DUKE ENERGY
BRUNSWICK TRAINING SECTION
JOB PERFORMANCE MEASURE

NRC 2015 SIM 5

LESSON TITLE: **Secure Recirculation Pump IAW AOP-14 - THI**

LESSON NUMBER: **LOT-SIM-JP-302D-01**

REVISION NO: **0**

Lou Sosler 9/10/2015
PREPARER / DATE

John Biggs 9/15/2015
TECHNICAL REVIEWER / DATE

Drek Picket 9/10/2015

Thomas Baker 9/10/2015
VALIDATOR / DATE

Jerry Pierce 9/24/2015
LINE SUPERVISOR / DATE

Jim Barry 9/25/2015
TRAINING SUPERVISION APPROVAL / DATE

RELATED TASKS:

223603B401 Respond to High Primary Containment Pressure per AOP-14.0

K/A REFERENCE AND IMPORTANCE RATING:

295024 High Drywell Pressure

EA1	Ability to operate and/or monitor the following as they apply to HIGH DRYWELL PRESSURE	
1.21	Recirculation System	3.4/3.8

REFERENCES:

0AOP-14.0

TOOLS AND EQUIPMENT:

None

SAFETY FUNCTION (from NUREG 1123, Rev. 2, Supp. 1):

5 - Containment Integrity (IAW ES-401-1)

SETUP INSTRUCTIONS

Insert seal failure:

RC007F (Seal #1, 100%)
RC009F (Seal #2, 5%)

Set up THI on Trigger 1: NB016F- IP-STDY

SAFETY CONSIDERATIONS:

None

EVALUATOR NOTES: (Do not read to performer)

1. The applicable procedure section **WILL** be provided to the trainee.
2. Prior to the first JPM of the JPM set, provide the JPM briefing contained in NUREG-1021, Appendix E, or similar briefing (for non-regulated exams) to the trainee(s).
3. This JPM may be performed on Unit 2 Simulator.
4. Critical Step Basis
   a) Prevents Task Completion
   b) May Result in Equipment Damage
   c) Affects Public Health and Safety
   d) Could Result in Personal Injury

Read the following to the JPM performer.

TASK CONDITIONS:

1. Reactor Recirculation Pump A seals have been determined to be failing.
2. 0AOP-14.0 has been entered and is being executed.

INITIATING CUE:

You are directed by the CRS to isolate the Reactor Recirculation Pump A IAW 0AOP-14.0 Step 4.2.7.8.

PERFORMANCE CHECKLIST

NOTE: Sequence is assumed unless otherwise indicated, comments required for any step evaluated as UNSAT.

Step 1 - Perform take a minute at job site prior to beginning task.

Examinee should cover the following questions, as deemed necessary. What are the hazards in the area? What PPE is required? Tools/PPE inspected prior to use? Energy sources secured/isolated? Is Clearance/Tag Out sufficient? What's the worst that can happen? Any ALARA concerns? Will I affect plant status? HU Tools needed?

TIME START ___________

NOTE: Steps 2 through 7 may be performed concurrently.

Step 2 – Stop Recirculation Pump A by depressing the Recirc Pump Emerg Stop pushbutton.

Emergency Stop Pushbutton for Recirc Pump A depressed.

****CRITICAL STEP** SAT/UNSAT**

Step 3 – Confirm Recirc VFD 2A 4KV Supply Bkr, is OPEN.

Observes Recirc VFD 4KV Supply Bkr is open indicated by green light on and red light off.

SAT/UNSAT

Step 4 – Confirm affected recirc pump Speed Demand is 0.00 AND pump speed is lowering.

Observes Recirc Pump A Speed Demand is 0.00 and Recirc Pump speed is lowering on P601 panel.

SAT/UNSAT

Step 5 – Close B32-F031A (Pump A Disch Vlv).

Recirc Pump A discharge valve is closed by rotating the control switch in the counterclockwise direction and indicated by the green light on and red light off.

****CRITICAL STEP** SAT/UNSAT**

NOTE: When the B32-F031A valve is full closed the Thermal Hydraulic Instability malfunction will initiate.

Step 6 – Close B32-V22 (Seal Injection Vlv).

Recirc Pump A Seal Injection valve is closed by rotating the control switch in the counterclockwise direction and indicated by the green light on and red light off.

****CRITICAL STEP** SAT/UNSAT**

Step 7 – Close B32-F032A (Disch Bypass Vlv).

Recirc Pump A discharge bypass valve is closed by rotating the control switch in the counterclockwise direction and indicated by the green light on and red light off.

****CRITICAL STEP** SAT/UNSAT**

Step 8 – Close B32-F023A (Pump A Suction Vlv).

Recirc Pump A suction valve is closed by rotating the control switch in the counterclockwise direction and indicated by the green light on and red light off.

****CRITICAL STEP** SAT/UNSAT**

NOTE: The examinee should determine that a reactor scram is required due to thermal hydraulic instabilities, the examinee may continue on to the OP until it is determined that a reactor scram is required.

Step 9 – Determines thermal hydraulic instabilities.

Observes indications of THI (Reactor period fluctuations, LPRM/APRM fluctuations, alarm?. Setup and run...............

SAT/UNSAT

Step 10 – Inserts a reactor manual scram by depressing both manual scram pushbuttons.

Depresses both manual scram pushbuttons on the P601 panel and observes all rods insert.

****CRITICAL STEP** SAT/UNSAT**

TERMINATING CUE: When both manual scram pushbuttons have been depressed this, this JPM is complete.

TIME COMPLETE: ____________

COLLECT AND CONTROL ALL JPM EXAM MATERIALS FOR EXAM SECURITY.

Step	Critical / Not Critical	Reason
1	Not Critical	Administrative
2	Critical	Required to complete task.
3-4	Not Critical	Confirm steps.
5-8	Critical	Required to complete task.
9	Not Critical	Not measurable.
10	Critical	Required to complete task.

REVISION SUMMARY

0	New JPM.

Validation Time: 10 Minutes (approximate).

Time Taken: ____ Minutes

APPLICABLE METHOD OF TESTING

Performance:	Simulate	____	Actual	X	Unit:	2
Setting:	In-Plant	____	Simulator	X	Admin	___
Time Critical:	Yes	____	No	X	Time Limit	N/A
Alternate Path:	Yes	X	No	_____		

EVALUATION

Performer: ____

JPM: Pass _____ Fail _____

Remedial Training Required: Yes _____ No _____

Comments: ____

____

____

____

____

____

____

☐ Comments reviewed with Performer

Evaluator Signature: ________________________________ Date: ______________

TASK CONDITIONS:

1. Reactor Recirculation Pump A seals have been determined to be failing.
2. 0AOP-14.0 has been entered and is being executed.

INITIATING CUE:

You are directed by the CRS to isolate the Reactor Recirculation Pump A IAW 0AOP-14.0 Step 4.2.7.8.

DUKE ENERGY

BRUNSWICK TRAINING SECTION

JOB PERFORMANCE MEASURE

NRC 2015 SIM 6

LESSON TITLE: **Manual Transfer of Bus E3 from the Normal Feeder to the DG3**

LESSON NUMBER: **LOT-SIM-JP-050-B01**

REVISION NO: **8**

Lou Sosler 9/10/2015

PREPARER / DATE

John Biggs 9/15/2015

TECHNICAL REVIEWER / DATE

Thomas Baker 9/10/2015

VALIDATOR / DATE

Jerry Pierce 9/24/2015

LINE SUPERVISOR / DATE

Jim Barry 9/25/2015

TRAINING SUPERVISION APPROVAL / DATE

RELATED TASKS:

262016B101
Complete a Control Room Manual Transfer of Emergency Bus Supply from Normal Feeder to Diesel Generator per OP-50.1

K/A REFERENCE AND IMPORTANCE RATING:

264000 A4.04 3.7/3.7
Ability to manually operate and/or monitor in the control room: Manual start, loading, and stopping of emergency generator.

REFERENCES:

0OP-50.1, Emergency Diesel Generator Power System Operating Procedure

TOOLS AND EQUIPMENT:

None

SAFETY FUNCTION (from NUREG 1123, Rev. 2, Supp. 1):

6 - Electrical Distribution

SETUP INSTRUCTIONS

A. Initial Conditions:

Any IC w/o DG auto start signal

- Start DG3 in Control Room Manual
- Place DG3 Output Breaker (AI5) Synch Switch to ON.
- Adjust DG3 output voltage to less than E3 bus voltage
- Adjust DG frequency so that the Synch Scope is rotating slowly in the SLOW direction.
- Place DG3 Output Breaker (AI5) Synch Switch to OFF.
- Place RHRSW Pump A in service @ 4000gpm flow through the RHR heat exchanger. Start 2A NSW Pump. (Running these pumps places 1600 KW on Bus E3)

B. Malfunctions:
None

C. Overrides:
None

D. Triggers
None

SAFETY CONSIDERATIONS:

None

EVALUATOR NOTES: (Do not read to performer)

1. The applicable procedure section **WILL** be provided to the trainee.
2. Prior to the first JPM of the JPM set, provide the JPM briefing contained in NUREG-1021, Appendix E, or similar briefing (for non-regulated exams) to the trainee(s).
3. This JPM may be performed on Unit 2.
4. Critical Step Basis
   a) Prevents Task Completion
   b) May Result in Equipment Damage
   c) Affects Public Health and Safety
   d) Could Result in Personal Injury

Read the following to the JPM performer.

TASK CONDITIONS:

1. E3 is energized from BOP Bus 2D.
2. DG3 is running in Control Room Manual IAW 0OP-39, Diesel Generator Operating Procedure
3. The Load Dispatcher has been notified that E3 load will be shifted to DG3.
4. An AO is stationed at DG3.
5. An AO is stationed at compartment AI2 to monitor amperage.
6. Bus E3 is being placed on the diesel generator to facilitate work on the master supply breaker from Bus 2D.

INITIATING CUE:

You are directed by the Unit CRS to perform the Control Operator actions associated with the manual transfer of E3 from the Normal Feeder to DG3 IAW 0OP-50.1, Diesel Generator Emergency Power System Operating Procedure. Notify the Unit CRS when all required actions are complete.

PERFORMANCE CHECKLIST

NOTE: Sequence is assumed unless otherwise indicated, comments required for any step evaluated as UNSAT.

Step 1 - Perform take a minute at job site prior to beginning task.

Examinee should cover the following questions, as deemed necessary. What are the hazards in the area? What PPE is required? Tools/PPE inspected prior to use? Energy sources secured/isolated? Is Clearance/Tag Out sufficient? What's the worst that can happen? Any ALARA concerns? Will I affect plant status? HU Tools needed?

SAT/UNSAT

TIME START: ________

NOTE: The AVAIL light indicates the DG is running at proper speed and voltage. The NO LOAD light indicates the DG output breaker is open.

PROMPT: If asked, respond as an AO that you are observing amperage at Compartment AI2 on E3. Reading approximately 300 amps.

Step 2 – Confirm AVAIL and NO LOAD lights are illuminated for DG3.

DG3 Avail and No Load lights are verified on.

SAT/UNSAT

NOTE: The generator voltage is monitored on diesel generator voltage output meter and emergency bus voltage is assumed to be the equivalent of the normal feeder supply.

Step 3 – Adjust DG3 voltage to slightly greater than Bus E3 voltage with the Auto Voltage Regulator.

DG3 voltage adjusted. Should indicate ~4160V.

SAT/UNSAT

Step 4 – Place synchroscope for DG3 output breaker (AI5) to ON.

DG3 synchroscope turned on.

****CRITICAL STEP**** **SAT/UNSAT**

Step 5 – Adjust DG3 speed with the GOVERNOR control switch until the synchroscope is rotating slowly in the FAST direction.

DG3 governor switch is adjusted until the synchroscope is rotating slowly in the FAST direction (clockwise)

****CRITICAL STEP**   SAT/UNSAT**

Step 6 – Adjust DG3 output voltage to match running-incoming AC voltage (Bus E3 voltage) using the Auto Voltage Regulator.

DG3 voltage adjusted until they are match.

****CRITICAL STEP**   SAT/UNSAT**

NOTE:	After the output breaker is closed, the diesel should be loaded quickly to prevent a reverse power trip from occurring.

Step 7 – When the synchroscope is at 12 o'clock then close the DG output breaker and observe the following actions to occur:

a. Generator output breaker closes
b. Synchroscope remains at 12 o'clock

DG3 output breaker closed when the synchroscope is at "12 o'clock" and observes the generator output breaker is closed light indication and synchroscope remains at the 12 o'clock position.

****CRITICAL STEP**   SAT/UNSAT**

Step 8 – Raise DG3 load to between 900-1000 KW by momentarily placing the Governor Switch to RAISE.

DG3 load is raised to 900 – 1000 KW.

****CRITICAL STEP**   SAT/UNSAT**

Step 9 – Place synchroscope to OFF.

Synchroscope is placed in OFF.

SAT/UNSAT

Step 10 – While raising DG load, maintain generator kvars approximately one-half the KW load, using Voltage adjusting rheostat

Voltage adjust is manipulated to maintain kvars ~one-half the KW load.

SAT/UNSAT

PROMPT: After the performer raises generator load to ~1600KW, inform them that there is zero amperage on the normal supply.

Step 11 – Raise generator load by momentarily placing the GOVERNOR motor control switch in RAISE, thus decreasing the normal supply amperage as reported by the AO.

Generator load is raised to ~1600KW, and zero amps reported.

****CRITICAL STEP** SAT/UNSAT**

Step 12 – When zero amperage is reported then place and hold the control switch (Bus 2D to Bus E3) in TRIP until both MSTR and SLAVE breakers indicate open.

2AD1 and AI2 breakers are open.

****CRITICAL STEP** SAT/UNSAT**

Step 13 – Unit CRS is notified that DG3 is supplying E3.

Unit CRS notified.

SAT/UNSAT

TERMINATING CUE: When DG3 is supplying E3 and the normal supply breakers have been opened then this JPM is complete.

TIME COMPLETED: __________

NOTE: Comments required for any step evaluated as UNSAT.

COLLECT AND CONTROL ALL JPM EXAM MATERIALS FOR EXAM SECURITY.

Step	Critical / Not Critical	Reason
1	Not Critical	Administrative
2	Not Critical	Confirm step.
3	Not Critical	Can still complete JPM without this action.
4-8	Critical	Cannot complete JPM without these actions.
9-10	Not Critical	Can still complete JPM without these actions.
11-12	Critical	Required to complete JPM
13	Not Critical	Reporting action.

REVISION SUMMARY

8	New JPM format. Added Critical/Non Critical step explanation.
7	Updated to new JPM format.
6	Added additional setup instructions to establish ~1600KW on E3. Added prompt for initial amperage on E3 before Step 2. Removed Rev. number form References. Added Time Required for Completion and Time Taken blanks on page 9.

Validation Time: 20 Minutes (approximate).

Time Taken: ____ Minutes

APPLICABLE METHOD OF TESTING

Performance:	Simulate	____	Actual	X	Unit:	2
Setting:	In-Plant	____	Simulator	X	Admin	____
Time Critical:	Yes	____	No	X	Time Limit	N/A
Alternate Path:	Yes	____	No	X		

EVALUATION

Performer: ______________________________

JPM: Pass ______ Fail ______

Remedial Training Required: Yes ______ No ______

Comments: ______________________________

☐ Comments reviewed with Performer

Evaluator Signature: ______________________ Date: __________

TASK CONDITIONS:

1. E3 is energized from BOP Bus 2D.
2. DG3 is running in Control Room Manual IAW 0OP-39, Diesel Generator Operating Procedure
3. The Load Dispatcher has been notified that E3 load will be shifted to DG3.
4. An AO is stationed at DG3.
5. An AO is stationed at compartment AI2 to monitor amperage.
6. Bus E3 is being placed on the diesel generator to facilitate work on the master supply breaker from Bus 2D.

INITIATING CUE:

You are directed by the Unit CRS to perform the Control Operator actions associated with the manual transfer of E3 from the Normal Feeder to DG3 IAW 0OP-50.1, Diesel Generator Emergency Power System Operating Procedure. Notify the Unit CRS when all required actions are complete.

DUKE ENERGY

BRUNSWICK TRAINING SECTION

JOB PERFORMANCE MEASURE

NRC 2015 SIM 7

LESSON TITLE: Restoration of APRM Rod Block and Scram Setpoints from Single Loop Operation to Two Loop Operation

LESSON NUMBER: LOT-SIM-JP-09.6-02

REVISION NO: 3

Lou Sosler 9/10/2015
PREPARER / DATE

John Biggs 9/15/2015
TECHNICAL REVIEWER / DATE

Thomas Baker 9/10/2015
VALIDATOR / DATE

Jerry Pierce 9/24/2015
LINE SUPERVISOR / DATE

Jim Barry 9/25/2015
TRAINING SUPERVISION APPROVAL / DATE

RELATED TASKS:

215209B401, Operate the Power Range Neutron Monitoring System per OP-09

K/A REFERENCE AND IMPORTANCE RATING:

201005 A1.04 4.1/4.1
Ability to predict and/or monitor changes in Scram and Rod Block trip setpoints associated with operating APRM system controls.

REFERENCES:

2OP-09, NEUTRON MONITORING SYSTEM OPERATING PROCEDURE.

TOOLS AND EQUIPMENT:

None

SAFETY FUNCTION (from NUREG 1123):

7 - Instrumentation

SETUP INSTRUCTIONS:

SETUP:

A. Initial Conditions:

1. Recommended Initial Conditions

   Any IC <75% Reactor Power

B. Malfunctions

   None

C. Overrides

   None

D. Remote Function

   None

E. Special Instructions

   Implement APRM rod block and scram setpoints for single loop operation IAW 2OP-09, section 8.2 on APRM Channel 1.

SAFETY CONSIDERATIONS:

None

EVALUATOR NOTES: (Do not read to performer)

1. The applicable procedure section **WILL** be provided to the trainee.
2. Prior to the first JPM of the JPM set, provide the JPM briefing contained in NUREG-1021, Appendix E, or similar briefing (for non-regulated exams) to the trainee(s).
3. This JPM may be performed on Unit 2.
4. Critical Step Basis
   a) Prevents Task Completion
   b) May Result in Equipment Damage
   c) Affects Public Health and Safety
   d) Could Result in Personal Injury

Read the following to the JPM performer.

TASK CONDITIONS:

1. The 'A' Recirculation Pump has been returned to service from single loop.

2. APRM Channel 1 rod block and scram setpoints for single loop operation are in effect per 2OP-09, NEUTRON MONITORING SYSTEM OPERATING PROCEDURE.

3. The password, '1-2-3-4', has been obtained from the Work Release Center cyber security password locker.

INITIATING CUE:

You are directed by the Unit CRS to restore APRM Channel 1 Rod Block and Scram setpoints for two loop operation per 2OP-09, section 8.3. Notify the Unit CRS when all required actions are complete.

PERFORMANCE CHECKLIST

NOTE: Sequence is assumed unless otherwise indicated, comments required for any step evaluated as UNSAT.

Step 1 - Perform take a minute at job site prior to beginning task.

Examinee should cover the following questions, as deemed necessary. What are the hazards in the area? What PPE is required? Tools/PPE inspected prior to use? Energy sources secured/isolated? Is Clearance/Tag Out sufficient? What's the worst that can happen? Any ALARA concerns? Will I affect plant status? HU Tools needed?

SAT/UNSAT

TIME START: ___________

Step 2 – NOTIFY the Unit CRS that APRM 1 will be bypassed.

Unit CRS is notified.

SAT/UNSAT

Step 3 – PLACE APRM 1 in BYPASS.

APRM 1 is placed in Bypass.

SAT/UNSAT

Step 4 – CONFIRM, at all four APRM 2/4 Voters, BYPASSED LED is on for APRM 1.

Bypassed LED is verified at all four 2/4 voters.

SAT/UNSAT

Step 5 – PRESS ETC soft key to obtain ENTER SET MODE soft key.

Enter Set Mode soft key is obtained.

****CRITICAL STEP** SAT/UNSAT**

Step 6 – PRESS ENTER SET MODE soft key.

Enter Set Mode soft key is pressed.

****CRITICAL STEP** SAT/UNSAT**

Step 7 – ENTER password “1 2 3 4” AND PRESS ENT.

Password is entered and ENT is pressed.

**** CRITICAL STEP ** SAT/UNSAT**

Step 8 – At the OPER-SET PARAMETERS INDEX display, SELECT SLO/BSP CONTROL using the cursor keys.

Single Loop Operation is selected.

****CRITICAL STEP** SAT/UNSAT**

Step 9 – PRESS SET PARAMETERS soft key.

Set Parameters soft key is pressed.

****CRITICAL STEP** SAT/UNSAT**

Step 10 – CHANGE the SLO ENABLED “DESIRED:” field to NO using the UP/DOWN cursor keys.

SLO Enabled “DESIRED:” field is changed to NO.

****CRITICAL STEP** SAT/UNSAT**

Step 11 – PRESS ACCEPT soft key.

ACCEPT soft key is pressed.

****CRITICAL STEP** SAT/UNSAT**

Step 12 – CONFIRM SLO Enabled “PRESENT:” field changed to NO.

SLO Enabled “PRESENT:” field is verified to display NO.

SAT/UNSAT

Step 13 – PRESS EXIT soft key.

Exit soft key is pressed.

SAT/UNSAT

Step 14 – PRESS EXIT SET MODE soft key.

Exit Set Mode soft key is pressed.

SAT/UNSAT

Step 15 – PRESS YES soft key.

Yes soft key is pressed.

SAT/UNSAT

Step 16 – CONFIRM the APRM display header does NOT indicate SLO.

APRM display is verified to NOT indicate SLO.

SAT/UNSAT

Step 17 – PRESS TRIP MEMORY RESET on all four 2/4 Voters AND CONFIRM TRIP and MEM LEDs are OFF for APRM 1.

Trip Memory Reset is pressed on all four 2/4 Voters and the Trip and Mem LEDs are verified OFF for APRM 1.

SAT/UNSAT

Step 18 – CONFIRM the applicable APRM A-06 alarms are clear.

APRM alarms are verified clear.

SAT/UNSAT

PROMPT: If notified, respond as the Unit CRS and direct APRM be removed from BYPASS.

Step 19 – REMOVE APRM 1 from BYPASS.

APRM 1 is removed from Bypass.

Step 20 – NOTIFY Unit CRS.

Unit CRS is notified.

SAT/UNSAT

TERMINATING CUE: When APRM 1 two loop rod block and scram setpoints have been restored and the Unit CRS is notified, this JPM is complete.

TIME COMPLETED: ____________

COLLECT AND CONTROL ALL JPM EXAM MATERIALS FOR EXAM SECURITY.

Step	Critical / Not Critical	Reason
1	Not Critical	Administrative
2-4	Not Critical	Not required to complete JPM.
6-11	Critical	Required to complete JPM.
12-20	Not Critical	Verification and communication steps, not required to accomplish task.

REVISION SUMMARY

3	New JPM format. Added Critical/Non Critical step explanation.
2	Updated to current revision of 0GP-01 Modified to perform actions of PT-01.6.2 only (GP-01 no longer directs placing mode switch to shutdown if PT is unsatisfactory)

Validation Time: 20 Minutes (approximate).

Time Taken: ____ Minutes

APPLICABLE METHOD OF TESTING

Performance:	Simulate	____	Actual	X	Unit:	2
Setting:	In-Plant	____	Simulator	X	Admin	___
Time Critical:	Yes	____	No	X	Time Limit	N/A
Alternate Path:	Yes	____	No	X		

EVALUATION

Performer: ____

JPM: Pass _____ Fail _____

Remedial Training Required: Yes _____ No _____

Comments: ____

____

____

____

____

____

____

☐ Comments reviewed with Performer

Evaluator Signature: ______________________________ Date: _____________

TASK CONDITIONS:

1. The 'A' Recirculation Pump has been returned to service from single loop.

2. APRM Channel 1 rod block and scram setpoints for single loop operation are in effect per 2OP-09, NEUTRON MONITORING SYSTEM OPERATING PROCEDURE.

3. The password, '1-2-3-4', has been obtained from the Work Release Center cyber security password locker.

INITIATING CUE:

You are directed by the Unit CRS to restore APRM Channel 1 Rod Block and Scram setpoints for two loop operation per 2OP-09, section 8.3. Notify the Unit CRS when all required actions are complete.

DUKE ENERGY

BRUNSWICK TRAINING SECTION

JOB PERFORMANCE MEASURE

NRC 2015 SIM 9

LESSON TITLE: **SEP-04 – Restart RB HVAC with Failure to Isolate**

LESSON NUMBER: **LOT-SIM-JP-300-K11**

REVISION NO: **2**

Lou Sosler 9/11/2015

PREPARER / DATE

John Biggs 9/16/2015

TECHNICAL REVIEWER / DATE

Thomas Baker 9/11/2015

VALIDATOR / DATE

Jerry Pierce 9/24/2015

LINE SUPERVISOR / DATE

Jim Barry 9/25/2015

TRAINING SUPERVISION APPROVAL / DATE

RELATED TASKS:

288205B501
Restart Reactor Building HVAC per EOP-01-SEP-04

K/A REFERENCE AND IMPORTANCE RATING:

288000 A3.01 3.8, 3.8
Ability to monitor Plant Ventilation System automatic isolation/initiation signals in the control room

REFERENCES:

0EOP-SEP-04 Rev. 11

TOOLS AND EQUIPMENT:

Plant Page

SAFETY FUNCTION (from NUREG 1123, Rev 2):

9 – Radioactivity Release

SIMULATOR SETUP:

A. Initial Conditions:

Recommended Initial Conditions

IC	11
Rx. Pwr.	100%
Core Age	BOC

B. Required Plant Conditions

A Secondary Containment leak that results in tripping the D12-R609A/B monitors on high radiation, which isolates RB HVAC. RPV water level is below LL2 or DW pressure is >1.7 psig and the Rx Bldg Rad Monitors are tripped.

C. Malfunctions

Event	System	Tag	Title	Value (ramp rate)	Activate Time (sec)	Deactivate Time (sec)
A	RW	RH013F	RWCU Break in Triangle Room	100%/4 mins	00	NA
A	NB	NB006F	MSL D Break before flow restrictor	1%/0 mins	00	NA

E1: Manually initiated. G5B25G1G to 0.7 over 1 minute
Set up to cause the following:

RB Rad Monitor A indication (g5b25g1g) to start rising to 0.7 over a 1 minute time frame.
RB Rad Monitor B indication (g5b25g2g) to start rising to 0.7 over a 1 minute time frame.
PROCESS RX BLDG VENT RAD HI Annunciator (ZUA345) to actuate after 50 sec.
PROCESS RX BLDG VENT RAD HI-HI Annunciator (ZUA335) to actuate after 55 sec.

E2: trc:2,aod:g5b25g1g
Set up the RB Rad Monitor A meter override to be deleted on depressing the RB Isolation Dampers close switch (K5608JCV close is true).

E3: trc:3,aod:g5b25g2g
Set up the RB Rad Monitor B meter override to be deleted on depressing the RB Isolation Dampers close switch (K5608JCV close is true).

E. Special Instructions

1. Place simulator in RUN and activate malfunctions.
2. When drywell pressure rises to cause a reactor scram, carry out the RO immediate actions.

Make sure level set is at 187 inches (this will go back to 170 inches after coming out of freeze)
Make sure that SEP-04 is cleaned.

SAFETY CONSIDERATIONS:

None

EVALUATOR NOTES: (Do not read to performer)

1. The applicable procedure section **WILL** be provided to the trainee.
2. Prior to the first JPM of the JPM set, provide the JPM briefing contained in NUREG-1021, Appendix E, or similar briefing (for non-regulated exams) to the trainee(s).
3. This JPM may be performed on Unit 2.
4. Critical Step Basis
   a) Prevents Task Completion
   b) May Result in Equipment Damage
   c) Affects Public Health and Safety
   d) Could Result in Personal Injury

Read the following to the JPM performer.

TASK CONDITIONS:

1. EOP-03-SCCP has been entered on Unit Two.
2. A high-radiation condition sensed by the RB Vent Radiation Monitors (D12-R609A/B) resulted in the isolation of Reactor Building HVAC.
3. SBGT Trains are in operation.
4. Jumpers to bypass RPV low level and drywell high pressure interlocks have been installed.
5. Reactor Building Exhaust temperature has not exceeded 135°F.
6. The leak has been isolated and EOP-03-SCCP directs restoring RB HVAC.
7. Instrument air pressure to the latch actuators for the reactor building ventilation isolation valves was never lost.

INITIATING CUE:

The Unit CRS directs you to restart Reactor Building HVAC per SEP-04 and inform him when your actions are complete.

PERFORMANCE CHECKLIST

NOTE: Sequence is assumed unless otherwise indicated, comments required for any step evaluated as UNSAT.

Step 1 - Perform take a minute at job site prior to beginning task.

Examinee should cover the following questions, as deemed necessary. What are the hazards in the area? What PPE is required? Tools/PPE inspected prior to use? Energy sources secured/isolated? Is Clearance/Tag Out sufficient? What's the worst that can happen? Any ALARA concerns? Will I affect plant status? HU Tools needed?

SAT/UNSAT

TIME START: ____________

PROMPT:	If asked, indicate the peak radiation levels on D12-RR-R605 at 4mr/hr and the Reactor Building Exhaust temperature has not exceeded 135°F.

Step 2 Place the CAC PURGE VENT ISOL OVRD, CAC-CS-5519 switch to OVERRIDE.

CAC-CS-5519 switch in OVERRIDE.

****CRITICAL STEP** SAT/UNSAT**

NOTE:	Process Reactor Building Vent Exhaust Rad Monitors at Panel H12-P606, D12-RM-K609A and D12-RM-K609B are reset.

Step 3 Ensure RESET at Panel H12-P606:

D12-RM-K609A (Process Reactor Bldg Ventilation Radiation Monitor A)

D12-RM-K609B (Process Reactor Bldg Ventilation Radiation Monitor B)

Monitors verified reset.

SAT/UNSAT

Step 4 - Reset the PCIS Group 6 Isolation on RTGB Panel P601.

PCIS Group 6 Isolation reset by depressing pushbuttons S32 and S33 on P601.

****CRITICAL STEP** SAT/UNSAT**

<u>PROMPT</u>:	If asked, inform examinee that instrument air pressure to the Reactor Building ventilation isolation valve latch actuators was never lost, **OR:**
<u>PROMPT</u>:	If requested, inform the examinee as Reactor Building Auxiliary Operator that the latches for the Reactor Building Ventilation Isolation Dampers are in the unlatched position.

Step 5 - Open RB Vent Isol Vlvs:

a. C-BFIV-RB and A-BFIV-RB

C-BFIV-RB, A-BFIV-RB are open (by depressing the upper lens cover for the valves).

****CRITICAL STEP** SAT/UNSAT**

b. D-BFIV-RB and B-BFIV-RB

D-BFIV-RB, B-BFIV-RB are open (by depressing the upper lens cover for the valves).

****CRITICAL STEP** SAT/UNSAT**

<u>NOTE</u>:	The RB HVAC Exhaust Fans should be started prior to starting a Supply Fan. (i.e. An exhaust should be started followed by a supply fan, then an exhaust fan should be started followed by a supply fan until all exhaust and supply fans are running.

Step 6 - Start as many Reactor Building Exhaust and Supply Fans as possible to provide maximum ventilation.

All four Reactor Building Exhaust and Supply Fans are running.

****CRITICAL STEP** SAT/UNSAT**

<u>SIM OP</u>:	When all RB HVAC Supply Fans have been started insert Event Trigger E1 to cause REACTOR BLDG VENT RAD Monitor to start to rise and go above 4 mr/hr.

Step 7 – Recognizes Reactor Building Vent Rad Monitor increasing.

Radiation monitor readings are rising.

SAT/UNSAT

NOTE:	Closing the BFIV-RBs will cause the Reactor Building Vent and Supply fans to automatically trip.

Step 8 – When Process Rad Hi-HI alarms, manually stop the Reactor Building Supply and Exhaust Fans.

All Reactor Building Supply and Exhaust Fans are stopped.

SAT/UNSAT

NOTE:	Closing the BFIV-RBs will cause the RB Vent Rad Monitors overrides to be deleted and go to normal readings.

Step 9 – Manually close RB Vent Isol Vlvs:

a. C-BFIV-RB and A-BFIV-RB

C-BFIV-RB, A-BFIV-RB are closed (by depressing the lower lens cover for the valves).

****CRITICAL STEP** SAT/UNSAT**

b. D-BFIV-RB and B-BFIV-RB

D-BFIV-RB, B-BFIV-RB are closed (by depressing the lower lens cover for the valves).

****CRITICAL STEP** SAT/UNSAT**

Step 10 - Ensure initiated SBGT system.

Both SBGT trains are operating (verifies only, both trains were already in operation).

SAT/UNSAT

Step 11 – Unit SCO informed that SEP-04, RB HVAC Restart procedure cannot be performed at this time.

Unit SCO informed.

SAT/UNSAT

TERMINATING CUE:	RB HVAC has been isolated and SBGT has been verified running.

TIME COMPLETED: ____________

COLLECT AND CONTROL ALL JPM EXAM MATERIALS FOR EXAM SECURITY.

Step	Critical / Not Critical	Reason
1	Not Critical	Administrative
2-6	Critical	Required to complete JPM.
7-8	Not Critical	JPM can be completed without performing these steps.
9	Critical	Required to complete JPM.
10-11	Not Critical	Verify and report. Not required to complete JPM.

REVISION SUMMARY

2	New JPM format. Added Critical/Non Critical step explanation.
1	Changed to rad signal vs high temperature for the alternate path.

(Provide sufficient detail for reviewers and evaluators to understand the scope of any technical and/or administrative changes).

Validation Time: 22 Minutes (approximate).

Time Taken: ____ Minutes

APPLICABLE METHOD OF TESTING

Performance:	Simulate	____	Actual	X	Unit:	2
Setting:	In-Plant	____	Simulator	X	Admin	___
Time Critical:	Yes	____	No	X	Time Limit	N/A
Alternate Path:	Yes	X	No	____		

EVALUATION

Performer: ____

JPM: Pass ______ Fail ______

Remedial Training Required: Yes ______ No ______

Comments: ____

____

____

____

____

____

____

☐ Comments reviewed with Performer

Evaluator Signature: ______________________________ Date: ____________

TASK CONDITIONS:

1. EOP-03-SCCP has been entered on Unit Two.
2. A high-radiation condition sensed by the RB Vent Radiation Monitors (D12-R609A/B) resulted in the isolation of Reactor Building HVAC.
3. SBGT Trains are in operation.
4. Jumpers to bypass RPV low level and drywell high pressure interlocks have been installed.
5. Reactor Building Exhaust temperature has not exceeded 135°F.
6. The leak has been isolated and EOP-03-SCCP directs restoring RB HVAC.
7. Instrument air pressure to the latch actuators for the reactor building ventilation isolation valves was never lost.

INITIATING CUE:

The Unit CRS directs you to restart Reactor Building HVAC per SEP-04 and inform him when your actions are complete.

DUKE ENERGY

BRUNSWICK TRAINING SECTION

JOB PERFORMANCE MEASURE

LESSON TITLE: Resetting RCIC Mechanical Overspeed

LESSON NUMBER: AOT-OJT-JP-016-A01

REVISION NO: 3

Lou Sosler 9/24/2015
PREPARER / DATE

Matt Wooldridge 9/24/2015
TECHNICAL REVIEWER / DATE

John Biggs 9/24/2105
VALIDATOR / DATE

Jerry Pierce 9/25/2015
LINE SUPERVISOR / DATE

Jim Barry 9/25/2015
TRAINING SUPERVISION APPROVAL / DATE

RELATED TASKS:

217601B404, Reset RCIC Mechanical Overspeed Trip Per 1(2)OP-16

K/A REFERENCE AND IMPORTANCE RATING:

295031 Reactor Low Water Level
EA1 Ability to operate and/or monitor the following as they apply to REACTOR LOW WATER LEVEL : (CFR: 41.7 / 45.6)
05 Reactor core isolation system 4.3/ 4.3

REFERENCES:

1(2)OP-16

TOOLS AND EQUIPMENT:

None

SAFETY FUNCTION (from NUREG 1123, Rev. 2, Supp. 1):

2 – Reactor Water Inventory Control

SAFETY CONSIDERATIONS:

1. Notify SM/CRS of JPM performance prior to commencing In-plant JPM.
2. Determine actual radiological conditions and potentially contaminated areas to achieve ALARA.
3. Ensure all electrical safety requirements are observed.
4. Operating equipment hazards.
5. DO NOT OPERATE any plant equipment during performance of this JPM.

EVALUATOR NOTES: (Do not read to performer)

1. The applicable procedure section **WILL** be provided to the trainee.
2. Prior to the first JPM of the JPM set, provide the JPM briefing contained in NUREG-1021, Appendix E, or similar briefing (for non-regulated exams) to the trainee(s).
3. This JPM may be performed on Unit 1 or Unit 2 as selected by the evaluator.
4. Critical Step Basis
   a) Prevents Task Completion
   b) May Result in Equipment Damage
   c) Affects Public Health and Safety
   d) Could Result in Personal Injury

Read the following to the JPM performer.

TASK CONDITIONS:

1. This task will be performed on Unit ____.
2. The RCIC turbine has tripped due to the mechanical overspeed trip device.
3. RCIC speed has been verified to have stopped.
4. RCIC Turbine Trip and Throttle valve motor actuator is in the closed position.

INITIATING CUE:

You are directed by the Control Operator to locally reset the RCIC mechanical overspeed trip device in accordance with 1(2)OP-16, Section 8.3, Mechanical Overspeed Reset, and inform the control room when the required actions are complete.

PERFORMANCE CHECKLIST

NOTE: Sequence is assumed unless otherwise indicated, comments required for any step evaluated as UNSAT.

Step 1 - Perform take a minute at job site prior to beginning task.

Examinee should cover the following questions, as deemed necessary. What are the hazards in the area? What PPE is required? Tools/PPE inspected prior to use? Energy sources secured/isolated? Is Clearance/Tag Out sufficient? What's the worst that can happen? Any ALARA concerns? Will I affect plant status? HU Tools needed?

SAT/UNSAT

TIME START ___________

Step 2 – **ENSURE** *TURBINE TRIP & THROTTLE VLV* motor actuator, is closed.

Verifies the TURBINE TRIP & THROTTLE VLV motor actuator, is closed.

SAT/UNSAT

<u>NOTE:</u> When the emergency connection rod is pulled or pushed based on body position it should be held until the tappet assembly has dropped in place.

Step 3 – **PUSH OR PULL**, depending on body position, the emergency connection rod against spring pressure in the direction of *TURBINE TRIP & THROTTLE VLV, E51-V8*, (approximately inch).

Moves the emergency connection rod in the direction of the TURBINE TRIP & THROTTLE VLV, E51-V8 and holds it in that position until the tappet and ball assembly have dropped in place.

****CRITICAL STEP** SAT/UNSAT**

<u>PROMPT:</u> Inform examinee that the tappet assembly has NOT dropped into place.

Step 4 – **OBSERVE** the tappet assembly, which resembles a plunger, drop into place.

Determines that the tappet and ball assembly has NOT dropped in place.

SAT/UNSAT

PROMPT: After depressing the tappet assembly, Inform examinee that the tappet assembly HAS dropped into place.

Step 5 – **IF** the tappet assembly does **NOT** drop in place, **THEN LIGHTLY** depress the assembly.

Verifies the tappet and ball assembly has dropped in place.

****CRITICAL STEP** SAT/UNSAT**

Step 6 – **RELEASE** the emergency connection rod **AND ENSURE** the head lever is resting against the flat on the tappet nut. (approximately 1/16 inch of engagement will be provided).

Releases the emergency connection rod and verifies the tappet assembly remains reset.

****CRITICAL STEP** SAT/UNSAT**

Step 7 – **NOTIFIES** the Control Room that the RCIC Mechanical Overspeed device is reset.

Acknowledge the communication as the Control Operator.

SAT/UNSAT

TERMINATING CUE: When the RCIC Mechanical Overspeed device head lever is resting against the flat on the tappet nut. this JPM is complete.

TIME COMPLETED ____________

COLLECT AND CONTROL ALL JPM EXAM MATERIALS FOR EXAM SECURITY.

Step	Critical / Not Critical	Reason
1	Not Critical	Administrative
2	Not Critical	Part of given conditions.
3	Critical	Required to complete task.
4	Not Critical	No action required.
5	Critical	Conditional step required to complete task.
6	Critical	Required to complete task.
7	Not Critical	Communicate results.

REVISION SUMMARY

3	Updated to new JPM template. Added Duke logo. Added statement to reinforce plant equipment will not be operated.

Validation Time: 10 Minutes (approximate).

Time Taken: ____ Minutes

APPLICABLE METHOD OF TESTING

Performance:	Simulate	X	Actual	______	Unit:	1 / 2
Setting:	In-Plant	X	Simulator	______	Admin	___
Time Critical:	Yes	_____	No	X	Time Limit	N/A
Alternate Path:	Yes	_____	No	X		

EVALUATION

Performer: ____

JPM: Pass ______ Fail ______

Remedial Training Required: Yes ______ No ______

Comments: ____

☐ Comments reviewed with Performer

Evaluator Signature: ________________________________ Date: ______________

TASK CONDITIONS:

1. This task will be performed on Unit ____.
2. The RCIC turbine has tripped due to the mechanical overspeed trip device.
3. RCIC speed has been verified to be less than 4000 rpm.
4. RCIC Turbine Trip and Throttle valve motor actuator is in the closed position.
5. All applicable prerequisites have been satisfied.

INITIATING CUE:

You are directed by the Control Operator to locally reset the RCIC mechanical overspeed trip device in accordance with 1(2)OP-16 and inform the control room when the required actions are complete.

8.3 Mechanical Overspeed Reset

8.3.1 Initial Conditions

1. The RCIC turbine has tripped due to the mechanical overspeed trip device. ☐

8.3.2 Procedural Steps

> **CAUTION**
>
> **IF** the RCIC turbine trips on an overspeed condition, **THEN** *TURBINE TRIP & THROTTLE VLV, E51-V8*, should **NOT** be reset until turbine speed is less than 4000 rpm.

1. **ENSURE** *TURBINE TRIP & THROTTLE VLV* motor actuator, is closed. ☐
2. **PUSH OR PULL**, depending on body position, the emergency connection rod against spring pressure in the direction of *TURBINE TRIP & THROTTLE VLV, E51-V8*, (approximately 1 inch). ☐
3. **OBSERVE** the tappet assembly, which resembles a plunger, drop into place. ☐
4. **IF** the tappet assembly does **NOT** drop in place, **THEN LIGHTLY** depress the assembly. ☐
5. **RELEASE** the emergency connection rod **AND ENSURE** the head lever is resting against the flat on the tappet nut. (approximately 1/16 inch of engagement will be provided). ☐
6. **IF** the mechanical overspeed trip device did **NOT** reset, **THEN CONTACT** the Control Room for further instructions. ☐

DUKE ENERGY
BRUNSWICK TRAINING SECTION
JOB PERFORMANCE MEASURE

LESSON TITLE: Unloaded Maintenance Start of the Supp DG

LESSON NUMBER: AOT-OJT-JP-039.1-01

REVISION NO: 0

Bob Bolin 9/24/2015
PREPARER / DATE

Matt Wooldridge 9/24/2015
TECHNICAL REVIEWER / DATE

John Biggs 9/24/2015
VALIDATOR / DATE

Jerry Pierce 9/25/2015
LINE SUPERVISOR / DATE

Jim Barry 9/25/2015
TRAINING SUPERVISION APPROVAL / DATE

RELATED TASKS:

(Specify, as applicable)

K/A REFERENCE AND IMPORTANCE RATING:

264000 A3.03 3.4 / 3.4
Ability to monitor automatic operations of the EMERGENCY GENERATORS (DIESEL/JET) including indicating lights, meters, and recorders.

REFERENCES:

0OP-39.1, Supplemental Diesel Generator Operating Procedure

TOOLS AND EQUIPMENT:

PPE

SAFETY FUNCTION (from NUREG 1123, Rev. 2, Supp. 1):

6 - Electrical

SETUP INSTRUCTIONS

None

SAFETY CONSIDERATIONS:

1. Notify SM/CRS of JPM performance prior to commencing In-plant JPM.
2. Determine actual radiological conditions and potentially contaminated areas to achieve ALARA.
3. Ensure all electrical safety requirements are observed.
4. DO NOT OPERATE any plant equipment during performance of this JPM.

EVALUATOR NOTES: (Do not read to performer)

1. The applicable procedure section **WILL** be provided to the trainee.
2. Prior to the first JPM of the JPM set, provide the JPM briefing contained in NUREG-1021, Appendix E, or similar briefing (for non-regulated exams) to the trainee(s).
3. This JPM may be performed on Unit 1 or Unit 2 as selected by the evaluator.
4. Critical Step Basis
   a. Prevents Task Completion
   b. May Result in Equipment Damage
   c. Affects Public Health and Safety
   d. Could Result in Personal Injury

Read the following to the JPM performer.

TASK CONDITIONS:

1. This task will be performed on Unit _2_.
2. All applicable Prerequisites of Section 5.0 have been met.
3. The alignment of the 480 VAC source at the Supp DG Manual Transfer Switch is to MCC 2TD (the Unit **NOT** anticipating MVP operation).
4. Supp DG is in standby per Section 6.1.1.
5. Air Box Drain Collector Tank bulls-eye does NOT indicate level.
6. Engine Lube oil sump level is above the LOW mark and temperature is 95°F.
7. Another Operator is performing Section 6.3.2
8. PA announcement has already been performed for the start of the Supp DG.

INITIATING CUE:

You are directed by the Unit CRS to perform the field actions for Section 6.1.2 in accordance with 0OP-39.1 and inform the CRS when the Supp DG is at rated speed and voltage.

PERFORMANCE CHECKLIST

NOTE: Sequence is assumed unless otherwise indicated, comments required for any step evaluated as UNSAT.

Step 1 - Perform take a minute at job site prior to beginning task.

Examinee should cover the following questions, as deemed necessary. What are the hazards in the area? What PPE is required? Tools/PPE inspected prior to use? Energy sources secured/isolated? Is Clearance/Tag Out sufficient? What's the worst that can happen? Any ALARA concerns? Will I affect plant status? HU Tools needed?

SAT / UNSAT

TIME START: ___________

NOTE: Unless noted otherwise, all actions are performed from the GEN OVERVIEW screen at the engine control panel HMI in the electrical enclosure.

NOTE: The HMI will prompt for confirmation prior to performing certain actions

Step 2 – **Depress** Maint Start button

Main Start pushbutton is selected from the GEN OVERVIEW screen at the engine control panel HMI

SAT / UNSAT

Step 3 – At the prompt, **depress** Yes to acknowledge the diesel start at idle and all operations are manually initiated and **select** to confirm intention of Maintenance Start of Supp DG.

On the Prompt screen "YES" is selected and pressed to acknowledge and confirm the Maintenance Start of Supp DG.

****CRITICAL STEP** SAT / UNSAT**

Step 4 – **Check** Supp DG starts and ramps to idle speed of approximately 350 rpm

Checks to hear engine noise and HMI reads 350 RPM and steady

SAT / UNSAT

Step 5 – **Confirm** the following:

The AC Lube Oil Soakback Pump is OFF		
The Lube Oil Circulating Oil Pump is OFF		
Radiator Fans Start:	#1	
	#2	
Enclosure Vent Fans Start:	#1	
	#2	
	#3	
	#4	
Diesel enclosure Intake louvers OPEN		
Diesel enclosure Exhaust louvers OPEN		

Checks the HMI to ensure the proper components are operating

SAT / UNSAT

Step 6 – **Confirm** the following at the HMI:

Lube Oil Pump Pressure >30 psig	
Eng Fuel Pump Pressure >12psig	

Checks the HMI to ensure the proper pressures for required components.

SAT / UNSAT

Step 7 - **WHEN** Idle Time Remaining indication reaches zero, **THEN depress** Rated Speed.

Waits until timer reaches 0 (zero) and then depresses the Rated Speed button on the HMI

SAT / UNSAT

Step 8 - At the prompt, **depress** Yes to acknowledge release to rated speed and **Check** the engine ramps to approximately 900 rpm.

Depresses "YES" to acknowledge release of engine to rated speed; hears and checks on HMI the engine ramp and steady out at 900 RPM

****CRITICAL STEP** SAT / UNSAT**

Step 9 - **Select** Turn On Regulator and **Depress** Yes to acknowledge voltage regulator operation.

Selects pushbutton for turning on the Voltage Regulator and Confirms the operation by selecting the "YES" acknowledgement pushbutton.

****CRITICAL STEP** SAT / UNSAT**

Step 10 – **Check** voltage of approximately 4160VAC on all three phases at the HMI screen.

Observes voltage of the Supp DG on the HMI screen to be ~4160VAC

SAT / UNSAT

Step 11 – **Inform** the CRS that the Supp DG is operating at rated speed and Voltage.

Contacts the CRS and informs him/her that the Supp DG is at rated speed and voltage.

SAT / UNSAT

TERMINATING CUE: When the Supp DG is at rated speed and voltage, this JPM is complete.

TIME COMPLETED: ______________

COLLECT AND CONTROL ALL JPM EXAM MATERIALS FOR EXAM SECURITY.

Step	Critical / Not Critical	Reason
1	Not Critical	Administrative
2	Not Critical	Procedural Compliance
3	Critical	Required to complete task
4	Not Critical	Check for normal equipment response
5	Not Critical	Confirm Step
6	Not Critical	Confirm Step
7	Not Critical	Procedural Compliance
8	Critical	Required to complete task
9	Critical	Required to complete task
10	Not Critical	Confirm Step
11	Not Critical	Communication step

REVISION SUMMARY

0	New JPM from recent plant mods. Incorporated into new JPM template.

Validation Time: _20_ Minutes (approximate).

Time Taken: ____ Minutes

APPLICABLE METHOD OF TESTING

Performance:	Simulate	X	Actual	X	Unit:	0
Setting:	In-Plant	X	Simulator	____	Admin	___
Time Critical:	Yes	____	No	X	Time Limit	N/A
Alternate Path:	Yes	____	No	X		

EVALUATION

Performer: ____

JPM: Pass ______ Fail ______

Remedial Training Required: Yes ______ No ______

Comments: ____

____

____

____

____

____

____

☐ Comments reviewed with Performer

Evaluator Signature: ______________________________ Date: ______________

TASK CONDITIONS:

1. This task will be performed on Unit _2_.
2. All applicable Prerequisites of Section 5.0 have been met.
3. The alignment of the 480 VAC source at the Supp DG Manual Transfer Switch is to MCC 2TD (the Unit **NOT** anticipating MVP operation).
4. Supp DG is in standby per Section 6.1.1.
5. Air Box Drain Collector Tank bulls-eye does NOT indicate level.
6. Engine Lube oil sump level is above the LOW mark and temperature is 95°F.
7. Another Operator is performing Section 6.3.2
8. PA announcement has already been performed for the start of the Supp DG.

INITIATING CUE:

You are directed by the Unit CRS to perform the field actions for Section 6.1.2 in accordance with 0OP-39.1 and inform the CRS when the Supp DG is at rated speed and voltage.

6.1.2 Unloaded Maintenance Start of the Supp DG

1. **Check** the following initial conditions are satisfied:

   - Section 5.0, Prerequisites for Supp DG startup are satisfied. ..... ________

NOTE

0-DGS-MTS-LG1 (Supp DG Manual Transfer Swch) can be aligned to Transfer to Normal, as indicated by Load Connected to Normal light, or Transfer to Emergency position, as indicated by Load Connected to Emergency light. Normal position supplies 480VAC from MCC 2TD to the Supp DG MCC while Emergency position supplies power from MCC 1TD. To preclude possible MCC overloads during operation of a unit's Mechanical Vacuum Pumps, the Manual Transfer Switch will be aligned to the opposite unit. .... ☐

   - The alignment of the 480 VAC source at the Supp DG Manual Transfer Switch , either MCC 1TD or 2TD, is to the unit **NOT** anticipating MVP operation. .... ________
   - Supp DG is in standby in accordance with Section 6.1.1. .... ________
   - Unit CRS permission to start the Supp DG. .... ________

2. **IF** air box drain collector tank bulls-eye indicates level, **THEN** locally **drain** collector tank to a suitable container, using 0-DGS-DIE-V4 (Air Box Collector Tank Drn Vlv)........ ________

   a. **Contact** Environmental for appropriate disposal of contents. ..... ________

   b. **Ensure** 0-DGS-DIE-V4 (Air Box Collector Tank Drn Vlv) is closed........ ________

3. **Ensure** engine lube oil sump level is above the Low mark on the dipstick, it may be above the Full mark. .... ________

4. **IF** the Supp DG has **NOT** been in operation or barred within the previous 7 days, **THEN perform** Section 6.3.2........ ________

5. **Announce** over PA that the Supp DG will be started. .... ________

6. **Ensure** Lube Oil Inlet Temperature is greater than or equal to 90°F as indicated on 0-DGS-LO-TI-32 (Lube Oil Inlet Temperature Indicator)........ ________

6.1.2 Unloaded Maintenance Start of the Supp DG (continued)

NOTE

- Unless noted otherwise, all actions are performed from the GEN OVERVIEW screen at the engine control panel HMI in the electrical enclosure. ☐
- The HMI will prompt for confirmation prior to performing certain actions. ☐

7. **Depress** Maint Start button. ________
8. At the prompt, **depress** Yes to acknowledge the diesel start at idle and all operations are manually initiated. ________
9. At the prompt, **select** Yes to confirm intention of Maintenance Start of Supp DG. ________
10. **Check** Supp DG starts and ramps to idle speed of approximately 350 rpm. ________
11. **Confirm** the following:
    - The AC Lube Oil Soakback Pump is OFF. ________
    - The Lube Oil Circulating Oil Pump is OFF. ________
    - Radiator Fans start
        - ◊ Fan #1 ________
        - ◊ Fan #2 ________
    - Enclosure Vent Fans start
        - ◊ Fan #1 ________
        - ◊ Fan #2 ________
        - ◊ Fan #3 ________
        - ◊ Fan #4 ________
    - Diesel enclosure louvers OPEN
        - ◊ Intake louver ________
        - ◊ Exhaust louver ________

6.1.2 Unloaded Maintenance Start of the Supp DG (continued)

12. **Confirm** the following at the HMI:

    - Lube Oil Pump Pressure greater than 30 psig .... ________
    - Eng Fuel Pump Pressure greater than 12 psig .... ________

NOTE

- Engine will idle 4 minutes for warmup. After 4 minute warmup, engine will remain at idle until operator selects rated speed. If rated speed is selected during this 4 minute warmup, unit will accelerate to rated speed. Once rated speed is attained, the engine can be returned to idle speed by selection of Idle at the HMI. .... ☐
- Maintenance activities may or may **NOT** require rated speed or voltage regulator operation. The steps **NOT** needed for the maintenance activity may be NA. .... ☐

13. **WHEN** Idle Time Remaining indication reaches zero, **THEN depress** Rated Speed. .... ________

14. At the prompt, **depress** Yes to acknowledge release to rated speed. ... ________

    a. **Check** the engine ramps to approximately 900 rpm. .... ________

NOTE

The following steps place a voltage regulator in service, and if needed, energizes the Auxiliary Transformer, otherwise these steps may be marked NA. .... ☐

15. **IF** desired to use the non-active voltage regulator, **THEN select** voltage regulator, VR1 or VR2, at the Gen Bus HMI screen.. .... ________

    a. At the prompt, **depress** Yes to acknowledge selection of the desired voltage regulator.. ________

16. **Select** Turn On Regulator.. ________

17. At the prompt, **depress** Yes to acknowledge voltage regulator operation.. ________

18. **Check** voltage of approximately 4160VAC on all three phases at the HMI screen. .... ________

6.1.2 Unloaded Maintenance Start of the Supp DG (continued)

19. **IF** desired to close the Aux Breaker to ATS (52/SVC), **THEN post** Stop signs on both electrical enclosure doors. .... ________

WARNING

The Aux Breaker to ATS (52/SVC) will close 1 minute after confirming desire to close the breaker. For personnel safety, electrical enclosure should be cleared until breaker closes.. ☐

20. From the Gen Bus screen, **depress** position indication softkey for the Aux Breaker To ATS (52/SVC). .... ________
21. At the prompt, **depress** Yes to close the Auxiliary Breaker. .... ________
22. **Clear** the electrical enclosure until the Aux Breaker closes. .... ________
23. **Confirm** Aux Breaker To ATS (52/SVC) CLOSED. .... ________
24. **Check** Supp DG ATS shifts to Normal (Source 1) which is the normal supply from the output of the Supp DG. .... ________
25. **Remove** Stop signs from both electrical enclosure doors..................... ________
26. **Complete** the following:

Date/Time Completed ________

Performed By (Print)	Initials
________	________
________	________
________	________
________	________

Reviewed By ______________________

Unit CRS/SRO

DUKE ENERGY
BRUNSWICK TRAINING SECTION
JOB PERFORMANCE MEASURE

LESSON TITLE: Local Deluge System Manual Operation for SBGT Train

LESSON NUMBER: AOT-OJT-JP-010-A01

REVISION NO: 4

Lou Sosler 9/24/2015
PREPARER / DATE

Matt Wooldridge 9/24/2015
TECHNICAL REVIEWER / DATE

John Biggs 9/24/2015
VALIDATOR / DATE

Jerry Pierce 9/25/2015
LINE SUPERVISOR / DATE

Jim Barry 9/25/2015
TRAINING SUPERVISION APPROVAL / DATE

RELATED TASKS:

261503B104
Operate Deluge System Locally for SBGT per OP-10.

K/A REFERENCE AND IMPORTANCE RATING:

286000A2.08 3.2/3.3 Failure of Fire Protection System to Actuate When Required

REFERENCES:

1(2) OP-10, Section 8.3

TOOLS AND EQUIPMENT:

Plant page (or) Radio

SAFETY FUNCTION (from NUREG 1123, Rev. 2, Supp. 1):

8 - Plant Service Systems

SAFETY CONSIDERATIONS:

1. Notify SM/CRS of JPM performance prior to commencing In-plant JPM.
2. Determine actual radiological conditions and potentially contaminated areas to achieve ALARA.
3. Ensure all electrical safety requirements are observed.
4. DO NOT OPERATE any plant equipment during performance of this JPM.

EVALUATOR NOTES: (Do not read to performer)

1. The applicable procedure section **WILL** be provided to the trainee.
2. Prior to the first JPM of the JPM set, provide the JPM briefing contained in NUREG-1021, Appendix E, or similar briefing (for non-regulated exams) to the trainee(s).
3. This JPM may be performed on Unit 1 or Unit 2 as selected by the evaluator.
4. Critical Step Basis
   a) Prevents Task Completion
   b) May Result in Equipment Damage
   c) Affects Public Health and Safety
   d) Could Result in Personal Injury

Read the following to the JPM performer.

TASK CONDITIONS:

1. This task will be performed on Unit ____.
2. A fire has occurred in the A train of the SBGT System.
3. The SBGT train temperature is greater than 210°F.
4. The control room has started the B SBGT train.

INITIATING CUE:

You are directed by the CRS to manually initiate the deluge system for the Unit ____ A SBGT train IAW 1(2)OP-10 and inform the control room when the deluge system has been manually initiated and the A SBGT train can be stopped.

PERFORMANCE CHECKLIST

NOTE: Sequence is assumed unless otherwise indicated, comments required for any step evaluated as UNSAT.

Step 1 - Perform take a minute at job site prior to beginning task.

Examinee should cover the following questions, as deemed necessary. What are the hazards in the area? What PPE is required? Tools/PPE inspected prior to use? Energy sources secured/isolated? Is Clearance/Tag Out sufficient? What's the worst that can happen? Any ALARA concerns? Will I affect plant status? HU Tools needed?

SAT/UNSAT*

TIME START ___________

NOTE:	The solenoid operated pilot valve bleeder valve is a small brass lever on the hexagonal connector on the valve neck just below the solenoid. On Train A, they are located on the back side of the hexagonal connectors.
PROMPT:	If steps to OPEN the solenoid-operated pilot valve bleeder valve and manual isolation valves are performed correctly, inform the examinee that water is flowing after each valve group is opened.
NOTE:	Unit Two valves are in parenthesis.

Step 2 - OPEN the deluge valve solenoid-operated pilot valve bleeder valve FP-DVA-1A1 (FP-DVA-2A1), DELUGE VALVE, by rotating the small lever on solenoid-operated pilot valve neck 90°.

Bleeder valve rotated 90° CLOCKWISE.

****CRITICAL STEP** SAT/UNSAT**

Step 3 - UNLOCK and OPEN deluge valve, FP-DVA-1A1-B (FP-DVA-2A1-B), SBGT A-1 Deluge Valve Main Isolation Valve.

Main Isolation Valve is open.

****CRITICAL STEP** SAT/UNSAT**

Step 4 - UNLOCK and OPEN deluge valve WW-V237, SBGT A-1 Deluge Valve Outlet Valve.

Outlet Valve is open.

****CRITICAL STEP** SAT/UNSAT**

Step 5 - OPEN the deluge valve solenoid-operated pilot valve bleeder valve FP-DVA-1A2 (FP-DVA-2A2), DELUGE VALVE, by rotating the small lever on solenoid-operated pilot valve neck 90°.

Bleeder valve rotated 90° CLOCKWISE.

****CRITICAL STEP** SAT/UNSAT**

Step 6 - UNLOCK and OPEN deluge valve, FP-DVA-1A2-B (FP-DVA-2A1-B), SBGT A-2 Deluge Valve Main Isolation Valve.

MAIN ISOLATION VALVE is open.

****CRITICAL STEP** SAT/UNSAT**

Step 7 - UNLOCK and OPEN deluge valve WW-V235, SBGT A-2 Deluge Valve Outlet Valve.

OUTLET ISOLATION VALVE is open.

****CRITICAL STEP** SAT/UNSAT**

Step 8 - Contact the Control Room and report that manual deluge has been initiated on the A Train of SBGT and the SBGT train can be stopped.

Control Room contacted.

SAT/UNSAT

TERMINATING CUE: When a flowpath has been established via a deluge valve this JPM is complete.

TIME COMPLETED ______________

COLLECT AND CONTROL ALL JPM EXAM MATERIALS FOR EXAM SECURITY.

Step	Critical / Not Critical	Reason
1	Not Critical	Administrative
2	Critical	Required to complete task.
3	Critical	Required to complete task
4	Critical	Required to complete task
5	Critical	Required to complete task
6	Critical	Required to complete task
7	Critical	Required to complete task
8	Non Critical	Communication of task completion.

REVISION SUMMARY

4	Updated to new template. Added Duke logo. Updated safety considerations, prior to JPM performance. Added Core 4 to Work Practices section. Included electrical safety in work practices. Added statement to reinforce plant equipment will not be operated.

Validation Time: 15 Minutes (approximate).

Time Taken: ____ Minutes

APPLICABLE METHOD OF TESTING

Performance:	Simulate	X	Actual	____	Unit:	____
Setting:	In-Plant	X	Simulator	____	Admin	____
Time Critical:	Yes	____	No	X	Time Limit	N/A
Alternate Path:	Yes	____	No	X		

EVALUATION

Performer: ______________________________

JPM: Pass ____ Fail ____

Remedial Training Required: Yes ____ No ____

Comments: ______________________________

____

____

____

____

____

____

☐ Comments reviewed with Performer

Evaluator Signature: ______________________________ Date: ____________

TASK CONDITIONS:

1. This task will be performed on Unit ____.
2. A fire has occurred in the A train of the SBGT System.
3. The SBGT train temperature is greater than 210°F and a fire is indicated in the train.
4. The control room has started the B SBGT train.

INITIATING CUE:

You are directed by the CRS to manually initiate the deluge system for the Unit ____ A SBGT train IAW 1(2)OP-10 and inform the control room when the deluge system has been manually initiated and the A SBGT train can be stopped.

6.3.3 Local Deluge System Manual Operation

1. **Confirm** the following Initial Conditions are met:

   - The SBGT train temperature is greater than or equal to 210°F and a fire is indicated in the train. ______
   - Unit CRS approval is obtained prior to manual operation of deluge system. ______

2. **Start** unaffected SBGT train by placing SBGT A(B) control switch to ON. ______

NOTE

The solenoid operated pilot valve bleeder valve is a small brass lever on the hexagonal connector on the valve neck just below the solenoid. On Train A, they are located on the rear of the hexagonal connectors. On Train B, they are located on the right side of the hexagonal connectors. ☐

3. **Open** the deluge valve solenoid-operated pilot valve bleeder valve for FP-DVA-2A1 (FP-DVA-2B1) (Deluge Valve) by rotating the small lever on solenoid-operated pilot valve neck 90 degrees. ______

4. **Unlock** and **open** FP-DVA-2A1-B (FP-DVA-2B1-B) [SBGT A-1 (B-1) Deluge Valve Main Isolation Valve]. ______

5. **Unlock** and **open** WW-V237 (WW-V230) [SBGT A-1 (B-1) Deluge Valve Outlet Valve]. ______

6. **Open** the deluge valve solenoid-operated pilot valve bleeder valve for FP-DVA-2A2 (FP-DVA-2B2) (Deluge Valve) by rotating the small lever on solenoid-operated pilot valve neck 90 degrees. ______

7. **Unlock** and **open** FP-DVA-2A2-B (FP-DVA-2B2-B) [SBGT A-2 (B-2) Deluge Valve Main Isolation Valve]. ______

8. **Unlock** and **open** WW-V235 (WW-V231) [SBGT A-2 (B-2) Deluge Valve Outlet Valve]. ______

9. **Stop** affected SBGT train by placing SBGT A(B) control switch to STBY. ______

10. **Depress** SBGT A(B) Push Off pushbutton for the affected SBGT train. ______

6.3.3 Local Deluge System Manual Operation (continued)

11. **Close** the following valves to locally isolate the affected SBGT train and prevent possible water and smoke damage to the unaffected SBGT train:

    a. VA-2B-BFV-RB (VA-2E-BFV-RB) [SBGT Train 2A(B) Discharge Valve]. .... ______

    b. VA-2C-BFV-RB (VA-2G-BFV-RB) [SBGT Train 2A(B) Suction Valve]. .... ______

12. **WHEN** SBGT train temperature is less than or equal to 210°F, **THEN perform** the following:

    a. **Obtain** Unit CRS approval to secure deluge system. .... ______

    b. **Close** and **lock** FP-DVA-2A1-B (FP-DVA-2B1-B) [SBGT A-1 (B-1) Deluge Valve Main Isolation Valve]. .... ______

       ______ IV

    c. **Close** and **lock** FP-DVA-2A2-B (FP-DVA-2B2-B) [SBGT A-2 (B-2) Deluge Valve Main Isolation Valve]. .... ______

       ______ IV

    d. **Close** the deluge valve solenoid-operated pilot valve bleeder valve for FP-DVA-2A1 (FP-DVA-2B1) (Deluge Valve) by rotating the small lever on solenoid-operated pilot valve neck to the horizontal position. .... ______

       ______ IV

    e. **Close** the deluge valve solenoid-operated pilot valve bleeder valve for FP-DVA-2A2 (FP-DVA-2B2) (Deluge Valve) by rotating the small lever on solenoid-operated pilot valve neck to the horizontal position. .... ______

       ______ IV

6.3.3 Local Deluge System Manual Operation (continued)

f. **Close** and **lock** WW-V237 (WW-V230) [SBGT A-1 (B-1) Deluge Valve Outlet Valve]........ ______

______
IV

g. **Close** and **lock** WW-V235 (WW-V231) [SBGT A-2 (B-2) Deluge Valve Outlet Valve]........ ______

______
IV

h. **Depress** both of the following local deluge system Reset pushbuttons to reset system:

- Reset-Overheat Water Spray No. 1........ ______
- Reset-Overheat Water Spray No. 2........ ______

13. **Notify** Maintenance to change the SBGT filters in accordance with 0CM-FLT506, Maintenance Instructions for SBGT and Control Building Filtration System. .... ______

____
Person Notified

14. **Ensure** a Technical Specification LCO is written on the affected SBGT train........ ______

6.3.3 Local Deluge System Manual Operation (continued)

15. **Notify** RP to survey the area due to possible contamination from smoke and water runoff. .... _______

____
Person Notified

Date/Time Completed _______

Performed By (Print) Initials

_______________________ _______

_______________________ _______

_______________________ _______

_______________________ _______

Reviewed By _______________________________

Unit CRS/SRO

Facility: Brunswick   Scenario No.: NRC 1   Op-Test No.: FINAL

Examiners: ____________   Operators: SRO

____________   RO

____________   BOP

Initial Conditions: The plant is operating at 95% power at end of cycle.

Turnover: CRD Pumps are scheduled to be swapped for normal scheduled rotation. APRM 4 has failed downscale and is bypassed.

Event No.	Malf. No.	Event Type*	Event Description
1	NA	N	Swap CRD Pumps.
2	CW019F	C-BOP C-SRO	NSW Pumps 2B Trips – **AOP-18.0.**
3	NI031F	C-RO C-SRO	APRM 2 fails upscale – Tech Spec.
4	IAUPB2A6	C-BOP C-SRO	Loss of power to Main Stack Rad Monitor-Tech Spec.
5	CF036F RC019F	C-BOP C-SRO C-RO R	RFP A trip – **AOP-23.0.** Recirc Pump A fails to runback. Lower power.
6	MS031F	C-BOP C-SRO	MTLO temperature controller fails closed.
	NA	M	Turbine Trip/Reactor Scram
7	RP005F RP011F	M	Auto Scram Defeat Hydraulic ATWS
8	K2119A	C	SLC Pump A&B position failure.
9	RD036F	C	SDV Vents and Drains fail closed.

* (N)ormal, (R)eactivity, (I)nstrument, (C)omponent, (M)ajor

SCENARIO DESCRIPTION SUMMARY – 2015 NRC Scenario 1

Event	Description
1	Crew will swap CRD Pumps for rotation IAW 2OP-08. 'A' will be placed in service, 'B' in standby.
2	NSW Pump B will trip and the crew will start NSW Pump A. Since there are no NSW Pumps out of service on Unit 1, Tech Specs will not apply. Crew will enter 0AOP-18.0, Nuclear Service Water System failure, and carry out appropriate actions.
3	APRM 2 fails upscale resulting in a rod block. APRM 4 is already inop and bypassed. APRM 2 will be declared inoperable and the CRS will evaluate Tech Specs.
4	Power supply to Main Stack Rad Monitor will fail. Power loss will result in a Group 6 isolation but STBY Gas will fail to auto start. Crew will take action per APP and start SBGT system to maintain Secondary Containment. CRS will evaluate TS.
5	Reactor Feed Pump 2A will trip on low suction pressure. The crew will respond per 0AOP-23.0 Condensate/Feedwater System Failure. Reactor Recirc Pump A will fail to automatically run back to limiter #2. The crew will be required to manually runback the Recirculation Pumps to prevent approaching a Reactor Scram on low RPV water level. 0AOP-23.0 directs manual reduction of Recirc Pump speed to 48%. This action will stabilize Reactor Water Level without entering the scram avoidance region. Crew will be required to further reduce power using control rods to get below 60% for a single feed pump and verify location on the power to flow map.
6	Main Turbine Temperature Controller will fail closed and Turbine Lube Oil will heat up. The rising Lube Oil temperature will result in high Turbine Vibration. Vibration will approach the TSI setpoint requiring a manual scram and the turbine to be tripped.
7	Most control rods will fail to insert on the scram. The crew will respond per 0EOP-01-ATWS. When the main turbine is tripped EHC will control pressure
8	Crew will inject SLC per ATWS procedure. The SLC Control Switch will fail to work in the A&B pump position. Either position A or position B will work to inject boron.
9	Control rods can be manually driven into the core with RMCS per LEP-02. The SDV Vents & Drains will fail. When level has been lowered and level band has been established, the SDV V&D will be repaired. Control rods can then be inserted by repeated manual scram.

CREW CRITICAL TASKS

Description
Initiate SLC with reactor power >2% such that HCTL is not exceeded.
If reactor power is greater than 2% and level is greater than 90 inches, then lower reactor water level by terminating and preventing injection IAW ATWS procedure.
Insert control rods IAW LEP-02, Alternate Control Rod Insertion, to insert all control rods.

DUKE ENERGY

BRUNSWICK TRAINING SECTION

OPERATIONS TRAINING

INITIAL LICENSED OPERATOR

SIMULATOR EVALUATION GUIDE

2015 NRC SCENARIO 1

NSW PUMP TRIPS, RFPT TRIP, MTLO TEMPERATURE CONTROLLER FAILURE, TURBINE TRIP, HYDRAULIC ATWS

REVISION 0

Developer: Lou Sosler	**Date:** 9/11/2015
Technical Review: John Biggs	**Date:** 9/23/215
Validator: Thomas Baker	**Date:** 9/11/2015
Validator: Brian Moschet	**Date:** 9/11/2015
Facility Representative: Jerry Pierce	**Date:** 9/23/2015

REVISION SUMMARY	
0	Scenario developed for 2015 NRC Exam.

TABLE OF CONTENTS

1.0 SCENARIO OUTLINE

Event	Malf. No.	Type*	Event Description
1	NA	N	Swap CRD Pumps.
2	CW019F	C-BOP C-SRO	NSW Pumps 2B Trips – **AOP-18.0.**
3	NI031F	C-RO C-SRO	APRM 2 fails upscale – Tech Spec.
4	IAUPB2A6	C-BOP C-SRO	Loss of power to Main Stack Rad Monitor-Tech Spec.
5	CF036F RC019F	C-BOP C-SRO C-RO R	RFP A trip – **AOP-23.0.** Recirc Pump A fails to runback. Lower power.
6	MS031F	C-BOP C-SRO	MTLO temperature controller fails closed.
	NA	M	Turbine Trip/Reactor Scram
7	RP005F RP011F	M	Auto Scram Defeat Hydraulic ATWS
8	K2119A	C	SLC Pump A&B position failure.
9	RD036F	C	SDV Vents and Drains fail closed.

*(N)ormal, (R)eactivity, (C)omponent or Instrument, (M)ajor

2.0 SCENARIO DESCRIPTION SUMMARY

Event	Description
Event	**Description**
1	Crew will swap CRD Pumps for rotation IAW 2OP-08. 'A' will be placed in service, 'B' in standby.
2	NSW Pump B will trip and the crew will start NSW Pump A. Since there are no NSW Pumps out of service on Unit 1, Tech Specs will not apply. Crew will enter 0AOP-18.0, Nuclear Service Water System failure, and carry out appropriate actions.
3	APRM 2 fails upscale resulting in a rod block. APRM 4 is already inop and bypassed. APRM 2 will be declared inoperable and the CRS will evaluate Tech Specs.
4	Power supply to Main Stack Rad Monitor will fail. Power loss will result in a Group 6 isolation but STBY Gas will fail to auto start. Crew will take action per APP and start SBGT system to maintain Secondary Containment. CRS will evaluate TS.
5	Reactor Feed Pump 2A will trip on low suction pressure. The crew will respond per 0AOP-23.0 Condensate/Feedwater System Failure. Reactor Recirc Pump A will fail to automatically run back to limiter #2. The crew will be required to manually runback the Recirculation Pumps to prevent approaching a Reactor Scram on low RPV water level. 0AOP-23.0 directs manual reduction of Recirc Pump speed to 48%. This action will stabilize Reactor Water Level without entering the scram avoidance region. Crew will be required to further reduce power using control rods to get below 60% for a single feed pump and verify location on the power to flow map.
6	Main Turbine Temperature Controller will fail closed and Turbine Lube Oil will heat up. The rising Lube Oil temperature will result in high Turbine Vibration. Vibration will approach the TSI setpoint requiring a manual scram and the turbine to be tripped.
7	Most control rods will fail to insert on the scram. The crew will respond per 0EOP-01-ATWS. When the main turbine is tripped EHC will control pressure
8	Crew will inject SLC per ATWS procedure. The SLC Control Switch will fail to work in the A&B pump position. Either position A or position B will work to inject boron.
9	Control rods can be manually driven into the core with RMCS per LEP-02. The SDV Vents & Drains will fail. When level has been lowered and level band has been established, the SDV V&D will be repaired. Control rods can then be inserted by repeated manual scram.

3.0 CREW CRITICAL TASKS

Description
Initiate SLC with reactor power >2% such that HCTL is not exceeded.
If reactor power is greater than 2% and level is greater than 90 inches, then lower reactor water level by terminating and preventing injection IAW ATWS procedure.
Insert control rods IAW LEP-02, Alternate Control Rod Insertion, to insert all control rods.

4.0 TERMINATION CRITERIA

When all rods are inserted and level is being controlled above TAF the scenario may be terminated.

5.0 IMPLEMENTING REFERENCES

NOTE: Refer to the most current revision of each Implementing Reference.

Number	Title
A-5, 6-1	CRD FILTER PUMP INLET DP HI
A-5, 2-1	CHARGING WATER HI DP
UA-18, 6-1	BUS E4 4KV MOTOR OVLD.
UA-1, 1-10	NUCLEAR HDR SERV WATER PRESS LOW
UA-1, 4-10	NUCLEAR HDR SW PUMP B TRIP
0AOP-18.0	NUCLEAR SERVICE WATER SYSTEM FAILURES
A-06, 2-8	APRM UPSCALE
A-06, 3-7	APRM TROUBLE
A-06, 3-8	APRM UPSCALE TRIP/INOP
A-05, 2-2	ROD OUT BLOCK
A-06, 2-8	APRM UPSCALE TRIP/INOP
UA-5, 3-5	SBGT SYS B FAILURE
UA-5, 4-6	SBGT SYS A FAILURE
UA-4, 1-2	RFPT A TURBINE TRIPPED
UA-13, 6-5	RFPT A CONTROL TROUBLE
0AOP-23.0	CONDENSATE/FEEDWATER SYSTEM FAILURES
UA-23, 1-6	TURB OR RFP BRG TEMP HIGH
UA-23, 6-1	TURBINE VIBRATIONS HIGH

6.0 SETUP INSTRUCTIONS

1. **PERFORM** TAP-409, Miscellaneous Simulator Training Guidelines, Attachment 5, Checklist for Simulator Exam Security.
2. **RESET** the Simulator to IC-25.
3. **ENSURE** the RWM is set up as required for the selected IC.
4. **ENSURE** appropriate keys have blanks in switches.
5. **RESET** alarms on SJAE, MSL, and RWM NUMACs.
6. **ENSURE** no rods are bypassed in the RWM.
7. **PLACE** all SPDS displays to the Critical Plant Variable display (#100).
8. **ENSURE** hard cards and flow charts are cleaned up
9. **TAKE** the SIMULATOR OUT OF FREEZE
10. **LOAD** Scenario File.
11. **ALIGN** the plant as follows:

Manipulation
Ensure 2C TCC pump is in service on Unit One. Loaded in Scenario File

12. **IF desired**, take a **SNAPSHOT** and save into an available IC for later use.
13. **PLACE** a clearance on the following equipment.

Component	**Position**
APRM 4 (blue tag)	Bypassed

14. **INSTALL** Protected Equipment signage and **UPDATE** RTGB placard as follows:
15. **VERIFY** 0ENP 24.5 Form 2 (Immediate Power Reduction Form) for IC-25 is in place.
16. **ENSURE** each Implementing References listed in Section 7 is intact and free of marks.
17. **ENSURE** all materials in the table below are in place and marked-up to the step identified.

Required Materials

18. **ENSURE** Station Duty Manager and Work Week Manager names are filled in on the Shift Turnover Sheet.

19. **ADVANCE** the recorders to prevent examinees from seeing relevant scenario details.

20. **PROVIDE** Shift Briefing sheet for the CRS.

21. **VERIFY** all actions contained in TAP-409, Miscellaneous Simulator Training Guidelines, Attachment 4, Simulator Training Instructor Checklist, are complete.

7.0 INTERVENTIONS

TRIGGERS

Trig	Type	ID
1	Malfunction	CW025F - [NUC SERVICE WATER PUMP MOTOR OVERLOAD]
1	Malfunction	CW023F - [NUC SERVICE WATER PUMP SHAFT SEIZURE]
2	Malfunction	NI031F - [APRM FAILS HI]
3	Remote Function	ED_IAUPB2A6 - [UPS LOAD BKR DIST PNL 2A TO SMPL DT SKD]
4	Malfunction	CF036F - [RFP A LOW SUCT PRESSURE]
5	Malfunction	MS031F - [MTLO TEMP CNTRLR FAILS]
6	Malfunction	RD036F - [SCRAM DISC VOL DRN FAILS CLOSED]
7	Malfunction	MS017F - [TURBINE BEARING VIBRATION]
7	Malfunction	MS017F - [TURBINE BEARING VIBRATION]
7	Malfunction	MS017F - [TURBINE BEARING VIBRATION]
10	Trigger Command	did:k6101A
11	Trigger Command	did:k6103a
12	Remote Function	EP_IAEOPJP1 - [BYPASS LL-3 GROUP I ISOL (SEP-10)]
13	Remote Function	EP_IACS994P - [DW CLR B & C OVERIDE - NORMAL/STOP]
13	Remote Function	EP_IACS993P - [DW CLR A & D OVERIDE - NORMAL/STOP]
14	Remote Function	ED_IAUPDSSW - [UPS SAMPLE DET SKD XFER SW (N=U2/A=U1)]

Trig #	Trigger Text
6	K2213BXD - [DISCH VOL TEST]
7	ZUA2316 - [TURB OR RFP BRG TEMP HIGH]
10	K6101WOV - [SBGT SYS A]
11	K6103WOV - [SBGT SYS B]

MALFUNCTIONS

Malf ID	Mult ID	Description	Current Value	Target Value	Rmp time	Actime	Dactime	Trig
NI032F	APRM 4	APRM FAILS LO	True	True				
RC019F	RFP A 1	RFP SUCTION FLOW SWITCH FAILS CLOSED	True	True				
RP011F		ATWS 4	True	True				
RP005F		AUTO SCRAM DEFEAT	True	True				
CW025F	B	NUC SERVICE WATER PUMP MOTOR OVERLOAD	False	True				1
CW023F	B	NUC SERVICE WATER PUMP SHAFT SEIZURE	False	True		03:00		1
NI031F	APRM 2	APRM FAILS HI	False	True				2
CF036F		RFP A LOW SUCT PRESSURE	False	True				4
MS031F		MTLO TEMP CNTRLR FAILS	False	True				5
RD036F		SCRAM DISC VOL DRN FAILS CLOSED	False	True				6
MS017F	2	TURBINE BEARING VIBRATION	0.00	7.0	05:00			7
MS017F	3	TURBINE BEARING VIBRATION	0.00	8.0	05:00			7
MS017F	4	TURBINE BEARING VIBRATION	0.00	6.0	05:00			7

REMOTES

Remf Id	Mult Id	Description	Current Value	Target Value	Rmp time	Actime	Trig
CC_IACW4518		2C TBCCW PUMP UNIT ALIGNMENT	1	1			
ED_IAUPB2A6		UPS LOAD BKR DIST PNL 2A TO SMPL DT SKD	CLOSE	OPEN			3
EP_IAEOPJP1		BYPASS LL-3 GROUP I ISOL (SEP-10)	OFF	ON			12
EP_IACS993P		DW CLR A & D OVERIDE - NORMAL/STOP	NORMAL	STOP			13
EP_IACS994P		DW CLR B & C OVERIDE - NORMAL/STOP	NORMAL	STOP			13
ED_IAUPDSSW		UPS SAMPLE DET SKD XFER SW (N=U2/A=U1)	NORMAL	ALT			14

PANEL OVERRIDES

Tag ID	Description	Position / Target	Actual Value	Override Value	Rmp time	Actime	Dactime	Trig
K4B20A	NUC HDR SW PMP A DISCH VLV	AUTO	ON	OFF				
K2119A	S/B LIQ PUMP A & B	PUMP_A&B	OFF	OFF				
K6101A	SBGT SYS A CONT PUSH OFF	OFF	OFF	ON				
K6103A	SBGT SYS B CONT PUSH OFF	OFF	OFF	ON				

8.0 OPERATOR RESPONSE AND INSTRUCTIONAL STRATEGIES

EVENT 1: SWAP CRD PUMPS	
	Simulator Operator Actions
	Ensure Monitored Parameters is open and Scenario Based Testing Variables are loaded.

	Simulator Operator Role Play
	Pre-start checks complete for CRD Pump B. Steps 3a-e of 2OP-08, Section 6.3.2.
	E4/E3 clear of personnel.
	Step 9c of 2OP-08, Section 6.3.2 completed and IV'd.
	If asked, Stab Valve flow is 6 gpm.

Evaluator Notes	
Plant Response:	Swap CRD Pumps
Objectives:	SRO - Directs RO to swap CRD Pumps RO – Swap CRD Pumps BOP – Monitor Balance of Plant
Success Path:	CRD Pumps are swapped
Event Termination:	When directed by the Lead Evaluator, go to Event 2.

EVENT 1: SWAP CRD Pumps

Time	Pos	EXPECTED Operator Response	NOTES
	SRO	Conduct shift turnover shift briefing.	
		Direct CRD Pumps to be swapped.	2OP-08, Section 6.3.2
	BOP	Monitors the plant	
	RO	Swap CRD Pump: **IF** starting CRD Pump 2B **AND** securing CRD Pump 2A,**THEN perform** the following: **Ensure** C12-F013B (CRD Pump 2B Suction Isolation Valve) LOCKED OPEN **Ensure** C12-F014B (CRD Pump 2B Discharge Isolation Valve) LOCKED OPEN. **Ensure** C12-F015B (CRD Pump 2B Recirculation Line Isolation Valve) LOCKED OPEN **CAUTION** Failure to reduce CRD flow rate prior to starting the non-operating CRD pump could cause rod drifts and require a manual reactor scram Shift C12-FC-R600 (CRD Flow Control) to BAL. Null C12-FC-R600 (CRD Flow Control) using the manual potentiometer. Shift C12-FC-R600 (CRD Flow Control) to MAN **NOTE** When CRD flow rate is set to 35 gpm, annunciator A-05, 2-1, CRD Charging Wtr Press Hi, may ALARM Set CRD flow rate to 35 gpm IF starting CRD Pump 2B AND securing CRD Pump 2A, THEN perform the following: Start CRD Pump 2B Stop CRD Pump 2A. Ensure C12-F014A (CRD Pump 2A Discharge Isolation Valve) LOCKED OPEN. Null C12-FC-R600 (CRD Flow Control) using the setpoint tape Shift C12-FC-R600 (CRD Flow Control) to AUTO Go to Section 6.3.32 to adjust CRD parameters.	2OP-08, Section 6.3.2

EVENT 2: NSW PUMP FAILURE	
Simulator Operator Actions	
	At the direction of the Lead Evaluator, **Initiate Trigger 1** to trip the NSW Pump 2B.
	Note: Short time delay before first alarm (1min), followed by pump trip (1 min).

Simulator Operator Role Play	
	If contacted as OAO to investigate NSW pump and breaker, report 51 devices are tripped at NSW Pump B breaker on Bus E4.
	If requested, check 2B NSW in SW Bldg, no signs of any damage.
	If contacted as maintenance or I&C to investigate trip, acknowledge request

Evaluator Notes	
Plant Response:	The running NSW pump will TRIP on motor overload. The STBY NSW pump will fail to AUTO start. The BOP operator should recognize the failure and manually start the STBY NSW pump. With Unit 1 NSW Pumps operable, Tech Spec actions are not required.
Objectives:	SRO - Direct actions for loss of NSW RO - Respond to the failure of an automatic start of the A NSW pump
Success Path:	Start Standby NSW Pump
Event Termination:	Go to Event 3 at the direction of the Lead Evaluator.

EVENT 2: NSW PUMP FAILURE			
Time	**Pos**	**EXPECTED Operator Response**	**Comments**
	SRO	Direct entry into AOP-18 NSW System Failure.	
		Contact maintenance to investigate trip of 2B NSW Pump. May also report to I/C that 2A NSW Pump did not auto start.	
		Evaluate Tech Spec 3.7.2 Service Water System and Ultimate Heat Sink. • Determine 2B NSW pump inoperable • Per the Bases, 3 NSW pumps required site wide. • With Unit 1 NSW Pumps operable, no Tech Spec actions required.	
		May direct 2C CSW pump to be placed on the NSW header.	
	RO	Monitor reactor plant parameters during evolution.	

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EVENT 2: NSW PUMP FAILURE

Time	Pos	EXPECTED Operator Response	Comments
	BOP	Acknowledge / reference UA-18 (6-1) BUS E4 4KV MOTOR OVLD	This is the first alarm in. Will verify on the alarm log which pump has the overload.
		Recognize trip of 2B NSW pump and lowering NSW system pressure.	
		Announce and execute 0AOP-18.0, NSW System Failure.	
		Recognize the failure of the STBY NSW pump to start and starts standby pump. • Places 2A NSW pump in Manual. • Starts 2A NSW Pump.	
		Refer to alarms. UA-01 (1-10) NUCLEAR HEADER SERV WTR PRESS-LOW	
		May align the 2C CSW pump to the NSW header.	2OP-43, Section 6.3.40

EVENT 3: APRM 2 FAILURE - UPSCALE	
Simulator Operator Actions	
	At the direction of the Lead Evaluator, **Initiate Trigger 2** to fail APRM 2.

Simulator Operator Role Play	
	If contacted as I&C to investigate, acknowledge the request.
	If asked to pull fuses (for TRM 3.3 actions) acknowledge the request.
	After LCO entries have been determined and SRO is waiting for I&C, call as WCCSRO and request that the RO place APRM 2 in a tripped condition to support I&C trouble shooting.
	If asked, APRM 4 won't be returned to service for another 8-10 hours.

Evaluator Notes	
Plant Response:	APRM 2 will fail upscale resulting in a rod block. The APRM will be declared Inoperable per TS 3.3.1.1, Condition A and placed in trip within 12 hours. WCCSRO will request APRM TS Actions be taken in order to troubleshoot which requires the APRM mode selector switch to be place in INOP IAW 0OI-18.
Objectives:	SRO - Determine LCO for APRM 2 inoperability and direct placing channel in trip. RO - Diagnose APRM 2 failure and place in INOP.
Success Path:	ARPM 2 declared inoperable IAW TS 3.3.1.1 and placed in trip condition IAW 0OI-18.
Event Termination:	Go to Event 4 at the direction of the Lead Evaluator.

EVENT 3: APRM 2 FAILURE - UPSCALE

Time	Pos	EXPECTED Operator Response	Comments
	SRO	Direct actions of APPs	
		Direct I&C to investigate	
		Evaluate **Tech Spec 3.3.1.1** Reactor Protection System Instrumentation Determine APRM 4 and 2 are inoperable. Determine 2 of 4 the available channels are operable for Function 2. **Condition A.1**, Required Action with one or more required channels inoperable, place in trip condition in **12 hours** Evaluate **TRM 3.3** Control Rod Block Instrumentation Determine one of the required channels is not operable for Function 1 – **Condition A.1** - **24 hours** to restore to operable.	
		Refers to 0OI-18 for actions to place APRM 2 in a tripped condition.	
		Direct APRM 2 mode selector switch placed in INOP to allow I&C troubleshooting.	
	BOP	Monitors the plant.	
	RO	Acknowledges, refers to & reports annunciators A-6 2-8 *APRM UPSCALE* 3-7 *APRM TROUBLE* 3-8 *APRM UPSCALE TRIP/INOP* A-5 2-2 *ROD OUT BLOCK*	
		Determines ARPM 2 has a critical fault (CPU Failure) and cannot be bypassed (APRM 4 already bypassed).	
		Places APRM 2 in the tripped condition by placing APRM OPER/INOP mode selector switch in "INOP" on Panel P608.	Will need Key #112 from Control room Key locker

EVENT 4: STACK RAD MONITOR FAILURE – SBGT FAILS TO START	
Simulator Operator Actions	
	At the direction of the Lead Evaluator, **Initiate Trigger 3** to fail Stack Rad Monitor downscale (loss of power).
	Transfer Stack Rad Monitor to Unit 1 UPS if requested **(Trigger 14).**

Simulator Operator Role Play	
	If asked to investigate, report Ckt #6 on UPS Panel 2A to the Stack Rad Monitor is tripped
	If contacted as Unit One, report that Unit One has the same alarms present.
	If contacted as I&C to investigate, acknowledge the request. If asked, do not recommend transfer to the alternate power supply until the cause of the trip is investigated.

Evaluator Notes	
Plant Response:	Power failure to the Stack Rad Monitor will initiate a Group 6 Isolation. Group 6 valves will isolate, Reactor Building Ventilation will isolate, but SBGT will not start. It will require manual start.
Objectives:	SRO - Determine actions required for LCO per Technical Specifications RO - Respond to a process radiation monitoring downscale/inop annunciator
Success Path:	Evaluate Tech Specs to determine required actions as outlined in SRO actions below.
Event Termination:	Go to Event 5 at the direction of the Lead Evaluator.

EVENT 4: STACK RAD MONITOR FAILURE – SBGT FAILS TO START

Time	Pos	EXPECTED Operator Response	Comments
	SRO	Direct actions of the APPS for the Main Stack Rad Monitor	
		Direct SBGT start	
		May direct entry into 0AOP-12.0	
		Direct I/C to investigate loss of UPS 2A.	
		Technical Specification / TRM • 3.3.6.1 PCIS Instrumentation, Function 2c Condition A1, Place in trip condition in 24 hours. • 3.4.5 RCS Leakage Detection Instrumentation Condition B.1, Analyze grab samples every 12 hours Condition B.2, Restore operable in 30 days • TRM 3.4 Post Accident Monitoring, Functions 2,5, and 6 Condition A.1 (F2/F6), Restore 31 days Condition B.1, (F2/F5), Restore one required channel in 7 days. • ODCM 7.3.2 Gaseous Effluent Monitoring, Function 1 A.1, Enter the Condition referenced in Table 7.3.2-1 B.1, Take a grab sample once per 12 hours B.2, Analyze the grab sample for gross noble gas activity within 24 hrs B.3, Restore the channel in 30 days C.1, C.1.1, Immediately Initiate actions to establish auxiliary sampling equipment to continuously collect samples from the associated effluent release pathway as required by Table 7.3.7-1 C.2, Restore the channel in 30 days D.1, Estimate the flow rate through the associated pathway D.2, Restore the channel in 30 days	

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EVENT 4: STACK RAD MONITOR FAILURE – SBGT FAILS TO START

Time	Pos	EXPECTED Operator Response	Comments
	RO	Plant Monitoring: May open the SW-V111 or V117 to supply cooling water to the vital header IAW 2APP-UA-05 1-9 or 2-9.	
	BOP	Report loss of Main Stack Rad Monitor and references the following APPs: <u>UA-03</u> 5-4, PROCESS OG VENT PIPE RAD HI-HI 6-3, PROCESS SMPL OG VENT PIPE DNSC/INOP 6-4, PROCESS OG VENT PIPE RAD – HI <u>UA-05</u> 3-5, SBGT SYS B FAILURE 4-6, SBGT SYS A FAILURE 6-10, RX BLDG ISOLATED <u>UA-25</u> 1-8, CTMT ATMOS RAD MON DNSC/INOP	
		Report TS review for the CRS from the Annunciator reviews. • 3.4.5 • 3.3.6.1 Table 3.3.6.1-1, function 2c • ODCM 7.3.2 Table 7.3.2-1 Function 1, 7.3.7, and 7.3.13 • TRM 3.4, Table 3.4.2 function 5	
		Determine that SBGT should have started. Start SBGTs	
		Dispatch AO to investigate UPS 2A condition.	

EVENT 5: 2A RFP TRIP – FAILURE OF VFD A TO RUNBACK	
Simulator Operator Actions	
	At the direction of the Lead Evaluator, **Initiate Trigger 4** to trip the 2A RFP.
	Acknowledge the Woodward local alarm panel when directed. Provide the Sim Operator Role Player with the appropriate local alarms on the Woodward for reporting to the control room.

Simulator Operator Role Play	
	If contacted as the AO to acknowledge local Woodward alarms, wait 1 minute, have the Sim Operator acknowledge the local alarm and report the alarms on the local panel to the control room.
	If contacted as I&C to investigate, acknowledge the request.
	If contacted as the Reactor Engineer for recommendation for actions to get below the MELLL line, ask what the SRO recommendation would be and agree with the recommendation.

Evaluator Notes	
Plant Response:	Reactor Feed Pump A will trip with a failure to of Recirc Pump A to runback to limiter #2. Crew should enter 0AOP-23. Condensate/Feedwater System Failure, Immediate operator action to reduce Recirc controllers to 48% if a RFP trips and a runback does not occur. Crew may also enter 0AOP-4.0, Low Core Flow.
Objectives:	SRO - Direct actions to respond To A Condensate/Feedwater System Failure Per 0AOP-23. RO - Respond To A Condensate/Feedwater System Failure Per 0AOP-23. Respond to a Reactor Recirc pump runback failure.
Success Path:	The crew will respond per 0AOP-23 and lower recirc pump speeds to 48%. Rods may have to be inserted to maintain operation below the MELLL line.
Event Termination:	Go to Event 6 at the direction of the Lead Evaluator.

EVENT 5: 2A RFP TRIP – FAILURE OF VFD A TO RUNBACK

Time	Pos	EXPECTED Operator Response	Comments
	SRO	Direct Recirc to be lowered to maintain level. (If AOP immediate actions were not performed will direct Recirc to be lowered to 48%)	
		Direct entry of 0AOP-23, Condensate/Feedwater System Failures.	
		Direct entry into 2AOP-4.0, Low Core Flow	
		Contact I/C for the RFP A trip and failure of Limiter #2 for Recirc A VFD.	
		Direct RO to insert rods per 0ENP-24.5 to get on or below the MELLL Line, if required.	Power reduction to ≤60% with 1 RFP.
		Direct chemistry to sample RCS activity due to power change greater than 15%.	
	RO	May report level decreasing, but recovering.	
		Performs immediate operator action of 0AOP-23, Determines a runback signal did not occur Reduces Recirc Pump speeds to 48%.	
		Inserts control rods using 0ENP-24.5 to get below the MELLL line, if required. Turns control rod power on. Selects control rod in accordance with ENP-24.5 sheet. Continuously drives selected rod in using RMCS Repeats steps until operation is below the MELLL line	

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EVENT 5: 2A RFP TRIP – FAILURE OF VFD A TO RUNBACK			
Time	**Pos**	**EXPECTED Operator Response**	**Comments**
	BOP	Responds to annunciators: UA-13 (6-5) RFP A Control Trouble UA-04 (1-2) RFP A Turbine Tripped	
		Announce and enter AOP-23.0, Condensate/Feedwater System Failure	
		Announce and enter AOP-04.0, Low Core Flow	
		Dispatch an AO to acknowledge the local alarm panel for the RFP (Woodward).	
		Dispatches personnel to determine the cause of the RFP trip.	

EVENT 6: MTLO CONTOLLER FAILURE / REACTOR SCRAM – TURBINE TRIP	
	Simulator Operator Actions
	At the direction of the Lead Evaluator, **Initiate Trigger 5** to activate the Main Turbine Lube Oil Controller failure closed.
	Short time delay until response is seen.

	Simulator Operator Role Play
	If asked as the TB AO to investigate, report that the temperature control valve to the MTLO is closed. (There is no bypass valve).
	Verify turbine vibrations on 2, 3, and 4 begin to rise after receiving UA-23 1-6.
	If asked as I&C to investigate, acknowledge the request

	Evaluator Notes
Plant Response:	Main Turbine lube oil cooler controller fails closed. When high temperature alarm annunciates activate trigger to accelerate turbine vibrations.
Objectives:	SRO -Direct action in response to an abnormal turbine vibration per UA-23 6-1 and UA-23 6-3 RO - Respond to an abnormal turbine vibration per UA-23 6-1 and UA-23 6-3.
Success Path:	Reactor Scram and Turbine trip when vibrations reach setpoint IAW APPs.
Event Termination:	When a manual scram is inserted go to the next event.

EVENT 6: MTLO CONTOLLER FAILURE / REACTOR SCRAM – TURBINE TRIP

Time	Pos	EXPECTED Operator Response	Comments
	SRO	Direct actions of APP's.	
		When vibrations rise to above the TSI setpoint, direct manual scram and turbine trip per the vibration APP	(12 mils on bearings 1-8, 10 mils on bearings 9-10) (As conservative decision making may insert before setpoint).
		Direct breaking condenser vacuum.	
	BOP	Recognize and report rising lube oil temperatures.	
		Dispatch TB AO to investigate TCV.	
		Perform actions of APPs UA-23: 1-6 TURB OR RFP BRG TEMP HIGH 6-1 TURBINE VIBRATION HIGH 6-3 TSI HIGH VIBRATION TRIP	If vibration is at or above 12 mils on bearings 1-8 or 10 mils on bearings 9 & 10 and an adjacent bearing has also exhibited a significant increase in vibration, then perform the following: (1) SCRAM the reactor (2) Trip the turbine (3) If directed by the Unit SCO, break condenser vacuum.
		Monitor turbine bearing temperatures and vibrations	PC display 630
		May place the Main Turbine Lube Oil controller in manual and attempt to operate the valve.	
		Break Condenser vacuum when directed by the SRO.	

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	RO	Plant Monitoring	
		When directed by the SRO, insert a manual scram and trip the main turbine.	
		Recognize and report an ATWS.	

EVENTS 7, 8, 9: ATWS – SLC SWITCH FAILURE – SDV VENTS AND DRAINS FAIL	
Simulator Operator Actions	
	If requested to defeat Group I LL3, wait 2 minutes, and install jumpers **(Trigger 12)**
	If requested to install LEP-02, Section 2.3 jumpers, wait 5 minutes, and inform the SRO that the jumpers are installed (RP005F already active).
	If requested to defeat Drywell Cooler LOCA Lockout, wait three minutes, then install jumpers **(Trigger 13)**.

Simulator Operator Role Play	
	Acknowledge request as I&C to investigate failure of SLC switch.
	If requested as I&C to investigate the failure of the scram discharge volume vents and drains, acknowledge the request.

Evaluator Notes	
Plant Response:	Most control rods will fail to insert on the scram. The crew will respond to the ATWS per EOP-01-ATWS. When SLC initiation is attempted, the A&B switch position will not work. The crew will enter LEP-03 and align for alternate boron injection using CRD. The scram cannot be reset due to failure of the SDV Vents and Drains.
Objectives:	SRO - Direct actions to control reactor power per EOP-01-ATWS.. RO - Perform actions for an ATWS per EOP-01-ATWS.
Success Path:	Lower level to control power, inject SLC, insert control rods.

EVENTS 7, 8, 9: ATWS – SLC SWITCH FAILURE – SDV VENTS AND DRAINS FAIL

Time	Pos	EXPECTED Operator Response	Comments
	SRO	Enter RSP and transition to ATWS. Direct mode switch to shutdown when steam flow < 3 Mlbs/hr. Direct ARI initiation. Direct Recirc Pumps Tripped. **Direct SLC initiation.** Direct ADS inhibited. Direct RWCU isolation verification. ***Direct LEP-02, Alternate Rod Insertion***	***CRITICAL TASK*** ***CRITICAL TASK***
		Direct Group 10 switches to override reset.	
		Direct terminate and prevent HPCI/Feedwater (CS/RHR when LOCA signal received) to lower level to 90 inches.	***CRITICAL TASK***
		When level reaches 90 inches, evaluate Table Q-2: If not met, establishes a level band of LL4 to +90 inches. When met, direct injection be or remain terminated.	
		When Torus temperature is greater than 95° F, enters PCCP and directs Torus Cooling.	Enclosure 5
		Directs Drywell cooling restored per SEP-10.	
		Direct injection established to maintain RPV level LL4 to TAF (or the level at which APRMs indicate downscale)	

EVENTS 7, 8, 9: ATWS – SLC SWITCH FAILURE – SDV VENTS AND DRAINS FAIL

Time	Pos	EXPECTED Operator Response	Comments
	RO	Place mode switch to shutdown when steam flow < $3x10^6$ lb/hr.	
		Initiates ARI.	
		Trips Recirc Pumps.	
		Initiates SLC. Verifies Isolation of RWCU.	***CRITICAL TASK***
		Recognizes failure of SLC switch and reports to SRO.	
		Monitor APRMs for downscale.	
		Performs LEP-02, Alternate Rod Insertion. (RMCS Section) Insert IRMs. When < range 3 on IRMs insert SRMs. Start both CRD pumps. Place CRD Flow Controller to Manual. Throttle open flow controller to establish > 260 drive water psid. Bypass RWM. ***Selects control rods and drives in using Emerg rod in notch override.***	***CRITICAL TASK***

EVENTS 7, 8, 9: ATWS – SLC SWITCH FAILURE – SDV VENTS AND DRAINS FAIL

Time	Pos	EXPECTED Operator Response	Comments
	RO	Performs LEP-02 Section 2.3: Inhibit ARI Places ARI Initiation Switch to INOP Places ARI Reset Switch to RESET and maintains for 5 seconds. Verifies red TRIP light above ARI Initiation is OFF Request LEP-02 Section 2.3 Jumpers be installed. Reset RPS when scram jumpers installed. Ensures Dish Vol Vent & Drain Test switch is in Isolate. Confirms Disch Vol Vent Valves V139 and CV-F010 are closed Confirms Disch Vol Drain valves V140 and CV-F011 are closed. Resets RPS. Place Disch Vol Vent & Drain Test switch to Normal Recognize/report failure of scram discharge volume vents and drains.	

EVENTS 7, 8, 9: ATWS – SLC SWITCH FAILURE – SDV VENTS AND DRAINS FAIL

Time	Pos	EXPECTED Operator Response	Comments
	BOP	Places ADS in inhibit.	
		Places Group 10 switches to override / reset	
		Terminate and prevent injection to RPV. Terminates and prevents HPCI IAW Hard Card. Terminates and Prevents Feedwater IAW Hard Card.	***CRITICAL TASK*** See Enclosure 1 for actions for HPCI T/P. See Enclosure 3 for actions for C&F T/P.
		May place HPCI in service for level control during ATWS when directed by the SRO.	See Enclosure 2 for HPCI Restart actions
		Restart RFP to maintain level as directed by SRO.	See Enclosure 4 for RFP Restart actions
		When Torus temperature is greater than 95° F, places Torus Cooling in service.	See Enclosure 2 for Torus Cooling Hard Card actions
		Break Condenser vacuum when directed by SRO	

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ALL RODS IN	
	Simulator Operator Actions
	When directed by the Lead Evaluator, delete the following commands: Malfunction - RD036F, Scram Disch Vol Drn Fails Closed Malfunction – RP011F, ATWS 4 (Make sure RPS is reset and scram air header pressurized before deleting)
	When directed by the Lead Evaluator, place the simulator in FREEZE
	DO NOT RESET THE SIMULATOR PRIOR TO RECEIPT OF CONCURRENCE TO DO SO FROM THE LEAD EXAMINER

	Simulator Operator Role Play
	After Sim Operator has deleted SDV malfunction, Inform the CRS that a loose wire was found on the SDV vent and drain logic and it has been repaired.

Evaluator Notes	
Plant Response:	When actions are taken to control reactor water level during the ATWS after terminating and preventing, the SDV vents and drains will be repaired and rods can be inserted.
Objectives:	SRO - Directs actions for an ATWS. RO - Insert control rods IAW LEP-02.
Success Path:	Rods inserted with LEP-02, Alternate Rod Insertion.
Scenario Termination:	***When all rods are inserted and level is being controlled above TAF the scenario may be terminated.***
Remind students not to erase any charts and not to discuss the scenario until told to do so by the evaluator/instructor.	

EVENT 8: ALL RODS IN

Time	Pos	EXPECTED Operator Response	Comments
	SRO	Exit ATWS and enter RVCP when all rods are in.	
		Direct level restored to 170 – 200 inches after rods are all in.	
	RO	Confirms Disch Vol Vent & Drains are open when reported fixed.	
		Inserts a scram after discharge volume has drained for ~2 minutes.	
		Reports all rods in.	
	BOP	Maintains reactor pressure as determined by the CRS.	
		Maintains level as directed by the SCO.	
		Restores level to 170 – 200 inches after all rod inserted.	See Enclosure 4 for restart of Condensate & Feedwater.

ENCLOSURE 1

SECURING HPCI INJECTION

8.12.1 INITIAL CONDITIONS

1. **WHEN DIRECTED** BY 2EOP-01-LPC TO "TERMINATE AND PREVENT" HPCI INJECTION, **OR** ☐
2. **WHEN DIRECTED** BY 0EOP-01-RXFP TO "TERMINATE AND PREVENT" HPCI INJECTION, **OR** ☐
3. **WHEN** PERMISSION GIVEN BY THE UNIT CRS TO SECURE HPCI INJECTION WITH A HPCI AUTO START SIGNAL PRESENT. ☐

8.12.1 PROCEDURAL STEPS

1. **IF** HPCI IS **NOT** OPERATING, **PERFORM** THE FOLLOWING:
   a. **PLACE** HPCI AUXILIARY OIL PUMP CONTROL SWITCH IN PULL-TO-LOCK. ☐
2. **IF** HPCI IS OPERATING, **PERFORM** THE FOLLOWING:
   a. **DEPRESS AND HOLD** THE HPCI TURBINE TRIP PUSHBUTTON. ☐
   b. **WHEN** HPCI TURBINE SPEED IS 0 RPM, **AND** HPCI TURBINE CONTROL VALVE, E41-V9 IS CLOSED, **THEN PLACE** HPCI AUXILIARY OIL PUMP CONTROL SWITCH IN PULL-TO-LOCK. ☐
   c. **WHEN** HPCI TURB BRG OIL PRESS LO, A-01 4-2, IS SEALED IN, **THEN RELEASE** THE HPCI TURBINE TRIP PUSHBUTTON. ☐
   d. **ENSURE** HPCI TURBINE STOP VALVE, E41-V8, **AND** HPCI TURBINE CONTROL VALVE, E41-V9, REMAIN CLOSED, **AND** HPCI DOES **NOT** RESTART. ☐

ENCLOSURE 2

HPCI INJECTION IN EOPs

1. **IF** HPCI IS TRIPPED ON HIGH WATER LEVEL, **DEPRESS** HIGH WATER LEVEL SIGNAL RESET, E41-S25, PUSH BUTTON, **AND ENSURE** THE INDICATING LIGHT IS OFF. ☐
2. **ENSURE** AUXILIARY OIL PUMP IS **NOT** RUNNING ☐
3. **ENSURE** E41-V9 AND E41-V8 ARE CLOSED ☐
4. **OPEN** E41-F059 ☐
5. **PLACE** HPCI FLOW CONTROL, E41-FIC-R600, IN MANUAL (M), **AND ADJUST** OUTPUT DEMAND TO APPROXIMATELY MIDSCALE, USING THE MANUAL LEVER. ☐
6. **START** VACUUM PUMP **AND LEAVE** IN START ☐
7. **OPEN** E41-F001 ☐
8. **START** AUXILIARY OIL PUMP **AND LEAVE** IN START ☐
9. **OPEN** E41-F006, IMMEDIATELY AFTER E41-V8 HAS DUAL INDICATION ☐
10. **ENSURE** E41-V9 AND E41-V8 ARE OPEN ☐
11. **WHEN** SPEED STOPS INCREASING, **THEN ADJUST** SPEED TO APPROXIMATELY 2100 RPM ☐
12. **ADJUST** HPCI FLOW CONTROL, E41-FIC-R600, TO OBTAIN DESIRED FLOW RATE ☐
13. **ENSURE** E41-F012 IS CLOSED WHEN FLOW IS GREATER THAN 1400 GPM ☐
14. **ADJUST** HPCI FLOW CONTROL, E41-FIC-R600, SETPOINT TO MATCH SYSTEM FLOW, AND THEN PLACE E41-FIC-R600 IN AUTO (A) ☐
15. **ENSURE** E41-F025 AND E41-F026 ARE CLOSED ☐
16. **START** SBGT (OP-10) ☐
17. **ENSURE** BAROMETRIC CNDSR CONDENSATE PUMP IS OPERATING ☐

ENCLOSURE 3

Terminating and Preventing Injection From Condensate and Feedwater During EOP's (2OP-32)

1. **IF** desired **TRIP** all operating RFPs. ☐
2. **IF** one or more RFPs are in service **IDLE** one RFP as follows:
   a. **IF** two RFPs are operating **THEN TRIP** one. ☐
   b. **PERFORM** either of the following for the operating RFP:
      1. **PLACE** MAN/DFCS control switch to MAN. ☐
      2. **RAPIDLY REDUCE** speed to approximately 1000 rpm with the LOWER/RAISE speed control switch. ☐

      OR

      1. **PLACE** RFPT Speed Control in M (MANUAL) ☐
      2. **SELECT** DEM and **RAPIDLY REDUCE** speed to approximately 2550 rpm. ☐
3. **CLOSE** the following valves:
   – FW HTR 5A OUTLET VLVS, FW-V6 ☐
   – FW HTR 5B OUTLET VLVS, FW-V8 ☐

   OR

   – FW HTR 4A INLET VLV, FW-V118 ☐
   – FW HTR 4B INLET VLV, FW-V119 ☐
4. **ENSURE** the SULCV is closed by performing the following:
   a. **PLACE** SULCV, in M (Manual). ☐
   b. **SELECT** DEM and **DECREASE** signal until VALVE DEM indicates 0%. ☐
5. **ENSURE** FW-V120, is closed. ☐

ENCLOSURE 4

Feedwater Level Control Following a Reactor Scram

NOTE	This attachment is **NOT** to be used for routine system operation.

1. **ENSURE** the following:
   - FW-V6 **AND** FW-V8 **OR** FW-V118 **AND** FW-V119 closed ☐
   - FW-FV-177 closed ☐
   - FW-V120 closed ☐
   - FW control MODE SELECT in 1 ELEM ☐
   - SULCV in M (MANUAL) closed ☐
   - B21-F032A **AND/OR** B21-F032B open ☐
2. **PLACE** the MSTR RFPT SP/RX LVL CTL in M (MANUAL), **THEN:** ☐
   - **ADJUST** to 187" ☐
3. **IF** any RFP is running, **THEN**:
   a. **PLACE** RFP A(B) RECIRC VLV, control switch to open ☐
   b. **PLACE** RFPT A(B) SP CTL in M (MANUAL) ☐
4. **IF** no RFP is running, **THEN**:
   a. **PLACE** RFP A(B) RECIRC VLV, control switch to open ☐
   b. **ENSURE** the following:
      - RFP A(B) DISCH VLV, FW-V3(V4) open ☐
      - RFPT A(B) SP CTL in M (MANUAL) at lower limit ☐
      - RFPT A(B) MAN/DFCS control switch in MAN ☐
      - Reactor water level is less than +206 inches **AND** RFPT A&B HIGH LEVEL TRIP reset ☐
   c. **DEPRESS** RFPT A(B) RESET ☐

ENCLOSURE 4

Feedwater Level Control Following a Reactor Scram

d. **ENSURE** RFPT A(B) LP **AND** HP STOP VLVS open ☐

e. **ROLL** RFPT A(B) to 1000 rpm by depressing RFP A(B) START ☐

f. **RAISE** RFPT A(B) to approximately 2550 rpm using the LOWER/RAISE control switch ☐

g. **DEPRESS** RFPT A(B) DFCS CTRL RESET ☐

5. **ENSURE** MAN/DFCS control switch in DFCS ☐
6. **RAISE** RFPT A(B) SP CTL speed until discharge pressure is greater than or equal to 100 psig above reactor pressure ☐
7. **ADJUST** SULCV to establish desired injection ☐
8. **IF** desired, **THEN PLACE** SULCV in A (AUTO) ☐
9. **IF** needed, **THEN THROTTLE** FW-V120 ☐
10. **IF** needed, **THEN GO TO** 2OP-32 Section 8.17 for level control ☐

ENCLOSURE 5

Page 1 of 2

ATTACHMENT 8A
Page 1 of 1
Emergency Suppression Pool Cooling Using Loop A (2OP-17)

NOTE:	This attachment is NOT to be used for normal system operations.

START RHR SW A LOOP (CONV)

- OPEN *SW-V101* ☐
- CLOSE *SW-V143* ☐
- START CSW PUMPS AS NEEDED ☐
- IF LOCA SIGNAL IS PRESENT THEN PLACE RHR SW BOOSTER PUMPS A & C LOCA OVERRIDE SWITCH TO MANUAL OVERRIDE ☐
- START RHR SW PMP ☐
- ADJUST *E11-PDV-F068A* ☐
- ESTABLISH CLG WTR TO VITAL HDR ☐
- START ADDITIONAL RHR SW PUMP AND ADJUST FLOW AS NEEDED ☐

START RHR SW A LOOP (NUC)

- OPEN *SW-V105* ☐
- OPEN *SW-V102* ☐
- CLOSE *SW-V143* ☐
- START PUMPS ON NSW HDR AS NEEDED ☐
- IF LOCA SIGNAL IS PRESENT THEN PLACE RHR SW BOOSTER PUMPS A & C LOCA OVERRIDE SWITCH TO MANUAL OVERRIDE ☐
- START RHR SW PMP ☐
- ADJUST *E11-PDV-F068A* ☐
- ESTABLISH CLG WTR TO VITAL HDR ☐
- START ADDITIONAL RHR SW PUMP AND ADJUST FLOW AS NEEDED ☐

START RHR LOOP A

- IF LOCA SIGNAL IS PRESENT, THEN VERIFY COOLING LOGIC IS MADE UP ☐
- IF *E11-F015A* IS OPEN, THEN CLOSE *E11-F017A* ☐
- START LOOP A RHR PMP ☐
- OPEN *E11-F028A* ☐
- THROTTLE *E11-F024A* ☐
- THROTTLE *E11-F048A* ☐
- START ADDITIONAL LOOP A RHR PMP AND ADJUST FLOW AS NEEDED ☐

ENCLOSURE 5

ATTACHMENT 8B
Page 1 of 1
Emergency Suppression Pool Cooling Using Loop B (2OP-17)

NOTE: This attachment is **NOT** to be used for normal system operations.

START RHR SW B LOOP (NUC)

- OPEN *SW-V105* ☐
- CLOSE *SW-V143* ☐
- START PMPS ON NSW HDR AS NEEDED ☐
- IF LOCA SIGNAL IS PRESENT THEN PLACE RHR SW BOOSTER PUMPS B & D LOCA OVERRIDE SWITCH TO MANUAL OVERRIDE ☐
- START RHR SW PMP ☐
- ADJUST *E11-PDV-F068B* ☐
- ESTABLISH CLG WTR TO VITAL HDR ☐
- START ADDITIONAL RHR SW PUMP AND ADJUST FLOW AS NEEDED ☐

START RHR SW B LOOP (CONV)

- OPEN *SW-V101* ☐
- OPEN *SW-V102* ☐
- CLOSE *SW-V143* ☐
- START CSW PUMPS AS NEEDED ☐
- IF LOCA SIGNAL IS PRESENT THEN PLACE RHR SW BOOSTER PUMPS B & D LOCA OVERRIDE SWITCH TO MANUAL OVERRIDE ☐
- START RHR SW PMP ☐
- ADJUST *E11-PDV-F068B* ☐
- ESTABLISH CLG WTR TO VITAL HDR ☐
- START ADDITIONAL RHR SW PUMP AND ADJUST FLOW AS NEEDED ☐

START RHR LOOP B

- IF LOCA SIGNAL IS PRESENT, THEN VERIFY COOLING LOGIC IS MADE UP ☐
- IF *E11-F015B* IS OPEN, THEN CLOSE *E11-F017B* ☐
- START LOOP B RHR PMP ☐
- OPEN *E11-F028B* ☐
- THROTTLE *E11-F024B* ☐
- THROTTLE *E11-F048B* ☐
- START ADDITIONAL LOOP B RHR PMP AND ADJUST FLOW AS NEEDED ☐

ATTACHMENT 1 - Scenario Quantitative Attribute Assessment

Category	NUREG 1021 Rev. 2 Supp. 1 Req.	Scenario Content
Total Malfunctions	5-8	8
Malfunctions after EOP Entry	1-2	3
Abnormal Events	2-4	2
Major Transients	1-2	1
EOPs Used	1-2	2
EOP Contingency	0-2	1
Run Time	60-90 min	90
Crew Critical Tasks	2-3	3
Tech Specs	2	2
Instrument / Component Failures before Major	2 – OATC 2 - BOP	4
Instrument / Component Failures after Major	2	2
Normal Operations	1	1
Reactivity manipulation	1	1

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ATTACHMENT 2 – Shift Turnover

Brunswick Unit 2 Plant Status					
Station Duty Manager:				Workweek Manager:	
Mode:	1	Rx Power:	95%	Gross*/Net MWe*:	934 / 909
Plant Risk: Current EOOS Risk Assessment is:				Green	
SFP Time to 200 Deg F:	128.7 hrs			Days Online:	142 days
Turnover:	Feedwater Temperature Reduction will be implemented this weekend				
Protected Equipment:					
Comments:	APRM 4 has failed downscale and is bypassed. 2C TCC Pump is in service on Unit One. Swap CRD Pumps (place CRD Pump B in service and remove CRD Pump A from service for maintenance).				

Facility: Brunswick Scenario No.: NRC 2 Op-Test No.: FINAL

Examiners: ______ Operators: SRO ______

RO ______

BOP ______

Initial Conditions: The plant is operating at 95% power at end of cycle.

Turnover: 0PT-40.2.11, Main Generator Voltage Regulator Manual and Automatic Operational Check is scheduled to be performed. Core Spray 2A is inoperable.

Event No.	Malf. No.	Event Type*	Event Description
1	NA	N	Perform 0PT-40.2.11
2	NA	R	Raise power to 100% rated.
3	ZUA343	C-BOP C-SRO	Off Gas Filter Differential High
4	ES013F	C-RO C-SRO	HPCI Logic Bus A auto start fails – Tech Spec.
5	CW036F (A)	C-BOP C-SRO	CSW A trips on overcurrent – **AOP-19.0** - Tech Spec.
6	EE020F	C-RO C-SRO	SAT Relay Trip – Recirc Pumps Trip – **AOP-4.0**
7	NA	M	Reactor Scram
	EE009A	M	LOOP – **AOP-36.1**
	DG026F		DG3 auto starts and trips.
	DG006F	C	DG4 output breaker fails to auto close – closes when manually closed.
8	CA020F		SRV fails to close – tailpipe rupture – **AOP-30.0**
9	ES026F	C	RCIC Injection Valve motor thermal overload
10	RH020F	M	DW Sprays Fail – ED on PSP

* (N)ormal, (R)eactivity, (I)nstrument, (C)omponent, (M)ajor

SCENARIO DESCRIPTION SUMMARY – 2015 NRC Scenario 2

Event	Description
1	Crew will perform 0PT-40.2.11, Main Generator Voltage Regulator Manual and Automatic Operational Check.
2	Crew will raise power from 80% to 100%.
3	A Clogged Off-gas filter will require a response IAW 2APP UA-03, 5-3 and 2OP-30 to place the standby filter in service.
4	The HPCI logic power fuse will blow requiring HPCI to be manually isolated per the APP and declared Inoperable per TS 3.5.1, Condition D and also Condition E (HPCI & CS Inop) which requires restoration of one of the two within 72 hours.
5	A 4KV Bus E3 motor overload alarm will be received, followed by a CSW Pump A trip alarm, but the pump will fail to trip. The crew should trip CSW Pump A and start CSW Pump C per the APPs. CSW Pump C will fail to auto start on low header pressure.
6	The SAT will trip and lockout on fault, resulting in a trip of both Reactor Recirculation Pumps. The crew is required to insert a manual reactor scram per AOP-04.0.
7	When the reactor scram is inserted and the turbine is tripped, a loss of off-site power will result since the SAT is not available. Diesel Generator #3 auto starts and energizes Bus E3. Diesel Generator #4 starts but the output breaker fails to close. Operator action is required to close the output breaker to energize E4. The crew will respond per AOP-36.1.
8	When SRV F is opened for pressure control it will not reclose. A downcomer failure will cause Drywell and Torus pressure to rise.
9	When RCIC is started for level control, the Injection Valve thermals out but can be reset locally.
10	When RHR 2A is placed in drywell spray, the outboard spray valve (F016A) will fail to open. When crew attempts to place RHR 2B in service spray logic will fail to energize and Drywell Spray will not immediately be available. Emergency Depressurization will be required based on PSP.

CREW CRITICAL TASKS

Description
Ensure DG4 starts and close the output breaker to energize E4.
Emergency Depressurize the Reactor when Pressure Suppression Pressure (PSP) cannot be maintained in the safe region.

DUKE ENERGY

BRUNSWICK TRAINING SECTION

OPERATIONS TRAINING

INITIAL LICENSED OPERATOR

SIMULATOR EVALUATION GUIDE

2015 NRC SCENARIO 2

RBCCW PUMP TRIP, HPCI LOGIC POWER FAILURE, LOOP, LOCA, SRV TAILPIPE FAILURE, ED ON PSP

REVISION 0

Developer: Lou Sosler	**Date:** 9/11/2015
Technical Review: John Biggs	**Date:** 9/23/2015
Validator: Thomas Baker	**Date:** 9/11/2015
Validator: Brian Moschet	**Date:** 9/11/2015
Facility Representative: Jerry Pierce	**Date:** 9/23/1015

REVISION SUMMARY	
0	Exam scenario for 2015 NRC Exam.

TABLE OF CONTENTS

1.0 SCENARIO OUTLINE

Event	Malf. No.	Type*	Event Description
1	NA	N	Perform 0PT-40.2.11
2	NA	R	Raise power to 100% rated.
3	ZUA343	C-BOP C-SRO	Off Gas Filter Differential High
4	ES013F	C-RO C-SRO	HPCI Logic Bus A auto start fails – Tech Spec.
5	CW036F (A)	C-BOP C-SRO	CSW A trips on overcurrent – **AOP-19.0** - Tech Spec.
6	EE020F	C-RO C-SRO	SAT Relay Trip – Recirc Pumps Trip – **AOP-4.0**
7	NA	M	Reactor Scram
	EE009A	M	LOOP – **AOP-36.1**
	DG026F		DG3 auto starts and trips.
	DG006F	C	DG4 output breaker fails to auto close – closes when manually closed.
8	CA020F		SRV fails to close – tailpipe rupture – **AOP-30.0**
9	ES026F	C	RCIC Injection Valve motor thermal overload
10	RH020F	M	DW Sprays Fail – ED on PSP
*(N)ormal, (R)eactivity, (C)omponent or Instrument, (M)ajor			

2.0 SCENARIO DESCRIPTION SUMMARY

Event	Description
1	Crew will perform 0PT-40.2.11, Main Generator Voltage Regulator Manual and Automatic Operational Check.
2	Crew will raise power from 80% to 100%.
3	A Clogged Off-gas filter will require a response IAW 2APP UA-03, 5-3 and 2OP-30 to place the standby filter in service.
4	The HPCI logic power fuse will blow requiring HPCI to be manually isolated per the APP and declared Inoperable per TS 3.5.1, Condition D and also Condition E (HPCI & CS Inop) which requires restoration of one of the two within 72 hours.
5	A 4KV Bus E3 motor overload alarm will be received, followed by a CSW Pump A trip alarm, but the pump will fail to trip. The crew should trip CSW Pump A and start CSW Pump C per the APPs. CSW Pump C will fail to auto start on low header pressure.
6	The SAT will trip and lockout on fault, resulting in a trip of both Reactor Recirculation Pumps. The crew is required to insert a manual reactor scram per AOP-04.0.
7	When the reactor scram is inserted and the turbine is tripped, a loss of off-site power will result since the SAT is not available. Diesel Generator #3 auto starts and energizes Bus E3. Diesel Generator #4 starts but the output breaker fails to close. Operator action is required to close the output breaker to energize E4. The crew will respond per AOP-36.1.
8	When SRV F is opened for pressure control it will not reclose. A downcomer failure will cause Drywell and Torus pressure to rise.
9	When RCIC is started for level control, the Injection Valve thermals out but can be reset locally.
10	When RHR 2A is placed in drywell spray, the outboard spray valve (F016A) will fail to open. When crew attempts to place RHR 2B in service spray logic will fail to energize and Drywell Spray will not immediately be available. Emergency Depressurization will be required based on PSP.

3.0 CREW CRITICAL TASKS

Description
Ensure DG4 starts and close the output breaker to energize E4.
Emergency Depressurize the Reactor when Pressure Suppression Pressure (PSP) cannot be maintained in the safe region.

4.0 TERMINATION CRITERIA

Once the reactor is depressurized, and level is being restored to normal band, the scenario may be terminated.

5.0 IMPLEMENTING REFERENCES

NOTE: Refer to the most current revision of each Implementing Reference.

Number	Title
0PT-40.2.11	MAIN GENERATOR VOLTAGE REGULATOR MANUALL AND AUTOMATIC OPERATIONS CHECK
2APP UA-03, 5-3	OFFGAS STBY FILTER DIFF-HIGH
APP 2A-1, 5-5	HPCI LOGIC BUS A PWR FAILURE
APP 2A-1, 6-4	HPCI COND STORAGE TNK WTR LVL LO
APP 2UA-17, 6-1	E3 4KV MOTOR OVLD
APP 2UA-01, 1-8	CSW PUMP A TRIP
2AOP-4.0	LOW CORE FLOW
0AOP-19.0	CONVENTIONAL SERVICE WATER SYSTEM FAILURE
0AOP-36.1	LOSS OF ANY 4160V BUSES OR 480V E-BUSES
0AOP-30.0	SAFETY/RELIEF VALVE FAILURES

6.0 SETUP INSTRUCTIONS

1. **PERFORM** TAP-409, Miscellaneous Simulator Training Guidelines, Attachment 5, Checklist for Simulator Exam Security.

2. **RESET** the Simulator to IC-25.

3. **ENSURE** the RWM is set up as required for the selected IC.

4. **ENSURE** appropriate keys have blanks in switches.

5. **RESET** alarms on SJAE, MSL, and RWM NUMACs.

6. **ENSURE** no rods are bypassed in the RWM.

7. **PLACE** all SPDS displays to the Critical Plant Variable display (#100).

8. **ENSURE** hard cards and flow charts are cleaned up

9. **TAKE** the SIMULATOR OUT OF FREEZE

10. **ALIGN** the plant as follows:

Manipulation
1. Ensure 2C TCC pump is in service on Unit One. Part of Scenario load.
2. Lower power to 80% using Recirc Flow.

11. **LOAD** Scenario File.

12. **IF desired**, take a **SNAPSHOT** and save into an available IC for later use.

13. **PLACE** a clearance on the following equipment.

Component	**Position**
Core Spray Pump A	

14. **INSTALL** Protected Equipment signage and **UPDATE** RTGB placard as follows:

    All remaining low pressure ECCS Systems.

15. **ENSURE** each Implementing References listed in Section 7 is intact and free of marks.

16. **ENSURE** all materials in the table below are in place and marked-up to the step identified.

Required Materials

None

17. **ENSURE** Station Duty Manager and Work Week Manager names are filled in on the Shift Turnover Sheet.

18. **ADVANCE** the recorders to prevent examinees from seeing relevant scenario details.

19. **PROVIDE** Shift Briefing sheet for the CRS.

20. **VERIFY** all actions contained in TAP-409, Miscellaneous Simulator Training Guidelines, Attachment 4, Simulator Training Instructor Checklist, are complete.

7.0 INTERVENTIONS

TRIGGERS

Trig	Type	ID
1	Annunciator	ZUA343 - [OFF GAS FILTER DIFF-HIGH]
2	Malfunction	ES013F - [HPCI LOGIC BUS A AUTO START FAILS]
2	Malfunction	ES014F - [INADVERTANT HPCI SYS INITIATION]
3	Annunciator	ZUA118 - [CONV HDR SW PUMP A TRIP]
3	Malfunction	CW036F - [CONV SERVICE WATER PUMP MOTOR OVERLOAD]
4	Malfunction	EE020F - [UNIT 2 SAT RELAY FAILURE]
5	Malfunction	CA020F - [SRV F TAIL PIPE RUPTURE]
6	Malfunction	ES026F - [RCIC INJECTION VLV MOTOR OVLD]
7	Malfunction	ES004F - [ADS VALVE F FAILS OPEN]
8	Malfunction	DG026F - [DG3 DIFFERENTIAL FAULT]
9	Malfunction	RH020F - [2-E11-V32 OPEN COMMAND]
10	Trigger Command	and:ZUA118
11	Annunciator	ZA322 - [AUTO DEPRESS CONTROL PWR FAILURE]
11	DO Override	Q1508LGJ - [SRV VLV B21-F013F GREEN]
11	DO Override	Q1508RRJ - [SRV VLV B21-F013F RED]
12	Remote Function	HP_ZVHP041M - [SUPP SUCTION VLV E41-F041]
13	Remote Function	HP_ZVHP042M - [TORUS SUCTION VLV E41-F042]
14	Remote Function	ED_ZIEDH08 - [PNL 2AB PWR (E7=NORM/E8=ALT)]
15	Remote Function	ED_ZIEDH11 - [PNL 2AB-RX PWR (E7=NORM/E8=ALT)]
16	Remote Function	ED_ZIEDHX0 - [PNL 32AB PWR (E7=NORM/E8=ALT)]
17	Remote Function	SW_VHSW146L - [CONV SW TO RBCCW HXS V146]
18	Remote Function	ED_IARKAI0 - [X-TIE BKR E8-E7 (AI0) RACK STATUS]
18	Remote Function	ED_IARKAX5 - [X-TIE BKR E7-E8 (AX5) RACK STATUS]

Trig #	Trigger Text
6	Q1619RRM - [RCIC INJECT VLV E51-F013 RED]
7	Q1508RRJ - [SRV VLV B21-F013F RED]
8	Q4F06DG8 - [LOADED (RED) DG-3]
9	K1D26ENN - [CONT SPRAY VLV E11-F016A]
10	K4B39EP4 - [CONV HDR SW PMP A DISCH VLVS]

MALFUNCTIONS

Malf ID	Mult ID	Description	Current Value	Target Value	Rmp time	Actime	Dactime	Trig
DG006F	DG 4	DG OUTPUT BREAKER FAIL TO AUTO CLOSE	True	True				
ES013F		HPCI LOGIC BUS A AUTO START FAILS	False	True				2
ES014F		INADVERTANT HPCI SYS INITIATION	False	True				2
CW036F	A	CONV SERVICE WATER PUMP MOTOR OVERLOAD	False	True				3
EE020F		UNIT 2 SAT RELAY FAILURE	False	True				4
CA020F		SRV F TAIL PIPE RUPTURE	False	True				5
ES026F		RCIC INJECTION VLV MOTOR OVLD	False	True				6
ES004F		ADS VALVE F FAILS OPEN	False	True				7
DG026F		DG3 DIFFERENTIAL FAULT	False	True		00:02:00		8
RH020F	E11-F016A	CONTAINMENT SPRAY * VLV E11-F016A	False	True				9

REMOTES

Remf Id	Mult Id	Description	Current Value	Target Value	Rmp time	Actime	Trig
ED_IABKCF05		BKR CTL DC FUSES CORE SPRAY PUMP 2A	OUT	OUT			
CC_IACW4518		2C TBCCW PUMP UNIT ALIGNMENT	1	1			
HP_ZVHP041M		SUPP SUCTION VLV E41-F041	ON	OFF			12
HP_ZVHP042M		TORUS SUCTION VLV E41-F042	ON	OFF			13
ED_ZIEDH08		PNL 2AB PWR (E7=NORM/E8=ALT)	NORMAL	ALT			14
ED_ZIEDH11		PNL 2AB-RX PWR (E7=NORM/E8=ALT)	NORMAL	ALT			15
ED_ZIEDHX0		PNL 32AB PWR (E7=NORM/E8=ALT)	NORMAL	ALT			16
SW_VHSW146L		CONV SW TO RBCCW HXS V146	SHUT	OPEN			17
ED_IARKAX5		X-TIE BKR E7-E8 (AX5) RACK STATUS	OUT	IN			18
ED_IARKAI0		X-TIE BKR E8-E7 (AI0) RACK STATUS	OUT	IN			18

PANEL OVERRIDES

Tag ID	Description	Position / Target	Actual Value	Override Value	Rmp time	Actime	Dactime	Trig
K4B41A	CONV HDR SW PMP C DISCH VLVS	AUTO	ON	OFF				
K1727A	CONT SPRAY VLV CONTROL	NORMAL	ON	ON				
K1727A	CONT SPRAY VLV CONTROL	MANUAL	OFF	OFF				
K1727A	CONT SPRAY VLV CONTROL	RESET	OFF	OFF				
Q1508RRJ	SRV VLV B21-F013F RED	ON/OFF	OFF	OFF				11
Q1508LGJ	SRV VLV B21-F013F GREEN	ON/OFF	ON	OFF				11

ANNUCIATOR OVERRIDES

Window	Description	Tagname	Override Type	OVal	AVal	Actime	Dactime	Trig
4-3	OFF GAS FILTER DIFF-HIGH	ZUA343	ON	ON	OFF			1
1-8	CONV HDR SW PUMP A TRIP	ZUA118	ON	ON	OFF	00:03:00		3
2-2	AUTO DEPRESS CONTROL PWR FAILURE	ZA322	ON	ON	OFF			11

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8.0 OPERATOR RESPONSE AND INSTRUCTIONAL STRATEGIES

EVENT 1: SHIFT TURNOVER - 0PT-40.2.11	
Simulator Operator Actions	
	Ensure Monitored Parameters is open and Scenario Based Testing Variables are loaded.

Simulator Operator Role Play	
	As operator at the Excitation Regulator and Control cubicle in the Turbine Building on the 70 ft elevation west, monitor regulator output.
	As Load Dispatcher, provide name and acknowledge Main Generator Voltage Regulator will be placed in MANUAL.
	As Load Dispatcher, provide name and acknowledge Main Generator Voltage Regulator will be placed in AUTOMATIC.
	Alt Power performed SAT by NE.

Evaluator Notes	
Plant Response:	None
Objectives:	Perform 0PT-40.2.11, Main Generator Voltage Regulator Manual and Automatic Check
Success Path:	Perform PT IAW 0PT-40.2.11
Event Termination:	Go to Event 2 at the direction of the Lead Evaluator.

EVENT 1: SHIFT TURNOVER - 0PT-40.2.11

Time	Pos	EXPECTED Operator Response	Comments
	SRO	Conduct shift turnover shift briefing.	
		Direct RO to perform 0PT-40.2.11, Main Generator Voltage Regulator Manual and Automatic Operational Check.	
		Determine Tech Spec Action statement for 2A Core Spray inoperable" **T.S. 3.5.1:** **A.1 Restore low pressure ECCS injection spray subsystem to operable status within 7 days.**	
	RO	Monitors the plant	
	BOP	Perform 0PT-40.2.11	
		Operate 70CS (Gen Manual Volt Adj Rheo)	
		Ensure 43CS (Regulator Mode Selector) in AUTO. Station an operator at the Excitation Regulator and Control cubicle in the Turbine Building on the 70 ft elevation west to monitor regulatoroutput during the following steps Raise 70CS (Gen Manual Volt Adj Rheo) until the Upper Limit light comes ON Lower 70CS (Gen Manual Volt Adj Rheo) until the Low Limit light comes ON Using 70CS (Gen Manual Volt Adj Rheo) on the RTGB, null Gen Volt Reg Diff Volt meter	

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EVENT 1: SHIFT TURNOVER - 0PT-40.2.11

Time	Pos	EXPECTED Operator Response	Comments
		IF D1VM (D.C. Reg. Output) variation was smooth AND in the same direction as rheostat movement, THEN perform the following: Notify the Load Dispatcher the main generator voltage regulator is being placed in MANUAL. Person Notified Document the Load Dispatcher notification in the log Place 43CS (Regulator Mode Selector) in MAN.	
		Operate 90CS (Gen Auto Volt Adj Rheo) Raise 90CS (Gen Auto Volt Adj Rheo) until the Upper Limit light comes ON Lower 90CS (Gen Auto Volt Adj Rheo) until the Low Limit light comes ON. Null Gen Volt Reg Diff Volt meter on the RTGB using 90CS (Gen Auto Volt Adj Rheo). IF A1VM (A.C. Reg. Output) variation was smooth AND in the same direction as rheostat movement, THEN perform the following: Place 43CS (Regulator Mode Selector) in AUTO Notify the Load Dispatcher the main generator voltage regulator is in AUTOMATIC	

EVENT 2: RAISE POWER	
Simulator Operator Actions	

Simulator Operator Role Play	
	If contacted as the RE to address thermal limits, inform crew that you will monitor core performance on the computer.
	If asked as RE about how to raise power, ask for SRO suggestion and agree.

Evaluator Notes	
Plant Response:	Power will be raised using Recirc flow from 80% toward 100%.
Objectives:	SRO - Direct actions for power increase using Recirc flow. RO – Raise power. BOP – Monitor balance of plant.
Success Path:	Power is raised using Recirc flow from 80% to 100% monitoring the Power-to-Flow Map and balance of plant.
Event Termination:	Go to Event 3 at the direction of the Lead Evaluator.

EVENT 2: RAISE POWER

Time	Pos	EXPECTED Operator Response	Comments
	SRO	Directs power to be raised using recirculation flow	
		Conducts reactivity briefing	May ask for reactivity team.
	RO	May reference 2OP-02 section 7.1	
		Request peer checker / reactivity team.	
		Raises power using recirculation flow to ~100% power. Raise RR Pump speed by depressing the Master Raise Medium pushbutton Continues Raising Recirc pump speed until 100% power.	
	BOP	Verifies operation on the Power to Flow Map	
		Monitors and adjusts balance of plant conditions IAW 0GP-4	

EVENT 3: OFF GAS FILTER HIGH DP	
Simulator Operator Actions	
	At the direction of the lead evaluator, **Initiate Trigger 1** to bring in Off-Gas Filter Diff-Hi annunciator
	Once filter is swapped, delete malfunction (annunciator override).

Simulator Operator Role Play	
	IF contacted as Outside AO to verify Off-gas filter Diff pressure, report local DP indication is reading 13 inches water.
	IF contacted as Unit One report steps 6.3.3.2 and 3 are complete. (1-OG-FV-244-4, 1-OG-FV-244-5, and 1-AOG-HCV-101 are closed)
	IF contacted as AO report 1-OG-CD-V7 is CLOSED (Step 4)
	IF contacted as AO report 2-OG-CD-V7 is OPEN (Step 5)
	IF contacted as Outside AO to verify Off-gas filter Diff pressure after filter swap, report local DP indication is reading 3 inches water.
	2AOG-HCV-101 is in OPEN position.

Evaluator Notes	
Plant Response:	Off-Gas Filter Diff-Hi alarm annunciates
Objectives:	SRO -Direct actions in response to a Off-gas Filter Diff-Hi alarm RO – Monitor reactor BOP – Respond to clogged off-gas filter IAW APP and 2OP-30
Success Path:	Swap Off-gas filters per OP-30
Event Termination:	Go to Event 4 at the direction of the Lead Evaluator.

EVENT 3: OFF GAS FILTER HIGH DP

Time	Pos	EXPECTED Operator Response	Comments
	SRO	Direct crew to perform the actions of OFF GAS FILTER DIFF-HIGH alarm	
		Direct crew to swap Off-gas filters per OP-30 Section 6.3.3	
	BOP	Respond to OFF GAS FILTER DIFF-HIGH alarm	Dispatch Outside AO to verify Off-gas filter Diff Hi locally
		Place Off-gas Stby filter in service per Op-30 Section 6.3.3 as follows: ☐ Ensure 1-OG-FV-244-4 and 244-5 are closed ☐ Ensure 1-AOG-HCV-101 is closed ☐ Ensure 1-OG-CD-V7 is closed (Local) ☐ OPEN 2-OG-CD-V7 (Local) ☐ OPEN 2-OG-FV-244-4 and 244-5 ☐ OPEN 2-AOG-HCV-101 Remove Off-gas Filter from service as follows: ☐ CLOSE 2-OG-FV-244-1,2,3 ☐ Ensure Filter d/P is less than 10” water Place 2-AOG-HCV-101 control switch to OPEN (Local)	Unit 1
	RO	Monitor Plant	

EVENT 4: HPCI LOGIC BUS A FAILURE	
Simulator Operator Actions	
	When directed by lead evaluator, **Initiate Trigger 2** to activate HPCI fuse failure.
	If requested to open breakers for E41-F041 and F042 (MCC 2XDA) when valves are shut, monitor valve positions on panel mimic (P601 Section A2). At direction of RO, when E41-F041 indicates closed, OPEN breaker. At direction of RO, when E41-F042 indicates closed, OPEN breaker.

Simulator Operator Role Play	
	If asked as AO to investigate, report all circuit breakers in DC SWBD 2A & Panel 4A are closed
	If asked as I&C to investigate, wait 2 minutes and report fuse E41A-F1 in panel P620 is blown (blows again if replaced).
	If asked as WCC/OC SRO for clearance or Equipment Control tags, acknowledge request.
	If requested to open breakers for E41-F041 and F042 (MCC 2XDA) when valves are shut, monitor valve positions on panel mimic (P601 Section A2). **(Triggers 12 and 13).**

Evaluator Notes
Plant Response: The HPCI logic power fuse will blow requiring HPCI to be manually isolated per the APP and declared Inoperable per TS 3.5.1.
Objectives: SRO - Declare HPCI Inoperable RO - Recognize logic failure and Isolate HPCI BOP – Monitor Plant
Success Path: HPCI declared inoperable IAW TS 3.5.1 and isolated IAW APP
Event Termination: Go to Event 5 at the direction of the Lead Evaluator.

EVENT 4: HPCI LOGIC BUS A FAILURE			
Time	**Pos**	**EXPECTED Operator Response**	**Comments**
	SRO	Direct annunciator response for A-1: ☐ 5-5, HPCI LOGIC BUS A PWR FAILURE ☐ 6-4, HPCI COND STORAGE TNK WTR LVL LO	
		May identify requirement to initiate an impairment IAW 0PLP-01.5. (This action may be directed to Ops Center SRO)	HPCI is considered a Train A system for ASSD.
		Directs BOP to monitor the plant.	
		Determines depressurization of steam supply is NOT required.	
		Contacts I&C to investigate HPCI LOGIC BUS A PWR FAILURE	

EVENT 4: HPCI LOGIC BUS A FAILURE

Time	Pos	EXPECTED Operator Response	Comments
	SRO	Refers to Tech Spec 3.5.1 ECCS —Operating and Determines: CONDITION A. One low pressure ECCS injection/spray subsystem inoperable. REQUIRED ACTION A.1 Restore low pressure ECCS injection/spray subsystem to OPERABLE status within 7 days. CONDITION D REQUIRED ACTION: D.1 Verify by administrative means RCIC System is OPERABLE. Immediately AND D.2. Restore HPCI System to OPERABLE status. 14 days CONDITION E REQUIRED ACTION: E.1 Restore HPCI System to OPERABLE status. 72 hours OR E.2 Restore low pressure ECCS injection/spray subsystem to OPERABLE status. 72 hours	Condition A existed at turnover with Core Spray Pump 2A under clearance.
		May request equipment control tags to support abnormal HPCI system alignment.	

EVENT 4: HPCI LOGIC BUS A FAILURE

Time	Pos	EXPECTED Operator Response	Comments
	RO	Acknowledge and report annunciators A-1: 5-5 HPCI LOGIC BUS A PWR FAILURE 6-4 HPCI COND STORAGE TNK WTR LVL LO	
		Report HPCI Suction is aligned to both the CST and Suppression Pool.	
		Performs APP A-1 5-5 (HPCI LOGIC BUS A PWR FAILURE) actions: 1. Close the Condensate Storage Tank Suction Valve, E41-F004. 2. Isolate the HPCI Steam Supply per OP-19, Section 8.5. 1) CLOSE STEAM SUPPLY INBOARD ISOL VLV, E41-F002. 2) CLOSE STEAM SUPPLY OUTBOARD ISOL VLV, E41-F003. 3) Depressurizing steam supply is NOT required, but if performed will require performance of 0PT-02.3.1b, Suppression Pool to Drywell Vacuum Breaker Position Check, within 6 hours 3. Close the Turbine Exhaust Vacuum Breaker Valve, E41-F075. Contacts RBAO to standby for opening breakers on MCC 2XDA when the following valves indicate Full Closed. 4. Close the Torus Suction Valve, E41-F041. 5. Close the Torus Suction Valve, E41-F042.	Informs SRO of expected alarm A-1 1-1 HPCI VAC BKR VLV F075/F079 NOT FULL OPEN
		Notifies SRO APP actions are complete and to reference TS 3.5.1 and TRM 3.6.	No impact to TRM 3.6 BUS POWER MONITORS (alarm worked).
	BOP	Monitors the plant and reports CST at normal level.	

EVENT 5: CSW PUMP TRIP	
	Simulator Operator Actions
	At the direction of the Lead Evaluator, **Initiate Trigger 3** to activate CSW Pump A trip
	When CSW Pump 2A is tripped, ensure alarm override ZUA118 is deleted.

	Simulator Operator Role Play
	When asked as OAO to investigate, report 51 device actuation (phase B only before pump trip alarm, all 3 phases's after pump trip alarm) at breaker on E3. (Note: Phase B overcurrent brings in pump motor overload alarm, overcurrent on all 3 phases brings in pump trip alarm, but pump does not trip.)
	If requested as I&C to investigate CSW Pump malfunctions, acknowledge the request.

Evaluator Notes
Plant Response: E3 motor overload alarms. Two minutes later CSW 2A trip alarms, but pump fails to trip. CSW Pump 2C fails to auto start, can be manually started.
Objectives: SRO - Direct entry into AOP-19.0. RO - Monitors reactor plant parameters BOP – Take actions IAW 0AOP-19. 0
Success Path: CSW Pump A is tripped. CSW Pump C is started.
Event Termination: Go to Event 6 at the discretion of the Lead Evaluator.

EVENT 5: CSW PUMP TRIP

Time	Pos	EXPECTED Operator Response	Comments
	SRO	Direct entry into AOP-19.0.	
		Direct CSW Pump 2A be tripped and CSW Pump 2C be started.	
		Direct I&C to investigate CSW Pump 2A overcurrent and trip failure and failure of CSW Pump 2C to auto start.	
		Tech Spec 3.7.2 SW System and UHS Condition C.1 Verify the one OPERABLE CSW pump and one OPERABLE Unit 2 NSW pump are powered from separate 4.16 kV emergency buses Immediately. and C.2 Restore required CSW pump to OPERABLE Status within 7 days.	The SW System is considered OPERABLE when it has two OPERABLE CSW pumps (specifically the CSW 2A and CSW 2C pumps), three site NSW pumps (any combination of Unit 1 and Unit 2 NSW pumps), and an OPERABLE flow path capable of taking suction from the intake structure and transferring the water to the ECCS equipment and the DGs. For a CSW pump to be considered OPERABLE, it must be capable of supplying the CSW header and the NSW header.
	RO	Monitor plant parameters	
	BOP	Dispatch AO to investigate 4KV Motor Overload alarm.	
		Recognize/report trip alarm of CSW Pump 2A and Pump still running.	
		Manually trip CSW Pump 2A and manually start CSW Pump 2C.	

EVENTS 6 and 7: SAT RELAY TRIP – RECIRC PUMPS TRIP – MANUAL SCRAM	
Simulator Operator Actions	
	When directed by the lead evaluator, **Initiate Trigger 4** to trip the SAT.

Simulator Operator Role Play	
	If directed to investigate, report no signs of visible damage to the SAT.
	If asked as I&C or Maintenance to investigate, acknowledge the request

Evaluator Notes	
Plant Response:	SAT Fault, trip of both Recric Pumps, Manual Scram
Objectives:	SRO – Direct Reactor Scram – Enter RSP RO – Report trip of both Recirc Pumps, Scram Reactor BOP – Identify plant electrical response
Success Path:	Identify loss of BOP, trip of Recirc Pumps and Scram Reactor
Event Termination:	Go to Event 8 at the direction of the lead evaluator.

EVENTS 6 and 7: SAT RELAY TRIP – RECIRC PUMPS TRIP – MANUAL SCRAM

Time	Pos	EXPECTED Operator Response	Comments
	SRO	Enter AOP-04.0 and direct a manual reactor scram	
		Enter EOP-01-RSP	
		Direct RPV level be controlled +166-206 inches	
		Direct group isolations, ECCS and DGs verified	
	RO	Diagnose and report SAT failure and loss of Recirc Pumps	
		Insert a manual scram	
		Perform scram Immediate Actions	Enclosure 1
		Operate RCIC/SRVs to maintain level and pressure as directed by CRS	
	BOP	Report status of plant electrical system,	

EVENT 7: LOSS OF OFF-SITE POWER – DG4 FAILURE	
	Simulator Operator Actions
	If requested to align RBCCW to CSW cooling, wait 4 minutes and modify Remote Function SW_VHSW146L, OPEN **(Trigger 17)**
	If requested restart RPS MG sets check status of EPA Breakers (RPS A EPA breakers will probably be Set status since DG3 will most likely be at rated speed/voltage when the turbine is tripped) Modify remote functions under RPS as necessary (will have to start RPS MG set B and close the associated EPA breakers)
	If requested to monitor running DGs, acknowledge alarms using DG Local Alarm Panel (Instructor Aids/Panels) and report alarms if requested
	If requested to swap AB panels, wait 5 minutes then initiate: **Trigger 14** - 2AB to alt. **Trigger 15** - 2AB-RX to alt. **Trigger 16** - 32AB to alt. Report panels swapped to alternate.

	Simulator Operator Role Play

Evaluator Notes	
Plant Response:	When Reactor is Scammed, without SAT, off-site power is lost.
Objectives:	SRO – Respond IAW 0AOP-36.1, RSP, PCCP BOP – Diagnose and report electrical plant status RO – Control Reactor level and pressure
Success Path:	Start DG4 and close breaker to E4

EVENT 7: LOSS OF OFF-SITE POWER – DG4 FAILURE

Time	Pos	EXPECTED Operator Response	Comments
	SRO	Enter and direct EOP-01-RSP	
		Enter and direct the activities of AOP-36.1	
		Direct a UAT backfeed be established	
		Identify DG4 output breaker failure to auto close and direct actions to close breaker to energize E4	***CRITICAL TASK***
	RO	Control Reactor pressure and level as directed by the SRO	
		Start RHR in suppression pool cooling	
		Start CRD per OP-08	
	BOP	Enter and announce AOP-36.1	
		Close the output breaker of DG4 to energize E4	***CRITICAL TASK***
		Direct AO to monitor DG operation	
		Direct actions for UAT backfeed	
		Align available pneumatics to the drywell	

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EVENT 7: LOSS OF OFF-SITE POWER – DG4 FAILURE

Time	Pos	EXPECTED Operator Response	Comments
		Start available Service Water pumps	
		Ensure SW-V103/106 closed and direct AO to open SW-V146	
		Start Control Building HVAC	
		Ensure available drywell cooling is operating	
		Direct AO to start available RPS MG Sets	
		Ensure Service Air cross-tie 2-SA-PV-5071 is open	

EVENTS 8: SRV FAILS TO CLOSE – TAILPIPE RUPTURE	
Simulator Operator Actions	
	Ensure that SRV F stays open when close is attempted, or **Initiate Trigger 11** to fail open SRV F if SRVs are opened in an alternate sequence.
	At the direction of the Lead Evaluator, **Initiate Trigger 5,** to initiate tailpipe rupture.

Simulator Operator Role Play	
	If approval for cross-tie requested from Unit 1, grant permission.
	If cross-tie actions are requested, rack in cross tie breakers for E7-E8, **Initiate Trigger 18.**

Evaluator Notes	
Plant Response:	
Objectives:	SRO – Failure of the tailpipe will cause Drywell and Torus pressures to rise RO – Identify and report SRV not closed, enter 0AOP-30.0 BOP – Continue to execute)AOP-36.1
Success Path:	Rising Drywell and Torus pressures are identified and PSP chart is monitored

EVENTS 8: SRV FAILS TO CLOSE – TAILPIPE RUPTURE

Time	Pos	EXPECTED Operator Response	Comments
	SRO	Execute PCCP and direct Torus and Drywell sprays	
		Monitor PSP	Enclosure 2
		Direct entry into 0AOP-30.0, Safety/Relief Valve failures	
		Enter and direct action of PCCP Before Suppression Chamber pressure reaches 11.5 psig directs SP Spray IAW SEP-03 When Suppression Chamber exceeds 11.5 psig directs DW Spray IAW SEP-02	
	RO	Perform 0AOP-30.0 actions: NOTE: A full open SRV will not reseat until reactor pressure reduces to the reseat pressure for that SRV (approximately 900 to 1100 psig). CYCLE the control switch of the affected safety/relief valve to OPEN and CLOSE OR OPEN and AUTO several times. ENSURE the affected safety/relief valve control switch is left in CLOSE OR AUTO. IF a safety/relief valve is stuck open, THEN PERFORM the following: PULL the fuses in the order listed in Attachment 1 for the affected safety/relief valve. MONITOR the following to determine safety/relief valve position: o Tailpipe Temperatures (ERFIS Screen 241 Other indications as available	Immediate Operator Action of AOP-30 NOTE: Pulling safety/relief valve fuses will de-energize the red and green indicating lights on Panel P601.

		(feed/steam flow mismatch, generator MWE, etc.)	
		Perform Torus and Drywell Sprays as directed	SEP-03 Enclosure 3 SEP-02 Enclosure 4.
		Continue to perform 0AOP-36.1 as directed by SRO	

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EVENTSS 9 and 10: ED - RCIC INJECTION VALVE MOTOR OVERLOAD – RHR SPRAY FAILURE	
	Simulator Operator Actions
	Delete malfunction ES026F when thermal overload is reset for RCIC Injection Valve.
	Delete overrides for RHR Loop B spray logic once ED is commenced.
	When directed by the lead evaluator, place the simulator in FREEZE
	DO NOT RESET THE SIMULATOR PRIOR TO RECEIPT OF CONCURRENCE TO DO SO FROM THE LEAD EXAMINER

	Simulator Operator Role Play
	If asked to check breaker for E51-F013, RCIC Injection valve, on 2XDB, report thermal overload tripped. If asked to reset thermal overload, DELETE malfunction ES026F and report valve reset.
	If asked to check breaker for E11-F016A (MCC 2XC) report thermal overload tripped, if directed to reset thermal overload, report it trips again, if directed to manually open E11-F016A, report valve is bound.
	If asked as I&C, to investigate failure of spray logic for RHR Loop B, acknowledge request. Once PSP is exceeded and Emergency Depressurization is commenced, report loose wire found and deleted overrides for RHR Loop B so that Drywell and Torus sprays can be commenced.
	.

Evaluator Notes	
Plant Response:	
Objectives:	SRO – Direct ED when PSP is exceeded. RO – Perform ED when directed by the SRO BOP – Assist RO at direction of SRO
Success Path:	Reactor is depressurized and water level is in normal band.
Scenario Termination:	***Control Rods are inserted, Reactor is depressurized, level is being restored to normal band.***

EVENTS 9 and 10: ED - RCIC INJECTION VALVE MOTOR OVERLOAD – RHR SPRAY FAILURES			
Time	**Pos**	**EXPECTED Operator Response**	**Comments**
	SRO	Monitor Containment parameters and identify PSP exceeded.	
		Direct Emergency Depressurization when PSP cannot be maintained in the safe region.	***PSP Graph Enclosure 2*** ***CRITICAL TASK***
		Direct investigation of E51-F016	
		Direct investigation of RHR Loop B spray logic.	
		Manage controlled reflood after depressurization.	
	RO/ BOP	Recognize and report failure of E11-F016A to open as thermal overload.	
	RO/ BOP	Dispatch AO to check breaker and attempt to reset thermal overload per the APP.	
	RO/ BOP	***When directed by SRO, Open 7 ADS valves***	***CRITICAL TASK***
	RO/ BOP	Restore water level to normal band.	
	RO/ BOP		
	RO/ BOP		

Enclosure 1

Unit 2 Scram Immediate Actions (0EOP-01-UG)

SCRAM IMMEDIATE ACTIONS

1. **Ensure** SCRAM valves OPEN by manual SCRAM or ARI initiation.

2. **<u>WHEN</u>** steam flow less than 3 x 10^6 lb/hr,
   <u>THEN</u> **place** reactor mode switch in SHUTDOWN.

3. **<u>IF</u>** reactor power below 2% (APRM downscale trip),
   <u>THEN</u> **trip** main turbine.

4. **Ensure** master RPV level controller setpoint at +170 inches.

5. **<u>IF:</u>**

   - Two reactor feed pumps running

     <u>AND</u>

   - RPV level above +160 inches

     <u>AND</u>

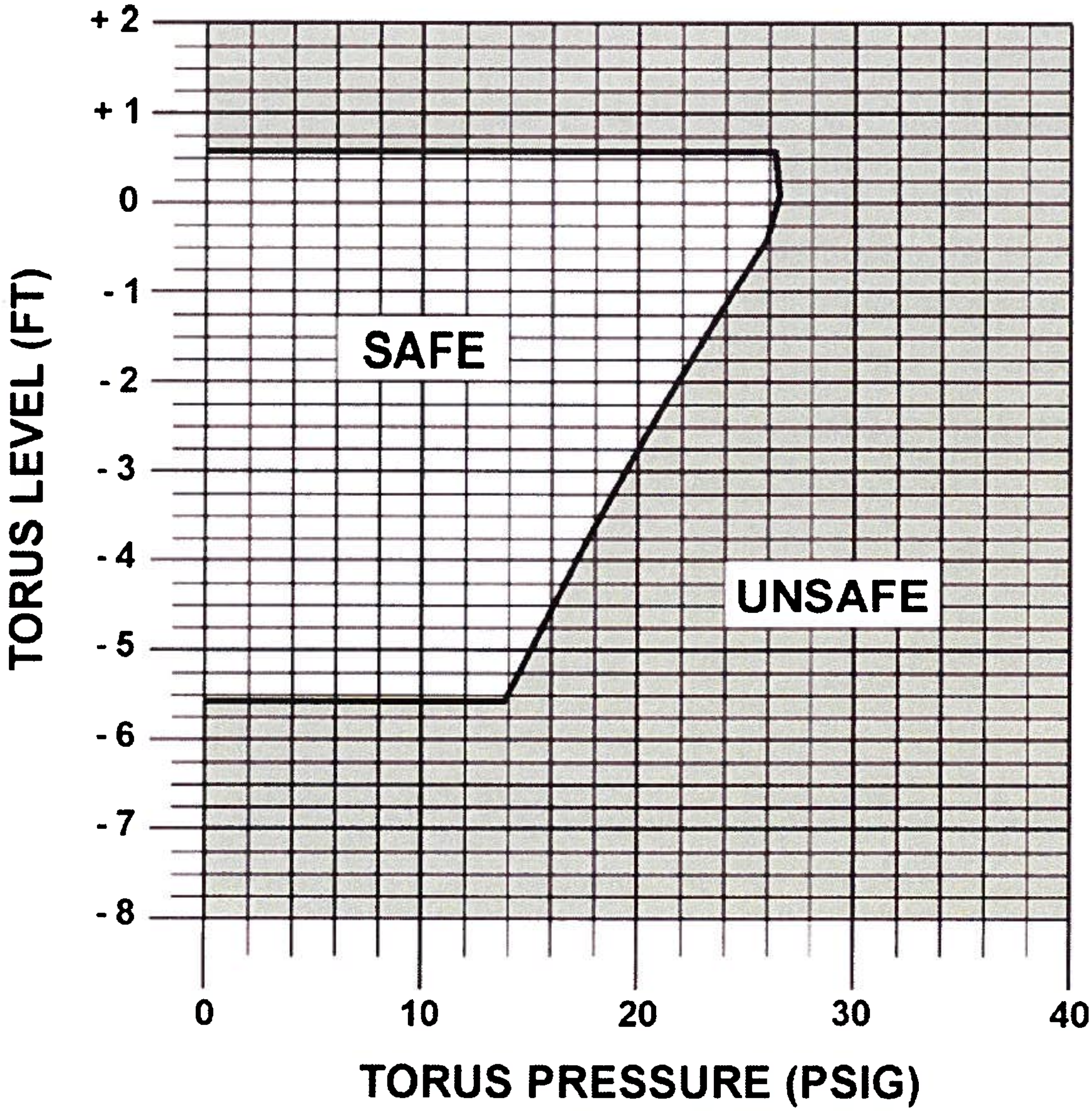
   - RPV level rising,

   <u>THEN</u> **trip** one.

Enclosure 2

ATTACHME
Page

Pressure Suppression Pressure



Enclosure 3

1.0 ENTRY CONDITIONS

- As directed by Emergency Operating Procedures (EOPs)

2.0 INSTRUCTIONS

2.1 <u>Torus Spray</u>

2.1.1 Manpower Required

- 1 Reactor Operator

2.1.2 Special Equipment

None

2.1.3 Torus Spray Actions

1. **Confirm** torus pressure above 2.5 psig. .... ☐ RO

2. <u>IF</u> Loop A RHR will be used,
   <u>THEN:</u>

   a. **Place** E11-CS-S18A (2/3 Core Height LPCI Initiation Override Switch) to MANUAL OVERRD.......... ☐ RO

   b. Momentarily **place** E11-CS-S17A (Containment Spray Valve Control Switch) to MANUAL. .... ☐ RO

   c. **Ensure** one Loop A RHR Pump running.......... ☐ RO

   d. **Ensure** E11-F028A (Torus Discharge Isol Vlv) OPEN.......... ☐ RO

   e. **Open** E11-F027A (Torus Spray Isol Vlv).......... ☐ RO

   f. **Ensure** operation in LPCI, Torus Cooling or Drywell Spray mode .... ☐ RO

Enclosure 3

Page 2 of 2

2.1.3 Torus Spray Actions (continued)

3. <u>IF</u> Loop B RHR will be used,
<u>THEN</u>:

   a. **Place** E11-CS-S18B (2/3 Core Height LPCI Initiation Override Switch) to MANUAL OVERRD ☐ RO

   b. Momentarily **place** E11-CS-S17B (Containment Spray Valve Control Switch) to MANUAL. ☐ RO

   c. **Ensure** one Loop B RHR Pump running ☐ RO

   d. **Ensure** E11-F028B (Torus Discharge Isol Vlv) OPEN ☐ RO

   e. **Open** E11-F027B (Torus Spray Isol Vlv) ☐ RO

   f. **Ensure** operation in LPCI, Torus Cooling **<u>OR</u>** Drywell Spray mode ☐ RO

4. **<u>WHEN</u>** torus pressure drops to 2.5 psig **<u>OR</u>** directed to terminate sprays,
<u>THEN</u> ensure CLOSED:

   - E11-F027A (Torus Spray Isol Vlv) ☐ RO

   - E11-F027B (Torus Spray Isol Vlv) ☐ RO

5. **<u>IF</u>** re-initiation of sprays required,
<u>THEN</u> return to Section 2.1.3 Step 1. ☐ RO

6. **<u>WHEN</u>** sprays **<u>NO</u>** longer required,
<u>THEN</u> go to Section 2.2 ☐ RO

Enclosure 4

Page 1 of 6

2.1.3 Drywell Spray Actions

1. **Ensure** both reactor recirculation pumps tripped. .... ☐ RO

2. IF E-bus load stripping has occurred,
   THEN:

   a. **Confirm** electrical power has been aligned per EOP-01-SBO-14. .... ☐ RO

   b. **Secure** drywell coolers per Attachment 1 and **continue** at Section 2.1.3 Step 2.c. .... ☐ RO

   c. IF RHR Loop A will be used for sprays,
      THEN go to Section 2.1.3 Step 9. .... ☐ RO

   d. IF RHR Loop B will be used for sprays,
      THEN go to Section 2.1.3 Step 10. .... ☐ RO

3. **Place** all drywell cooler control switches to OFF (L/O). .... ☐ RO

Enclosure 4

Page 2 of 6

2.1.3 Drywell Spray Actions (continued)

4. **<u>Unit 1 Only:</u> <u>IF</u>** drywell coolers continue to run,
   <u>THEN:</u>

   - In Panel XU-27, west side, **place** VA-CS-5993 (D/W Clr A&D Override Switch) in STOP. ☐ RO

   - In Panel XU-28, west side, **place** VA-CS-5994 (D/W Clr B&C Override Switch) in STOP. ☐ RO

5. **<u>Unit 2 Only:</u> <u>IF</u>** drywell coolers continue to run,
   <u>THEN:</u>

   - In Panel XU-27, west side, **place** VA-CS-5993 (D/W Clr A&D Override Switch) in STOP. ☐ RO

   - In Panel XU-28, east side, **place** VA-CS-5994 (D/W Clr B&C Override Switch) in STOP. ☐ RO

6. **<u>IF</u>** drywell coolers continue to run,
   <u>THEN</u> secure drywell coolers per Attachment 1 and **continue** at Section 2.1.3 Step 7. ☐ RO

7. **Ensure** SW-V141 (Well Water to Vital Header Vlv) CLOSED. ☐ RO

8. **Ensure** <u>one</u> valve OPEN:

   - SW-V111 (Conv SW To Vital Header Vlv) ☐ RO

   - SW-V117 (Nuc SW To Vital Header Vlv) ☐ RO

Enclosure 4

2.1.3 Drywell Spray Actions (continued)

9. **IF** Loop A RHR will be used for drywell spray,
THEN:

NOTE
E11-F017A will remain OPEN for five minutes following a LOCA signal. .... ☐

a. **IF** E11-F015A (Inboard Injection Vlv) OPEN,
THEN close E11-F017A (Outboard Injection Vlv). .... ☐
RO

b. **Place** E11-CS-S18A (2/3 Core Height LPCI Initiation Override Switch) to MANUAL OVERRD. .... ☐
RO

c. Momentarily **place** E11-CS-S17A (Containment Spray Valve Control Switch) to MANUAL. .... ☐
RO

d. **Ensure** E11-F024A (Torus Cooling Isol Vlv) CLOSED. .... ☐
RO

e. **Ensure** one Loop A RHR Pump running. .... ☐
RO

f. **Confirm** requirements for Drywell Spray Initiation met:

- Safe region of Drywell Spray Initiation Limit .... ☐
RO

- Torus level below +21 inches .... ☐
RO

g. **Open** E11-F021A (Drywell Spray Inbd Isol Vlv). .... ☐
RO

h. **Throttle open** E11-F016A (Drywell Spray Otbd Isol Vlv) to obtain between 8,000 gpm and 10,000 gpm flow. .... ☐
RO

i. **IF** E-bus load stripping has occurred,
THEN go to Section 2.1.3 Step 11. .... ☐
RO

Page 4 of 6

2.1.3 Drywell Spray Actions (continued)

j. **IF** additional flow required,
THEN start the other RHR pump and limit flow to less than or equal to 11,500 gpm. .... ☐ RO

k. **Ensure** RHRSW Loop A operating:

(1) **Place** E11-S19A (RHR SW Booster Pumps A & C LOCA Override Switch) in MANUAL OVERRD. .... ☐ RO

(2) **Align** RHRSW to the heat exchanger (OP-43). .... ☐ RO

l. **Establish** RHR flow through the heat exchanger:

(1) **Ensure** E11-F047A (Hx A Inlet Vlv) OPEN. .... ☐ RO

(2) **Ensure** E11-F003A (Hx A Outlet Vlv) OPEN. .... ☐ RO

NOTE

E11-F048A will remain OPEN for three minutes following a LOCA signal. .... ☐

(3) **Close** E11-F048A (Hx A Bypass Vlv). .... ☐ RO

10. **IF** Loop B RHR will be used for drywell spray,
THEN:

NOTE

E11-F017B will remain OPEN for five minutes following a LOCA signal. .... ☐

a. **IF** E11-F015B (Inboard Injection Vlv) OPEN,
THEN close E11-F017B (Outboard Injection Vlv). .... ☐ RO

b. **Place** E11-CS-S18B (2/3 Core Height LPCI Initiation Override Switch) to MANUAL OVERRD. .... ☐ RO

Enclosure 4

2.1.3 Drywell Spray Actions (continued)

c. Momentarily place E11-CS-S17B (Containment Spray Valve Control Switch) to MANUAL. .... ☐ RO

d. Ensure E11-F024B (Torus Cooling Isol Vlv) CLOSED. .... ☐ RO

e. Ensure one Loop B RHR Pump running. .... ☐ RO

f. Confirm requirements for Drywell Spray Initiation are met:

- Safe region of the Drywell Spray Initiation Limit .... ☐ RO
- Torus level below +21 inches .... ☐ RO

g. Open E11-F021B (Drywell Spray Inbd Isol Vlv). .... ☐ RO

h. Throttle open E11-F016B (Drywell Spray Otbd Isol Vlv) to obtain between 8,000 gpm and 10,000 gpm flow. .... ☐ RO

i. **IF** E-bus load stripping has occurred,
THEN go to Section 2.1.3 Step 11. .... ☐ RO

j. **IF** additional flow required,
THEN start the other RHR pump and limit flow to less than or equal to 11,500 gpm. .... ☐ RO

k. Ensure RHRSW Loop B operating:

(1) Place E11-S19B (RHR SW Booster Pumps B & D LOCA Override Switch) in MANUAL OVERRD. .... ☐ RO

(2) Align RHRSW to the heat exchanger (OP-43). .... ☐ RO

Enclosure 4

Page 6 of 6

2.1.3 Drywell Spray Actions (continued)

l. **Establish** RHR flow through the heat exchanger:

(1) **Ensure** E11-F047B (Hx B Inlet Vlv) OPEN. ....☐ RO

(2) **Ensure** E11-F003B (Hx B Outlet Vlv) OPEN............................☐ RO

NOTE
E11-F048B will remain OPEN for three minutes following a LOCA signal. ....☐

(3) **Close** E11-F048B (Hx B Bypass Vlv). ....☐ RO

11. **WHEN** drywell pressure drops to 2.5 psig **OR** directed to terminate drywell spray,
THEN ensure CLOSED:

a. E11-F016A (Drywell Spray Otbd Isol Vlv) ....☐ RO

b. E11-F021A(Drywell Spray Inbd Isol Vlv). ....☐ RO

c. E11-F016B (Drywell Spray Otbd Isol Vlv) ....☐ RO

d. E11-F021B (Drywell Spray Inbd Isol Vlv). ....☐ RO

12. **Ensure** either:

- RHR operated in LPCI mode ....☐ RO
- RHR operated in Torus Cooling. ....☐ RO
- RHR pumps are secured ....☐ RO

13. **IF** re-initiation of drywell spray required,
THEN return to Section 2.1.3 Step 9. ....☐ RO

ATTACHMENT 1 - Scenario Quantitative Attribute Assessment

Category	NUREG 1021 Rev. 2 Supp. 1 Req.	Scenario Content
Total Malfunctions	5-8	8
Malfunctions after EOP Entry	1-2	2
Abnormal Events	2-4	4
Major Transients	1-2	2
EOPs Used	1-2	2
EOP Contingency	0-2	2
Run Time	60-90 min	90
Crew Critical Tasks	2-3	2
Tech Specs	2	2
Instrument / Component Failures before Major	2 – OATC 2 - BOP	4
Instrument / Component Failures after Major	2	2
Normal Operations	1	1
Reactivity manipulation	1	1

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ATTACHMENT 2 – Shift Turnover

Brunswick Unit 2 Plant Status					
Station Duty Manager:				Workweek Manager:	
Mode:	1	Rx Power:	80%	Gross*/Net MWe*:	840 / 800
Plant Risk: Current EOOS Risk Assessment is:				Green	
SFP Time to 200 Deg F:	128.7 hrs			Days Online:	142 days
Turnover:	Feedwater Temperature Reduction will be implemented this weekend. Evolutions this shift: Perform 0PT-40.2.11, Main Generator Voltage Regulator Manual and Automatic Operational Check Raise power to 100% IAW GP-12.				
Protected Equipment:	RHR 2A and 2B, Core Spray 2B				
Comments:	2C TCC Pump is in service on Unit One. 2A Core Spray is under clearance for oil leak from last shift, declared inoperable at 0500, today.				

Facility: Brunswick   Scenario No.: NRC 3   Op-Test No.: FINAL

Examiners: ______   Operators: SRO
RO
BOP

Initial Conditions: The plant is operating at 95% power with APRM 2 failed downscale and bypassed. Plan for the day is to swap Condensate Pumps.

Turnover: Feedwater Temperature Reduction will be implemented this weekend.

Event No.	Malf. No.	Event Type*	Event Description
1	NA	N	Swap Condensate Pumps
2	RD001M (26-11)	C-RO C-SRO	Rod Drift – Tech Spec –**AOP-2.0**
3		R	Lower power for Thermal Limit verification
4	CF039F	C-BOP C-SRO	Heater Drain Level Controller Failure – **AOP-23.0**
5	ES022F	C-RO C-SRO	Inadvertent RCIC Initiation – Tech Spec – **AOP-03.0**
6	RP003F	C-BOP C-SRO	A RPS MG Set Trip
7	ES048F	M	HPCI Unisolable Steam Leak – **AOP-5.0**
8	RP005F RP006F	M C	Scram - Auto and Manual Scam failure - ARI
9		C	RHR Room Coolers Trip
10		M	ED
11		C	2 ADS Valves Fail to Open

* (N)ormal, (R)eactivity, (I)nstrument, (C)omponent, (M)ajor

SCENARIO DESCRIPTION SUMMARY – 2015 NRC Scenario 3

Event	Description
1	Crew will swap Condensate pumps for maintenance.
2	Control Rod 26-11 will start to drift in. The crew will enter 0AOP-02.0 and take action IAW 2APP-A-05 (3-2). When the high temperature alarm is received Engineering will report that scram times cannot be assured based on past history of the control rod. Determine TS 3.1.3 condition C1 to insert the control rod in 3 hours and C2 to disarm the control rod within 4 hours.
3	Reactor Engineering will request power lowered to 80% via Recirculation flow until thermal limits can be checked and for rod recovery.
4	The Heater Drain level controller will fail resulting in full opening of the pump discharge valves and lowering tank level. The crew will respond per AOP-23.0, reduce power per ENP-24.5, trip one Heater Drain Pump and open HD-V57 to control Deaerator level.
5	An inadvertent RCIC initiation will require the crew to respond IAW AOP-03.0 and trip RCIC.
6	RPS MG Set will trip requiring the crew to swap to alternate PRS power supply.
7	A HPCI steam line break will occur. The crew will enter AOP-05 and EOP-03-SCCP. HPCI will fail to isolate. The reactor will be manually scrammed when HPCI area exceeds its Maximum Safe Operating Temperature. Secondary Containment area temperatures will continue to rise. Multiple areas will exceed their Maximum Safe Operating Temperatures requiring emergency depressurization.
8	Reactor Scram
9	RHR Room Cooler Fans Trip
10	2 Areas will exceed Max Safe requiring the Crew to ED
11	2 ADS Valves will fail to come open on ED requiring an additional 2 SRV's be open

CREW CRITICAL TASKS

Description
Insert a reactor scram when HPCI reaches its Max Safe Operating Value. (HPCI Area Temperature Max normal is equal to Max Safe.)
Perform Emergency Depressurization when more than one area exceeds the same Max Safe Operating Value or EQ Envelope for the same parameter.

BRUNSWICK TRAINING SECTION

OPERATIONS TRAINING

INITIAL LICENSED OPERATOR

SIMULATOR EVALUATION GUIDE

2015 NRC SCENARIO 3

ROD DRIFT, HDD CONTROLLER FAILURE, INADVERTENT RCIC INITIATION, LOSS OF RPS, HPCI UN-ISOLABLE STEAM LEAK, SCCP, ED

REVISION 0

Developer: Lou Sosler	**Date:** 9/11/2015
Technical Review: John Biggs	**Date:** 9/23/2015
Validator: Brian Moschet	**Date:** 9/11/2015
Validator: Kevin Kingston	**Date:** 9/11/2015
Facility Representative: Jerry Pierce	**Date:** 9/23/2015

REVISION SUMMARY	
0	Exam scenario for 2015 NRC Exam.

TABLE OF CONTENTS

1.0 SCENARIO OUTLINE

Event	Malf. No.	Type*	Event Description
1	NA	N	Swap Condensate Pumps
2	RD001M (26-11)	C-RO C-SRO	Rod Drift – Tech Spec
3		R	Lower power for Thermal Limit verification
4	CF039F	C-BOP C-SRO	Heater Drain Level Controller Failure – AOP-23.0
5	ES022F	C-RO C-SRO	Inadvertent RCIC Initiation – Tech Spec – AOP-03.0
6	RP003F	C-BOP C-SRO	A RPS MG Set Trip
7	ES048F	M	HPCI Unisolable Steam Leak – AOP-5.0
8	RP005F RP006F	M C	Scram - Auto and Manual Scam failure - ARI
9		C	RHR Room Coolers Trip
10		M	ED
11		C	2 ADS Valves Fail to Open
*(N)ormal, (R)eactivity, (C)omponent or Instrument, (M)ajor			

2.0 SCENARIO DESCRIPTION SUMMARY

Event	Description
1	Crew will swap Condensate pumps for maintenance.
2	Control Rod 26-11 will start to drift in. The crew will enter 0AOP-02.0 and take action IAW 2APP-A-05 (3-2). When the high temperature alarm is received Engineering will report that scram times cannot be assured based on past history of the control rod. Determine TS 3.1.3 condition C1 to insert the control rod in 3 hours and C2 to disarm the control rod within 4 hours.
3	Reactor Engineering will request power lowered to 80% via Recirculation flow until thermal limits can be checked and for rod recovery.
4	The Heater Drain level controller will fail resulting in full opening of the pump discharge valves and lowering tank level. The crew will respond per AOP-23.0, reduce power per ENP-24.5, trip one Heater Drain Pump and open HD-V57 to control Deaerator level.
5	An inadvertent RCIC initiation will require the crew to respond IAW AOP-03.0 and trip RCIC.
6	RPS MG Set will trip requiring the crew to swap to alternate PRS power supply.
7	A HPCI steam line break will occur. The crew will enter AOP-05 and EOP-03-SCCP. HPCI will fail to isolate. The reactor will be manually scrammed when HPCI area exceeds its Maximum Safe Operating Temperature. Secondary Containment area temperatures will continue to rise. Multiple areas will exceed their Maximum Safe Operating Temperatures requiring emergency depressurization.
8	Reactor Scram
9	RHR Room Cooler Fans Trip
10	2 Areas will exceed Max Safe requiring the Crew to ED
11	2 ADS Valves will fail to come open on ED requiring an additional 2 SRV's be open

3.0 CREW CRITICAL TASKS

Description
Insert a reactor scram when HPCI reaches its Max Safe Operating Value. (HPCI Area Temperature Max normal is equal to Max Safe.)
Perform Emergency Depressurization when more than one area exceeds the same Max Safe Operating Value or EQ Envelope for the same parameter.

4.0 TERMINATION CRITERIA

When the Reactor is depressurized and level being restored to normal level band, the scenario may be terminated.

5.0 IMPLEMENTING REFERENCES

NOTE: Refer to the most current revision of each Implementing Reference.

Number	Title
2OP-32.0	CONDENSATE AND FEEDWATER SYSTEM OPERATING PROCEDURE
A5 (3-2)	ROD DRIFT
0AOP-02.0	CONTROL ROD MALFUNCTION/MISPOSITION
A-03 (1-10)	SAFETY/RELIEF VALVE OPEN
A-03 (1-1)	SAFETY OR DEPRESS VALVE LEAKING
UA-12 (5-4)	SPTMOS DIV I
UA-12 (5-5)	SPTMOS DIV II
A-2 (3-2)	RHR PUMP 2A SEAL CLR FLOW LOW
A-2 (4-2)	RHR PUMP 2C SEAL CLR FLOW LOW
A-3 (2-1)	CS OR RHR PUMPS RUNNING
UA-3 (2-7)	AREA RAD REACTOR BUILDING HIGH
A-2 (5-7)	STM LEAK DET AMBIENT TEMP HIGH
A-1 (4-1)	HPCI TURB TRIP SOL ENER
A-1 (3-5)	HPCI ISOL TRIP SIG A INITIATED
A-1 (4-5)	HPCI ISOL TRIP SIG B INITIATED

6.0 SETUP INSTRUCTIONS

1. **PERFORM** TAP-409, Miscellaneous Simulator Training Guidelines, Attachment 5, Checklist for Simulator Exam Security.
2. **RESET** the Simulator to IC-25.
3. **ENSURE** the RWM is set up as required for the selected IC.
4. **ENSURE** appropriate keys have blanks in switches.
5. **RESET** alarms on SJAE, MSL, and RWM NUMACs.
6. **ENSURE** no rods are bypassed in the RWM.
7. **PLACE** all SPDS displays to the Critical Plant Variable display (#100).
8. **ENSURE** hard cards and flow charts are cleaned up
9. **TAKE** the SIMULATOR OUT OF FREEZE
10. **ALIGN** the plant as follows:

Manipulation
Ensure 2C TCC pump is in service on Unit One. Loaded in Scenario File.

11. **LOAD** Scenario File.
12. **IF desired**, take a **SNAPSHOT** and save into an available IC for later use.
13. **PLACE** a clearance on the following equipment.

Component	Position
Bypass APRM 2 (Blue Tag)	Bypassed

14. **INSTALL** Protected Equipment signage and **UPDATE** RTGB placard as follows:

15. **ENSURE** each Implementing References listed in Section 7 is intact and free of marks.

16. **ENSURE** all materials in the table below are in place and marked-up to the step identified.

Required Materials
None

17. **ENSURE** Station Duty Manager and Work Week Manager names are filled in on the Shift Turnover Sheet.

18. **ADVANCE** the recorders to prevent examinees from seeing relevant scenario details.

19. **PROVIDE** Shift Briefing sheet for the CRS.

20. **VERIFY** all actions contained in TAP-409, Miscellaneous Simulator Training Guidelines, Attachment 4, Simulator Training Instructor Checklist, are complete.

7.0 INTERVENTIONS

TRIGGERS

Trig	Type	ID
1	Malfunction	RD001M - [CONTROL ROD SLOW INSERTION DRIFT]
2	Malfunction	CF039F - [HTR DRN DEAER LVL CNTRLR FAILURE]
3	Malfunction	ES022F - [RCIC INADVERTANT START]
4	Malfunction	RP003F - [RPS M.G. SET TRIP]
5	DI Override	K5624A - [RHR PMP ROOM VENT FAN B]
5	DI Override	K1507A - [AUTO DEPRESS VLV B21-F013C]
5	DI Override	K1511A - [AUTO DEPRESS VLV B21-F013A]
5	DI Override	K5624A - [RHR PMP ROOM VENT FAN B]
5	DI Override	K1507A - [AUTO DEPRESS VLV B21-F013C]
5	DI Override	K5624A - [RHR PMP ROOM VENT FAN B]
5	DI Override	K5623A - [RHR PMP RM VENT FAN A]
5	DI Override	K1511A - [AUTO DEPRESS VLV B21-F013A]
5	DI Override	K5623A - [RHR PMP RM VENT FAN A]
5	DI Override	K5623A - [RHR PMP RM VENT FAN A]
5	Malfunction	ES047F - [HPCI STM BRK HPCI ROOM]
8	Remote Function	HP_ZVMS402T - [E41-F002 INBD STM VLV]
9	Malfunction	HP001F - [BYP TO CONDS STG * VLV E41-F008]
10	Trigger Command	mfd:rd001m,26-11
11	Annunciator	ZA512 - [CRD HYD TEMP HIGH]

Trig #	Trigger Text
8	Q1116LG1 - [STM LINE VLV E41-F002 GREEN]
9	Q1117LG1 - [STM LINE VLV E41-F003 GREEN]
10	Q2BVNUGD - [FULL IN-ROD DISPLAY]
11	Q2BVNUGD - [FULL IN-ROD DISPLAY]

MALFUNCTIONS

Malf ID	Mult ID	Description	Current Value	Target Value	Rmp time	Actime	Dactime	Trig
RP005F		AUTO SCRAM DEFEAT	True	True				
RP006F		MANUAL SCRAM DEFEAT	True	True				
ES053F		E41-F002 FAILURE TO AUTO CLOSE	True	True				
NI032F	APRM2	APRM FAILS LO	True	True				
RD001M	26-11	ROD DRIFT	FALSE	TRUE				1
CF039F		HTR DRAIN CONTROL FAIL	FALSE	TRUE				2
ES022F		RCIC INADVERTENT START	FALSE	TRUE				3
RP003F		RPS MG SET TRIP	FALSE	TRUE				4
ES047F		HPCI STEAM LINE BREAK	0.00	15.00	20:00			5
HP001F	E41-F003	STEAM SUPPLY LINE VLV E41-F003	FALSE	TRUE				9

REMOTES

Remf Id	Mult Id	Description	Current Value	Target Value	Rmp time	Actime	Trig
CC_IACW4518		2C TBCCW PUMP UNIT 1 ALIGNMENT	1	1			
HP_ZVMS402T		E41-F002 INBD STM VLV	ON	OFF			8

PANEL OVERRIDES

Tag ID	Description	Position / Target	Actual Value	Override Value	Rmp time	Actime	Dactime	Trig
K5623A	RHR PMP RM VENT FAN A	AUTO	ON	OFF		00:08:00		5
K5623A	RHR PMP RM VENT FAN A	OFF	OFF	ON		00:08:00		5
K5623A	RHR PMP RM VENT FAN A	ON	OFF	OFF		00:08:00		5
K5624A	RHR PMP ROOM VENT FAN B	AUTO	ON	OFF		00:04:00		5
K5624A	RHR PMP ROOM VENT FAN B	OFF	OFF	ON		00:04:00		5
K5624A	RHR PMP ROOM VENT FAN B	ON	OFF	OFF		00:04:00		5
K1507A	AUTO DEPRESS VLV B21-F013C	AUTO	ON	ON				5
K1507A	AUTO DEPRESS VLV B21-F013C	OPEN	OFF	OFF				5
K1511A	AUTO DEPRESS VLV B21-F013A	AUTO	ON	ON				5
K1511A	AUTO DEPRESS VLV B21-F013A	OPEN	OFF	OFF				5

ANNUCIATOR OVERRIDES

Window	Description	Tagname	Override Type	OVal	AVal	Actime	Dactime	Trig
1-2	CRD HYD TEMP HIGH	ZA512	ON	ON	OFF			11

8.0 OPERATOR RESPONSE AND INSTRUCTIONAL STRATEGIES

EVENT 1: SHIFT TURNOVER / SWAP RUNNING CONDENSATE PUMPS	
	Simulator Operator Actions
	Ensure Monitored Parameters is open and Scenario Based Testing Variables are loaded.

	Simulator Operator Role Play
	When requested modify RF to transfer Unit Trip and LOCA Load Shed Switches for selected pumps: EE_LSHED3 to DISABLE EE_LSHED2 to ENABLE EE_VTSHED3 to DISABLE EE_VTSHED2 to ENABLE
	Acknowledge requests to Radwaste.
	Prestart checks complete on 2C Condensate Pump.
	2D Bus clear.
	If asked to check pump, 2C Condensate Pump running fine.

Evaluator Notes	
Plant Response:	None
Objectives:	Transfer Running Condensate Pumps
Success Path:	Condensate Pumps are swapped.
Event Termination:	Go to Event 2 at the discretion of the Lead Evaluator.

EVENT 1: SHIFT TURNOVER / SWAP RUNNING CONDENSATE PUMPS

Time	Pos	EXPECTED Operator Response	Comments
	SRO	Conduct shift turnover shift briefing.	
	SRO	Direct RO to swap condensate pumps.	
	OATC	Monitors the plant	
	BOP	Swap Condensate Pumps IAW 2OP-32, Section 8.5	
		2OP-32, Section 8.5.	Enclosure 1

EVENT 2: ROD DRIFT	
Simulator Operator Actions	
	At the direction of the Lead Evaluator, Initiate Trigger 1 to drift CR 26-11 into the core.
	When the control rod is inserted to 00, verify that CRD High Temperature alarm comes in.
	If control rod is scrammed, delete the rod drift malfunction.
	Two minutes after control rod is disarmed or scrammed, delete CRD HYD TEMP HIGH alarm.

Simulator Operator Role Play	
	If contacted as the RE to address thermal limits, inform crew that you will monitor core performance on the computer.
	If asked as the RBAO to investigate HCU for control 26-11, report that the HCU scram outlet riser is hot to the touch.
	When contacted as the RBAO and after high temperature alarm has been actuated, report that the CRD temperature is 390°F
	When contacted as the System Engineer report that based on past history of this rod (26-11) scram times cannot be guaranteed.
	If asked as the RBAO to disarm control rod, coordinate with Sim Operator after 5 minutes.
	As RE, request power lowered to 80% via Recirculation flow until thermal limits can be checked and for rod recovery directions.
	If requested, close 113 valve, the reopen. (Charging Header Isolation Valve)
	Report Accumulator pressure 980#

Evaluator Notes	
Plant Response:	Control Rod 26-11 will drift full in. Crew should enter AOP-02.0 and take action IAW 2APP-A-05 (3-2). When the high temperature alarm is received, Engineering will report that scram times cannot be assured based on past history of the control rod. Determine TS 3.1.3 condition C1 to insert the control rod in 3 hours <u>and</u> C2 to disarm the control rod within 4 hours.
Objectives:	SRO - Direct actions in response to a drifting control rod and evaluate Tech Specs. RO - Respond to a drifting control rod.
Success Path:	The drifting control rod is fully inserted, determined that the control rod must be

	placed under clearance and electrically disarmed.
Event Termination:	Go to Event 3 at the direction of the Lead Evaluator.

EVENT 2: ROD DRIFT

Time	Pos	EXPECTED Operator Response	Comments
	SRO	Direct actions of 2APP-A-05 (3-2) *ROD DRIFT*	
	SRO	Direct entry into 0AOP-02.0, Control Rod Malfunction/Misposition.	
	SRO	According to Note 2 in TS Table 3.1.4-1 the rod must be declared inoperable. Tech Spec 3.1.3 Control Rod Operability Condition C. One or more control rods inoperable for reasons other than Condition A or B <u>Required Action</u> C.1 Fully insert inoperable control rod (3 hrs) C.2 Disarm the associated CRD (4 hrs)	
	SRO	Contact System Engineer on high temperature condition of control rod.	
	SRO	May direct the control rod to be scrammed to attempt to reseat the leaking outlet valve.	
	BOP	Plant monitoring	

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EVENT 2: ROD DRIFT			
	OATC	Acknowledge alarms: A-05 (2-2) ROD OUT BLOCK A-05 (3-2) Rod Drift	
	OATC	Perform the actions of APP-A-05 (3-2) ROD DRIFT as follows: ☐ Determine which control rod is drifting. ☐ Select the drifting control rod and determine direction of drift. ☐ Attempt to arrest the drift by giving a withdraw signal. ☐ If rod continues to drift in, apply an RMCS insert signal and fully insert to position 00. ☐ Attempt to locate and correct the cause of the rod malfunction as follows: ☐ Check and adjust cooling water header pressure if required. ☐ Direct AO to check for leaking scram valve.	
	OATC	Monitor core parameters, main steam line radiation and off-gas activity.	

EVENT 3: POWER REDUCTION	
Simulator Operator Actions	

Simulator Operator Role Play	
	If contacted as the NE for power reduction guidance, inform crew the reactivity plan has power reduced to ~80% (~56 Mlbms) using recirc flow.
	If contacted as the NE to monitor power reduction, inform crew that you will monitor core performance on the computer.
	If contacted as Radwaste operator acknowledge any requests.
	If contacted as the Load Dispatcher, acknowledge report that Brunswick Unit Two will be lowering power.

Evaluator Notes	
Plant Response:	Reactor power will be reduced IAW 0ENP-024.5
Objectives:	SRO - Direct actions power reduction RO – Reduce power as directed by the SRO BOP – Control balance of plant
Success Path:	Reduce power IAW 0ENP-24.5
Event Termination:	Go to Event 4 at the direction of the Lead Evaluator.

EVENT 3: POWER REDUCTION

Time	Pos	EXPECTED Operator Response	Comments
	SRO	Directs power to be reduced using recirculation flow IAW 0ENP-24.5	
	BOP	Monitors the plant	
	RO	May reference 2OP-02 section 7.1	
	RO	Request peer checker / reactivity team.	
	RO	Reduces power using recirculation flow to ~80% power. Reduce RR Pump speed by depressing the Master Lower fast or Lower medium pushbutton Continues lowering Recirc pump reductions until ~80% power.	
	RO	Verifies operation on the Power to Flow Map	

EVENT 4: HDD CONTROLLER FAILURE	
Simulator Operator Actions	
	At the direction of the Lead Evaluator, **Initiate Trigger 2** to fail Heater Drain Controller.
	If directed to place controller in Manual or to swap master controllers, Delete CF039F.

Simulator Operator Role Play	
	If contacted as TBAO to investigate, report LC-91 is in master and is sending a full open signal.
	If asked by I&C to investigate controller failure, acknowledge the request.
	When HDD level is stabilized and if directed to place controller in Manual or to swap master controllers, have Sim Operator delete CF039F and report controller in manual maintaining level

Evaluator Notes	
Plant Response:	Heater Drain Tank Lowers Low level alarm at 32" Both Heater Drain pumps trip at 24" Condensate Booster Pump C auto start if power is not sufficiently reduced
Objectives:	Enter 0AOP-23.0 Reduce power Control level using HD-57
Success Path:	Reduce power Manually control level in Heater Drain Tank using the HD-57
Event Termination:	Go to Event 5 at the discretion of the Lead Evaluator.

EVENT 4: HDD CONTROLLER FAILURE			
Time	**Pos**	**EXPECTED Operator Response**	**Comments**
	SRO	Direct entry in 0AOP-23.0, Condensate/Feedwater System Failure	
	SRO	Directs power reduction to stabilize Condensate/Feedwater	
	SRO	Directs manual control with HD-57 to stabilize HD Tank level	
	SRO	Directs I&C to investigate	
	SRO	May contact Shift Manager	
	OATC	Monitor plant	
	OATC	Announce entry into AOP-23.0	
	OATC	Reduce Reactor power IAW 0ENP-24.5 as directed by SRO	May initiate a manual runback using the pushbutton.

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EVENT 4: HDD CONTROLLER FAILURE

Time	Pos	EXPECTED Operator Response	Comments
	BOP	Acknowledge and report alarm: UA-4 2-10 HD DEAERATOR LEVEL HIGH-LOW.	Alarm at 30 inches and lowering. Pump trip at 24 inches and lowering.
		Diagnose HD Pump discharge valves full open	
		Enter and announce 0AOP-23.0	
		Trips one of the operating Heater Drain pump	
		Maintains heater drain deaerator level less than 60 inches indicated on HEATER DRAIN DEAERATOR LEVEL, HD-LI-97	If level reaches 60 inches UA-4, 3-10 may alarm and the HDD Moisture removal valves will open. Move to the next event when level is being controlled with the HD-V57.
		May dispatch TBAO to check HD Pump Air-Operated Discharge Level Control Valves, HD-LV-91-1, 2, & 3.	
		May direct TBAO to place HDD level control in Manual IAW 2OP-35 Section 6.3.8. or swap controller IAW 2OP-35, Section 6.3.8	
		Monitors main condenser vacuum and condensate parameters	
		May have to secure a CBP if one auto started during the evolution.	

EVENT 5: INADVERTENT RCIC INITIATION	
	Simulator Operator Actions
	At the direction of the Lead Evaluator, **Initiate Trigger 3** to activate the an inadvertent RCIC Initiation.

	Simulator Operator Role Play
	If RCIC has been running for >5 minutes, and crew has not recognized RCIC running, call control room as AO and ask why RCIC is running.
	If asked as I&C to investigate, acknowledge the request.
	If asked as RE to monitor thermal limits, acknowledge the request.
	If asked as chemistry for Rx Coolant Sample, acknowledge the request.

Evaluator Notes	
Plant Response:	RCIC will inadvertently initiate. The crew should respond per 0AOP-03.0, Positive Reactivity Addition and trip RCIC. RCIC should be declared inoperable per TS 3.5.3.
Objectives:	SRO - Direct actions IAW 0AOP-03.0 RO - Take actions IAW 0AOP-03.0 BOP – Monitors reactor plant parameters
Success Path:	RCIC is shutdown and Tech Spec. 3.5.3 is addressed.
Event Termination:	Go to Event 6 at the discretion of the Lead Evaluator.

EVENT 5: RCIC INADVERTENT INITIATION

Time	Pos	EXPECTED Operator Response	Comments
	SRO	Direct entry into 2AOP-03.0, Positive Reactivity Addition	
		Direct / concur RCIC operation to be terminated.	
		Enter PCCP when torus temp reaches 95° F.	
		Contact I&C to investigate RCIC logic.	
		Tech Spec 3.5.3 RCIC System Determine Condition A applies Required Action A.1, Immediately verify HPCI is OPERABLE AND Required Action A.2, Restore RCIC to OPERABLE within 14 days.	
	RO	Recognize and report RCIC injection	
		Enter and announce 2AOP-03.0	
		Verify inadvertent initiation by two independent indications and trip RCIC.	HPCI & RCIC auto start on LL2 - RPV Water Level Lo Lo (105 inches) – should have automatically scrammed prior to this level
		Depress TURBINE TRIP, E51-S17, push button to trip the RCIC turbine	
		Monitors reactor power.	

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EVENT 5: RCIC INADVERTENT INITIATION

Time	Pos	EXPECTED Operator Response	Comments
		A-03 3-5, RCIC TURBINE STM LINE DRN POT LEVEL HI will annunciate requiring the operator to perform the following if it has been in for 5 minutes: Close TURBINE TRIP & THROTTLE VLV, E51-V8, motor operator.	
	BOP	Monitor plant parameters	

EVENT 6: RPS MG SET A TRIP	
Simulator Operator Actions	
	When directed by the Lead Evaluator, **Initiate Trigger 4** to trip RPS MG Set A
	If directed to isolate RWCU filter demins modify the following Remote Functions: RW_IAFLTFVA and RW_IAFLTFVB, MANUAL (controller mode) and RS_IAFLTFVD, ZERO (valve demand) for Filter A and Filter B

Simulator Operator Role Play	
	If asked as TBAO to investigate, report tripped breaker on MCC 2CA to RPS MG Set tripped and MG Set A Motor is abnormally hot
	If asked as I&C to investigate, acknowledge the request
	If requested to report status of MSIV coil lights in back panel report Inboard DC and Outboard AC lights lit, Inboard AC and Outboard DC out (prior to transferring RPS and resetting PCIS), after transfer and reset, all logic lights lit
	If requested as E&RC, report RWCU sample lines in service
	If requested to vent RWCU seal cooling loops, report action complete

Evaluator Notes

Plant Response: RBS Bus A will deenergize and the following plant response:

a. Half scram and half MSIV Group 1
b. Rx Bldg HVAC isolates and SBGT starts
c. CREV initiates
d. Closure of inboard isolation valves for Group 1 (steam line drains and sample valves), Group 2, Group 3, Group 6 (full CAM)

Objectives:

SRO – Direct actions for loss of RPS A

RO – Identify and report loss of RPS A and components

BOP – Verify plant response for loss of RPS A.

Success Path:	Verify actions for loss of RPS A. Restore power to RPS A and place effected systems back in service.
Event Termination:	Go to Event 7 at the direction of the Lead Evaluator.

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EVENT 6: RPS MG SET TRIP			
Time	**Pos**	**EXPECTED Operator Response**	**Comments**
	SRO	Diagnose loss of RPS MG Set A	
		Contact I&C to investigate	
		Direct RPS A transferred to alternate per 2OP-03 (Reactor Protection System Operating Procedure).	
		Direct Reactor Building HVAC started per 2OP-37.1 (Reactor Building Heating and Ventilation System Operating Procedure).	
		Direct placing RWCU in service per 2OP-14 (Reactor Water Cleanup System Operating Procedure)	
	BOP	Dispatch TBAO to investigate	
		Start Reactor Building HVAC per OP-37.1 (Reactor Building Heating and Ventilation System Operating Procedure).	
		Shutdown SBGT per 2OP-10 (Standby Gas Treatment System Operating Procedure).	
		Place RWCU in service	
	RO	Diagnose loss of RPS MG Set A	
		Transfer RPS A to alternate per 2OP-03, Reactor Protection System Operating Procedure	Restoration actions specified in 2OP-03.

EVENT 7: HPCI STEAM LEAK	
Simulator Operator Actions	
	At the discretion of the Lead Evaluator, **Initiate Trigger 5** to a HPCI Steam Leak
	Verify RHR Room Cooler B trips 4 minutes after HPCI Steam leak is activated, and RHR Room Cooler A trips 8 minutes after steam leak.

Simulator Operator Role Play	
	As **Unit 1 RO/SRO** report multiple Unit 2 fire alarms after ARM alarm
	If directed as OS AO to close PIV-33, wait 4 minutes and report PIV closed.
	If directed as Unit 1 SRO to perform PEP-3.4.7, acknowledge request.
	If contacted as Maintenance or I&C, acknowledge request.

Evaluator Notes	
Plant Response:	Rx Bldg temperatures and rad levels rise. Rx Bldg negative pressure is lost. 2A RHR Room Cooler starts at 120°F in the HPCI room and 2B RHR Room Cooler starts at 145°F in the HPCI room. The 2B RHR Room Cooler will stop after 4 minutes, and 2A RHR Room Cooler will stop after 8 minutes..
Objectives:	SRO – Respond IAW 0AOP-5.0 and 0EOP-03-SCCP BOP – Respond to Reactor Building radiation alarms RO – Diagnose and report HPCI steam leak – attempt to isolate
Success Path:	Scram Reactor and Emergency Depressurize to slow leak.

EVENT 7: HPCI STEAM LEAK

Time	Pos	EXPECTED Operator Response	Comments
	SRO	Enter and direct activities of AOP-05.0	
		Direct HPCI isolation	
		Direct Rx Bldg evacuation	
		Enter and direct the activities of EOP-03-SCCP	
		Contact TSC/Engineering for EQ envelope evaluations	
		Direct service water alignment to vital header and RHR Room Cooler start	
		Direct manual scram when HPCI exceeds MSOT (HPCI Max Safe = Max Norm of 165°F)	

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EVENT 7: HPCI STEAM LEAK

Time	Pos	EXPECTED Operator Response	Comments
	RO	Diagnose HPCI steam line break	
		Acknowledge and report A-02 5-7 STM LEAK DET AMBIENT TEMP HIGH	
		Acknowledge A-01 4-1 HPCI TURB TRIP SOL ENER	
		Acknowledge A-01 3-5 HPCI ISOL TRIP SIG A INITIATED	
		Acknowledge A-01 4-5 HPCI ISOL TRIP SIG B INITIATED	
		Attempt HPCI isolation and recognize and report failure to isolate	
		Acknowledge and report A-01 5-4 HPCI VALVES MTR OVERLOAD	

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EVENT 7: HPCI STEAM LEAK

Time	Pos	EXPECTED Operator Response	Comments
	BOP	Acknowledge and report UA-03 2-7 AREA RAD RX BLDG HIGH (SCCP entry)	
		Enter and announce AOP-05.0, direct AO to close PIV-33	
		Announce Rx Bldg evacuation	
		Acknowledge UA-05 1-9 FAN CLG UNIT CS PUMP RM A INL PRESS LO	
		Acknowledge UA-05 2-9 FAN CLG UNIT CS PUMP RM B INL PRESS LO	
		Align service water to vital header and start RHR Room Coolers	
		Acknowledge UA-05 6-7 RX BLDG STATIC PRESS DIFF – LOW	

EVENTS 8 and 9: SCRAM / RHR ROOM COOLERS TRIP	
Simulator Operator Actions	

Simulator Operator Role Play	
	If contacted by I&C to investigate RHR Room Cooler trip, acknowledge request.

Evaluator Notes	
Plant Response:	
Objectives:	SRO – Direct Reactor Scram when HPCI reaches Max. Safe condition Direct action of RSP RO – Perform Scram immediate actions Restore and maintain RPV water level as directed by SRO Stabilize RPV pressure as directed by SRO BOP – Maintain stable condition on Balance of Plant Perform actions as directed by CR
Success Path:	Scram immediate actions are completed, RPV pressure and level are stabilized and controlled within band.

EVENTS 8 and 9: SCRAM / RHR ROOM COOLERS TRIP			
Time	**Pos**	**EXPECTED Operator Response**	**Comments**
	SRO	***Direct a reactor manual scram when HPCI area reaches its Max Safe Operating Value.***	***Critical Task - Insert a reactor scram when HPCI reaches its Max Safe Operating Value. (HPCI Area Temperature Max normal is equal to Max Safe)***
		Direct actions in RSP and RVCP.	
		Provide pressure band to ROs, 800-1000 psig	
		Direct RPV level be maintained 166-206 inches	
		Monitor Containment parameters	
		Contact I&C/Maintenance to for DG4 and HPCI failures	
	RO	***Insert Reactor scram as directed by SRO***	***Critical Task - Insert a reactor scram when HPCI reaches its Max Safe Operating Value. (HPCI Area Temperature Max normal is equal to Max Safe)***
		Perform Scram Immediate Actions	Enclosure 2
		Recognize failure of manual scram – Insert ARI	
		Stabilize pressure as directed by the SRO, 800-1000 psig	
		Restore and maintain RPV water level 166-206 inches	
	BOP	Perform Balance of Plant actions and as directed by CRS	

EVENTS 10 and 11: EMERGENCY DEPRESSURIZATION / SRVs FAILED TO OPEN / Termination	
Simulator Operator Actions	
	When directed by the Lead Evaluator, place the simulator in FREEZE
	DO NOT RESET THE SIMULATOR PRIOR TO RECEIPT OF CONCURRENCE TO DO SO FROM THE LEAD EXAMINER

Simulator Operator Role Play	

Evaluator Notes	
Plant Response:	After room trip, 2 areas will reach Max Safe Temperature and ED will be required. Two ADS valves will fail to open which will require opening 2 additional SRVs.
Objectives:	SRO – Evaluate plant conditions and direct Emergency Depressurization. RO – Perform actions for Emergency Depressurization BOP – Assist with re-flooding as directed by SRO
Success Path:	Reactor is depressurized and water level is in normal band.
Scenario Termination:	***Control Rods are inserted, Reactor is depressurized, level is being restored to normal band.***

EVENT 8: EMERGENCY DEPRESSURIZATION / SRVs FAILED TO OPEN

Time	Pos	EXPECTED Operator Response	Comments
	SRO	Continue reactor cooldown per RVCP direction.	
		Direct Emergency Depressurization when two plant areas exceed their Max Safe Temperature.	***Critical Task - Perform Emergency Depressurization when two plant areas exceed Max Safe Temperature.***
		Direct RO/BOP to open 7 ADS valves.	
		If informed by RO/BOP that 2 SRVs failed to open, direct opening additional SRVs until 7 SRVs are open.	
		Enter PCCP when torus temperature exceeds 95°F. Directs all available loops to be placed in Torus Cooling.	
	RO/ BOP	Recognize and report failure of RHR Room coolers.	S RHR will not start. N RHR will trip 5 min after the leak started.
	RO/ BOP	***Open seven ADS valves as directed by SRO.***	***Critical Task - Perform Emergency Depressurization when two plant areas exceed Max Safe Temperature.***
	RO/ BOP	Recognize failure of 2 ADS valves to OPEN and report to SRO.	SRVs A and C fail to open
	RO/ BOP	Open 2 additional SRVs as directed by SRO.	
	RO/ BOP	Maintain reactor water level as directed by SRO.	Should use condensate system via SULCV.
	RO/ BOP	Place available loops in Torus Cooling IAW hard card.	See Enclosure 3 for SPC Hard Card actions.

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Enclosure 1, 2OP-32, Section 8.5

C
Continuou
Use

8.5 Transferring to Standby Condensate Pump

8.5.1 Initial Conditions

1. At least one condensate pump in operation. ☐

8.5.2 Procedural Steps

1. **DESIGNATE** the on coming and off going condensate pumps below: ☐

   Oncoming Condensate Pump:________________

   Offgoing Condensate Pump:__________________

2. **DIRECT** Radwaste Operator to perform the following:

   – **PLACE** an additional CFD in service to prevent override or bypass condition during condensate pump transfer. ☐

   – **PLACE** an additional CDD in service as necessary. ☐

   – **MONITOR** for proper operation of hotwell level control. ☐

3. **ENSURE** proper motor oil level for oncoming condensate pump. ☐

4. **ENSURE** the condensate pump motor TBCCW outlet temperature for the oncoming condensate pump being started is less than or equal to 95°F:

   – *COND PMP 1A MOT CCW OUTLET TEMP IND, TCC-TI-770* ☐

   – *COND PMP 1B MOT CCW OUTLET TEMP IND, TCC-TI-771* ☐

   – *COND PMP 1C MOT CCW OUTLET TEMP IND, TCC-TI-772* ☐

Enclosure 1, 2OP-32, Section 8.5

8.5.2 Procedural Steps

5. **CHECK** status of the following alarms.

   – *GEN-XFMR* PRIMARY *L/O UNIT TRIP* (UA-13 1-1) ☐

   – *GENERATOR DIFF L/O UNIT TRIP* (UA-13 1-2) ☐

   – *GEN-XFMR BACKUP L/O UNIT TRIP* (UA-13 1-3) ☐

6. **IF** all the alarms listed in Step 8.5.2.5 are clear, **THEN PERFORM** the following:(CR-717090)

   a. **PLACE** the following switches in *ENABLED* for the designated offgoing condensate pump:

      – *UNIT TRIP LOAD SHED SELECTOR SWITCH* ☐

      – *LOCA LOAD SHED SELECTOR SWITCH* ☐

   b. **PLACE** oncoming condensate pump mode selector switch in *MAN*. ☐

      – **CONFIRM** its discharge valve closes. ☐

Enclosure 1, 2OP-32, Section 8.5

8.5.2 Procedural Steps

CAUTION

When only one reactor feed pump is in service, then starting a third condensate pump may cause *OFF GAS A CONDENSERS CONDENSATE SUPPLY LINE RELIEF VALVE, CO-RV-2*, or *OFF GAS B CONDENSERS CONDENSATE SUPPLY LINE RELIEF VALVE, CO-RV-3*, to lift. This will result in increased leakage to the Equipment Drain System.

CAUTION

Experience has shown that condensate system dissolved oxygen transients can cause automatic isolation of the condensate oxygen injection system during condensate pump starting evolutions.

c. **START** the oncoming condensate pump.

- **CONFIRM** its discharge valve opens. ☐

d. **WHEN** condensate pump discharge pressure stabilizes, **THEN PERFORM** the following:

(1) **STOP** designated offgoing condensate pump ☐

(2) **IF** *FEEDWATER LINE ISOLATION VALVES, B21-F032A* and *FEEDWATER LINE ISOLATION VALVES, B21-F032B* are open, **THEN PLACE** the stopped condensate pump mode switch in *AUTO*. ☐

e. **Place** the following switchs for the pump started in step 8.5.2.6.c in *DISABLED*:

- *UNIT TRIP LOAD SHED SELECTOR SWITCH* ☐
- *LOCA LOAD SHED SELECTOR SWITCH* ☐

f. **GO TO** Step 8.5.2.12. ☐

Enclosure 1, 2OP-32, Section 8.5

8.5.2 Procedural Steps

7. **PLACE** oncoming condensate pump mode selector switch in *MAN*. ☐

   – **CONFIRM** its discharge valve closes. ☐

CAUTION

When only one reactor feed pump is in service, then starting a third condensate pump may cause *OFF GAS A CONDENSERS CONDENSATE SUPPLY LINE RELIEF VALVE, CO-RV-2*, or *OFF GAS B CONDENSERS CONDENSATE SUPPLY LINE RELIEF VALVE, CO-RV-3*, to lift. This will result in increased leakage to the Equipment Drain System.

CAUTION

Experience has shown that condensate system dissolved oxygen transients can cause automatic isolation of the condensate oxygen injection system during condensate pump starting evolutions.

8. **PLACE** the following switches in *DISABLED* for the condensate pump to be started:

   – *UNIT TRIP LOAD SHED SELECTOR SWITCH* ☐

   – *LOCA LOAD SHED SELECTOR SWITCH* ☐

9. **START** the selected oncoming condensate pump. ☐

   – **CONFIRM** its discharge valve opens. ☐

10. **WHEN** condensate pump discharge pressure stabilizes, **THEN PERFORM** the following:

    a. **STOP** selected condensate pump. ☐

    b. **IF** *FEEDWATER LINE ISOLATION VALVE, B21-F032A* and *FEEDWATER LINE ISOLATION VALVE, B21-F032B* are open, **THEN PLACE** the stopped condensate pump mode switch in *AUTO*. ☐

Enclosure 1, 2OP-32, Section 8.5

8.5.2 Procedural Steps

11. **PLACE** the following switches in *ENABLED* for the condensate pump just stopped in Step 8.5.2.10.a.
    - *UNIT TRIP LOAD SHED SELECTOR SWITCH* ☐
    - *LOCA LOAD SHED SELECTOR SWITCH* ☐
12. **DIRECT** Radwaste Operator to remove additional CFD or CDD placed in service in Step 8.5.2.1. ☐
13. **DIRECT** Radwaste Operator to monitor effluent conductivity for each CDD in service. ☐
14. **COMPLETE** Attachment 7A. ☐

Enclosure 2, SCRAM Actions

Unit 2 Scram Immediate Actions (0EOP-01-UG)

SCRAM IMMEDIATE ACTIONS

1. **Ensure** SCRAM valves OPEN by manual SCRAM or ARI initiation.
2. **WHEN** steam flow less than 3×10^6 lb/hr,
   THEN place reactor mode switch in SHUTDOWN.
3. **IF** reactor power below 2% (APRM downscale trip),
   THEN trip main turbine.
4. **Ensure** master RPV level controller setpoint at +170 inches.
5. **IF:**
   - Two reactor feed pumps running

     AND
   - RPV level above +160 inches

     AND
   - RPV level rising,

   THEN trip one.

Enclosure 3, Page 1 of 2

Emergency Suppression Pool Cooling Using Loop A (2OP-17)

NOTE: This attachment is NOT to be used for normal system operations.

START RHR SW A LOOP (CONV)

- OPEN *SW-V101* ☐
- CLOSE *SW-V143* ☐
- START CSW PUMPS AS NEEDED ☐
- IF LOCA SIGNAL IS PRESENT THEN ☐
  PLACE RHR SW BOOSTER PUMPS
  A & C LOCA OVERRIDE SWITCH
  TO MANUAL OVERRIDE
- START RHR SW PMP ☐
- ADJUST *E11-PDV-F068A* ☐
- ESTABLISH CLG WTR TO VITAL HDR ☐
- START ADDITIONAL RHR SW PUMP AND ADJUST FLOW AS NEEDED ☐

START RHR SW A LOOP (NUC)

- OPEN *SW-V105* ☐
- OPEN *SW-V102* ☐
- CLOSE *SW-V143* ☐
- START PUMPS ON NSW HDR AS NEEDED ☐
- IF LOCA SIGNAL IS PRESENT THEN ☐
  PLACE RHR SW BOOSTER PUMPS A & C LOCA
  OVERRIDE SWITCH TO MANUAL OVERRIDE
- START RHR SW PMP ☐
- ADJUST *E11-PDV-F068A* ☐
- ESTABLISH CLG WTR TO VITAL HDR ☐
- START ADDITIONAL RHR SW PUMP AND ADJUST FLOW AS NEEDED ☐

START RHR LOOP A

- IF LOCA SIGNAL IS PRESENT, THEN VERIFY SPRAY LOGIC IS MADE UP ☐
- IF *E11-F015A* IS OPEN, THEN CLOSE *E11-F017A* ☐
- START LOOP A RHR PMP ☐
- OPEN *E11-F028A* ☐
- THROTTLE *E11-F024A* ☐
- THROTTLE *E11-F048A* ☐
- START ADDITIONAL LOOP A RHR PMP AND ADJUST FLOW AS NEEDED ☐

1
1

2/1061
S/1062

Enclosure 3, Page 2 of 2

Emergency Suppression Pool Cooling Using Loop B (2OP-17)

NOTE: This attachment is **NOT** to be used for normal system operations.

START RHR SW B LOOP (NUC)

- OPEN *SW-V105* ☐
- CLOSE *SW-V143* ☐
- START PMPS ON NSW HDR AS NEEDED ☐
- IF LOCA SIGNAL IS PRESENT THEN ☐
  PLACE RHR SW BOOSTER PUMPS
  B & D LOCA OVERRIDE SWITCH
  TO MANUAL OVERRIDE
- START RHR SW PMP ☐
- ADJUST *E11-PDV-F068B* ☐
- ESTABLISH CLG WTR TO VITAL HDR ☐
- START ADDITIONAL RHR SW PUMP AND ADJUST FLOW AS NEEDED ☐

START RHR SW B LOOP (CONV)

- OPEN *SW-V101* ☐
- OPEN *SW-V102* ☐
- CLOSE *SW-V143* ☐
- START CSW PUMPS AS NEEDED ☐
- IF LOCA SIGNAL IS PRESENT THEN ☐
  PLACE RHR SW BOOSTER PUMPS B & D LOCA
  OVERRIDE SWITCH TO MANUAL OVERRIDE
- START RHR SW PMP ☐
- ADJUST *E11-PDV-F068B* ☐
- ESTABLISH CLG WTR TO VITAL HDR ☐
- START ADDITIONAL RHR SW PUMP AND ADJUST FLOW AS NEEDED ☐

START RHR LOOP B

- IF LOCA SIGNAL IS PRESENT, THEN VERIFY SPRAY LOGIC IS MADE UP ☐
- IF *E11-F015B* IS OPEN, THEN CLOSE *E11-F017B* ☐
- START LOOP B RHR PMP ☐
- OPEN *E11-F028B* ☐
- THROTTLE *E11-F024B* ☐
- THROTTLE *E11-F048B* ☐
- START ADDITIONAL LOOP B RHR PMP AND ADJUST FLOW AS NEEDED ☐

ATTACHMENT 1 - Scenario Quantitative Attribute Assessment

Category	NUREG 1021 Rev. 2 Supp. 1 Req.	Scenario Content
Total Malfunctions	5-8	7
Malfunctions after EOP Entry	1-2	2
Abnormal Events	2-4	3
Major Transients	1-2	2
EOPs Used	1-2	2
EOP Contingency	0-2	2
Run Time	60-90 min	90
Crew Critical Tasks	2-3	2
Tech Specs	2	2
Instrument / Component Failures before Major	2 – OATC 2 - BOP	4
Instrument / Component Failures after Major	2	2
Normal Operations	1	1
Reactivity manipulation	1	1

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ATTACHMENT 2 – Shift Turnover

Brunswick Unit 2 Plant Status					
Station Duty Manager:				Workweek Manager:	
Mode:	1	Rx Power:	95%	Gross*/Net MWe*:	934 / 909
Plant Risk: Current EOOS Risk Assessment is:				Green	
SFP Time to 200 Deg F:	128.7 hrs			Days Online:	142 days
Turnover:	Feedwater Temperature Reduction will be implemented this weekend.				
Protected Equipment:					
Comments:	APRM 2 is INOP and bypassed. 2C TCC Pump is in service on Unit One. Swap Condensate Pumps (Start 2C, Shutdown 2B), for routine maintenance.				

Facility: Brunswick Scenario No.: NRC 4 Op-Test No.: FINAL

Examiners: ______ Operators: SRO

RO

BOP

Initial Conditions: The plant is operating at 3.8% power, IRM A is bypassed due to spiking and the paperwork is being evaluated for its return to service. Reactor power will be raised by pulling control rods.

Turnover: Raise Reactor power and un-bypass IRM A when paper work is completed.

Event No.	Malf. No.	Event Type*	Event Description
1	ZA411	C-RO C-SRO	DWEDT Pump fails to auto start
2		N	Place RFPT level control in automatic
3		R	Raise Power
4	NI018F	C-RO C-SRO	IRM fails upscale – Tech Spec
5	CF035F	C-BOP C-SRO	SULCV fails closed – AOP-23.0
6	SL_IASL RB	C	SLC Pump Breaker trip – Tech Spec
7		C-BOP C-SRO	CW Pump breaker trip
8	EE009A	M	LOOP – AOP-36.1
9	DG004F	C	DG3 Fails to start
10	DG027F	C	DG4 Trips on Differential Overcurrent
11	NB009F	M	Small break LOCA
12	ES020F ES013F	C	Loss of HP Injection
13		M C	ED on level – LP ECCS Auto Start Failures

* (N)ormal, (R)eactivity, (I)nstrument, (C)omponent, (M)ajor

SCENARIO DESCRIPTION SUMMARY – 2015 NRC Scenario 4

Event	Description
1	Annunciator A-04 1-1, Drywell Equip Drain Sump Lvl Hi, will annunciate and the sumps will not auto start. One of the sump pumps will need to be manually started
2	Step 6.3.46 of 0GP-02, Approach to Criticality and Pressurizations of the Reactor will be completed starting at Step 6.3.46.
3	The crew will raise power by pulling control rods in preparation for placing the Mode switch to RUN. Rod pulls will commence at Step 166 (10-23 @ 12) of the A2X sequence.
4	While withdrawing control rods, IRM C will fail upscale causing a rod block and half scram. SRO will address IRM A and C inoperability IAW TS 3.3.1.1. Once addressed, I&C will report IRM A is ready to be returned to service following proper channel check. The crew will take the actions of the APP and bypass IRM C and reset the half scram.
5	Circulating Water Pump A will trip on motor winding fault, and another Circ Water pump will be started.
6	SLC Pump A breaker will trip and Tech Spec 3.1.7 will be entered.
7	Control rods will continue to be withdrawn raising power. The SULCV will fail closed stopping feed flow to the vessel. Reactor water level will drop requiring action to re-establish flow to the vessel.
8, 9, 10	A loss of off-site power will occur and DG3 will not auto start. DG4 will trip on differential overcurrent shortly after starting.
11, 12	A small break LOCA with failure of HP injection systems will require Emergency Depressurization when level reaches LL4.
13	ED will be required when level reaches LL4. Low pressure ECCS systems will fail to auto start.

CREW CRITICAL TASKS

Description
Start DG3 and ensure the output breaker closes to energize E3.
Perform Emergency Depressurization when RPV level cannot be restored and maintained above LL4.

DUKE ENERGY

BRUNSWICK TRAINING SECTION

OPERATIONS TRAINING

INITIAL LICENSED OPERATOR

SIMULATOR EVALUATION GUIDE

2015 NRC SCENARIO 4

LOW POWER SCENARIO, LOOP, LOSS OF HP INJECTION, ED ON LEVEL

REVISION 0

Developer: Lou Sosler	Date: 9/11/2015
Technical Review: John Biggs	Date: 9/23/2015
Validator: Thomas Baker	Date: 9/11/2015
Validator: Brian Moschet	Date: 9/11/2015
Facility Representative: Jerry Pierce	Date: 9/23/2015

REVISION SUMMARY	
0	Exam scenario for 2015 NRC Exam.

TABLE OF CONTENTS

1.0 SCENARIO OUTLINE

Event	Malf. No.	Type*	Event Description
1	ZA411	C-RO C-SRO	DWEDT Pump fails to auto start
2		N	Place RFPT level control in automatic
3		R	Raise Power
4	NI018F	C-RO C-SRO	IRM fails upscale – Tech Spec
5	CF035F	C-BOP C-SRO	SULCV fails closed – AOP-23.0
6	SL_IASLRB	C	SLC Pump Breaker trip – Tech Spec
7		C-BOP C-SRO	CW Pump breaker trip
8	EE009A	M	LOOP – AOP-36.1
9	DG004F	C	DG3 Fails to start
10	DG027F	C	DG4 Trips on Differential Overcurrent
11	NB009F	M	Small break LOCA
12	ES020F ES013F	C	Loss of HP Injection
13		M C	ED on level – LP ECCS Auto Start Failures
*(N)ormal, (R)eactivity, (C)omponent or Instrument, (M)ajor			

2.0 SCENARIO DESCRIPTION SUMMARY

Event	Description
1	Annunciator A-04 1-1, Drywell Equip Drain Sump Lvl Hi, will annunciate and the sumps will not auto start. One of the sump pumps will need to be manually started
2	Step 6.3.46 of 0GP-02, Approach to Criticality and Pressurizations of the Reactor will be completed starting at Step 6.3.46.
3	The crew will raise power by pulling control rods in preparation for placing the Mode switch to RUN. Rod pulls will commence at Step 166 (10-23 @ 12) of the A2X sequence.
4	While withdrawing control rods, IRM C will fail upscale causing a rod block and half scram. SRO will address IRM A and C inoperability IAW TS 3.3.1.1. Once addressed, I&C will report IRM A is ready to be returned to service following proper channel check. The crew will take the actions of the APP and bypass IRM C and reset the half scram.
5	Circulating Water Pump A will trip on motor winding fault, and another Circ Water pump will be started.
6	SLC Pump A breaker will trip and Tech Spec 3.1.7 will be entered.
7	Control rods will continue to be withdrawn raising power. The SULCV will fail closed stopping feed flow to the vessel. Reactor water level will drop requiring action to re-establish flow to the vessel.
8, 9, 10	A loss of off-site power will occur and DG3 will not auto start. DG4 will trip on differential overcurrent shortly after starting.
11, 12	A small break LOCA with failure of HP injection systems will require Emergency Depressurization when level reaches LL4.
13	ED will be required when level reaches LL4. Low pressure ECCS systems will fail to auto start.

3.0 CREW CRITICAL TASKS

Description
Start DG3 and ensure the output breaker closes to energize E3.
Perform Emergency Depressurization when RPV level cannot be restored and maintained above LL4.

4.0 TERMINATION CRITERIA

When control rods are inserted, the Reactor is depressurized, level is being restored to normal band, and Containment and Drywell Sprays are being placed in service, the scenario may be terminated.

5.0 IMPLEMENTING REFERENCES

NOTE: Refer to the most current revision of each Implementing Reference.

Number	Title
A-04, 1-1	DRYWELL EQUIP DRAIN SUMP LVL HI
UA-02, 4-5	GLAND SEAL VACUUM LOSS
2OP-26.1, Section 8.1	SHIFTING STEAM PACKING EXHAUSTERS
A-05, 2-4	IRM UPSCALE
A-05, 3-4	IRM A UPSCALE/INOP
A-05, 1-7	REACTOR AUTO SCRAM SYS A
A-05, 4-7	NEUT MON SYS TRIP
A-05, 2-2	ROD OUT BLOCK
A-07, 2-2	REACTOR WATER LEVEL HIGH/LOW
0AOP-23.0	CONDENSATE/FEEDWATER SYSTEM FAILURES
A-04, 4-5	SQUIB VALVE CONTINUITY LOSS
UA-01, 1-7	CW PUMP A TRIP
0AOP-36.1	LOSS OF ANY 4160 V BUSSES OR 480V E-BUSSES
2EOP-01-RSP	REACTOR SCRAM PROCEDURE
2EOP-01-PCCP	PRIMARY CONTAINMENT CONTROL PROCEDURE
0EOP-01-EDP	EMERGENCY DEPRESSURIZATION PROCEDURE
0EOP-02-PCCP	PRIMARY CONTAINMENT CONTROL PROCEDURE

6.0 SETUP INSTRUCTIONS

1. **PERFORM** TAP-409, Miscellaneous Simulator Training Guidelines, Attachment 5, Checklist for Simulator Exam Security.
2. **RESET** the Simulator to IC-06 (Saved in IC-180).
3. **ENSURE** the RWM is set up as required for the selected IC.
4. **ENSURE** appropriate keys have blanks in switches.
5. **RESET** alarms on SJAE, MSL, and RWM NUMACs.
6. **ENSURE** no rods are bypassed in the RWM.
7. **PLACE** all SPDS displays to the Critical Plant Variable display (#100).
8. **ENSURE** hard cards and flow charts are cleaned up
9. **TAKE** the SIMULATOR OUT OF FREEZE
10. **ALIGN** the plant as follows:

Manipulation
Insert control rods until Step 165 of GP-10, Sequence A2X is completed.

11. **LOAD** Scenario File.
12. **IF desired**, take a **SNAPSHOT** and save into an available IC for later use.
13. **PLACE** a clearance on the following equipment.

Component	Position
IRM A (Blue Tag)	Bypassed

14. **INSTALL** Protected Equipment signage and **UPDATE** RTGB placard as follows:
    a. ADHR / FPC/ Demin Transfer Pump
15. **ENSURE** each Implementing References listed in Section 7 is intact and free of marks.
16. **ENSURE** all materials in the table below are in place and marked-up to the step identified.

Required Materials
None

17. **ENSURE** Station Duty Manager and Work Week Manager names are filled in on the Shift Turnover Sheet.

18. **ADVANCE** the recorders to prevent examinees from seeing relevant scenario details.

19. **PROVIDE** Shift Briefing sheet for the CRS.

20. **VERIFY** all actions contained in TAP-409, Miscellaneous Simulator Training Guidelines, Attachment 4, Simulator Training Instructor Checklist, are complete.

7.0 INTERVENTIONS

TRIGGERS

Trig	Type	ID
1	Annunciator	ZA411 - [DRYWELL EQUIP DRAIN SUMP LVL HI]
2	Malfunction	NI018F - [IRM C FAILS HI]
3	Malfunction	CW039F - [CIRC WATER INTAKE PUMP MOTOR WINDING FAULT]
4	Remote Function	SL_IASLRB - [2B SLC PUMP MOTOR BKR]
5	Malfunction	CF035F - [S/U LVL CONT VLV FAILS CLOSED]
6	Malfunction	DG027F - [DG4 DIFFERENTIAL FAULT]
6	Malfunction	EE009F - [LOSS OF OFF-SITE POWER]
7	Malfunction	ES020F - [RCIC TURBINE SPEED CONTROL FAILURE]
7	Malfunction	NB009F - [SMALL RECIRC PMP SUCT LINE RUPTURE]
8	Trigger Command	mfd:dg004f
10	Malfunction	CW039F - [CIRC WATER INTAKE PUMP MOTOR WINDING FAULT]
11	Remote Function	ED_ZIEDH14 - [PNL 2AB-TB PWR (E8=NORM/E7=ALT)]
12	Remote Function	SW_VHSW146L - [CONV SW TO RBCCW HXS V146]
13	Remote Function	EP_IACS993P - [DW CLR A & D OVERIDE - NORMAL/STOP]
13	Remote Function	EP_IACS994P - [DW CLR B & C OVERIDE - NORMAL/STOP]
14	Remote Function	SL_IASLCSRC - [SLC JUMPER HOSE SOURCE (ALT=FP / NORM=DEMIN)]
14	Remote Function	SL_IASLCTST - [SLC SUCT. LINEUP (NORM=SLC TNK / ALT=JUMPER HOSE)]
15	Remote Function	ED_IARKAI0 - [X-TIE BKR E8-E7 (AI0) RACK STATUS]
15	Remote Function	ED_IARKAX5 - [X-TIE BKR E7-E8 (AX5) RACK STATUS]

Trig #	Trigger Text
8	K4F14AB8 - [DIESEL GENERATOR AUTO-MODE START]

MALFUNCTIONS

Malf ID	Mult ID	Description	Current Value	Target Value	Rmp time	Actime	Dactime	Trig
DG004F		DG3 AUTO START FAILURE	True	True				
ES041F		RCIC FAILURE TO AUTO START	True	True				
ES016F		HPCI HYDRAULIC SYSTEM FAILURE	True	True				
ES043F		CORE SPRAY A FAILURE TO AUTO START	True	True				
ES044F		CORE SPRAY B FAILURE TO AUTO START	True	True				
ES045F		RHR A FAILURE TO AUTO START	True	True				
ES046F		RHR B FAILURE TO AUTO START	True	True				
NI018F		IRM C FAILS HI	False	True				2
CW039F	A	CIRC WATER INTAKE PUMP MOTOR WINDING FAULT	False	True				3
CF035F		S/U LVL CONT VLV FAILS CLOSED	False	True				5
DG027F		DG4 DIFFERENTIAL FAULT	False	True		00:02:00		6
EE009F		LOSS OF OFF-SITE POWER	False	True				6
ES020F		RCIC TURBINE SPEED CONTROL FAILURE	False	True		00:02:00		7
NB009F	A	SMALL RECIRC PMP SUCT LINE RUPTURE	0.00	50.00000	00:10:00			7
CW039F	D	CIRC WATER INTAKE PUMP MOTOR WINDING FAULT	False	True				10

REMOTES

Remf Id	Mult Id	Description	Current Value	Target Value	Rmp time	Actime	Trig
SL_IASLRB		2B SLC PUMP MOTOR BKR	CLOSE	OPEN			4

PANEL OVERRIDES

Tag ID	Description	Position / Target	Actual Value	Override Value	Rmp time	Actime	Dactime	Trig

ANNUCIATOR OVERRIDES

Window	Description	Tagname	Override Type	OVal	AVal	Actime	Dactime	Trig
1-1	DRYWELL EQUIP DRAIN SUMP LVL HI	ZA411	ON	ON	OFF			1

8.0 OPERATOR RESPONSE AND INSTRUCTIONAL STRATEGIES

EVENT 1: SHIFT TURNOVER / DWEDT PUMP FAILURE	
Simulator Operator Actions	
	Ensure Monitored Parameters is open and Scenario Based Testing Variables are loaded.
NOTE	If the simulator is left in run the DWED Sump Lvl Hi Alarm will annunciate on its own after approximately 50 minutes. (The malfunctions will still work if it is allowed to annunciate)
	At the direction of the Lead Evaluator, **Initiate Trigger 1** to activate the DWED Sump Lvl Hi Annunciator.
	When either sump pump has been running for ~30 seconds **delete malfunction** for the DWED Sump Lvl Hi Annunciator.

Simulator Operator Role Play	
	Acknowledge requests as I&C for troubleshooting DWED Sump Pump auto start failure.

Evaluator Notes	
Plant Response:	Annunciator A-04 (1-1), Drywell Equip Drain Sump Lvl Hi.
Objectives:	RO - Pump the DWEDT
Success Path:	Pumps the DEWDT.
Event Termination:	Go to Event 2 at the direction of the lead evaluator.

EVENT 1: SHIFT TURNOVER / DWEDT PUMP FAILURE

Time	Pos	EXPECTED Operator Response	Comments
	SRO	Conduct shift turnover shift briefing.	
		Direct actions of APPs Direct RO to start DWEDS Pump, if asked.	
	RO	Refer to APP: A-04 (1-1), Drywell Equip Drain Sump Lvl Hi	
		Diagnose failure of DWEDS Pump	
		Start a DWEDS Pump Verifies pump shuts off after a period of time.	
	BOP	Monitors the plant	

EVENT 2: PLACING RFPT CONTROLLER IN AUTOMATIC	
Simulator Operator Actions	

Simulator Operator Role Play	

Evaluator Notes	
Plant Response:	Place RFPT Master Controller in Automatic IAW 0GP-02, Step 6.3.46
Objectives:	SRO – Direct RO to perform Step 6.3.46 of 0GP-02 RO – Place RFPT Level Controller is placed in Automatic
Success Path:	RFPT Master Level Controller is in Automatic and Reactor water level is controlled in band.
Event Termination:	Go to Event 3 at the direction of the Lead Evaluator.

EVENT 2: PLACING RFPT CONTROLLER IN AUTOMATIC

Time	Pos	EXPECTED Operator Response	Comments
	SRO	Direct RO to perform Step 6.3.46 of 0GP-02	
	RO	Monitors the plant	
	BOP	Place RFPT Master Controller in Automatic IAW 0GP-02, Step 6.3.46.	Enclosure 1 contains 0GP-02, Step 6.3.46.

EVENTS 3 and 4: RAISE REACTOR POWER, IRM C FAILURE

	Simulator Operator Actions
	While withdrawing control rods, at the direction of the Lead Evaluator, **Initiate Trigger 2**, to fail IRM C upscale.

	Simulator Operator Role Play
	If asked as the RE, continuous rod withdrawal is allowed.
	If contacted as the RE for IRM C inoperability, acknowledge request.
	When IRM C inoperability has been addressed and by Lead Examiners direction, contact the control room as Ops Center SRO and report IRM A can be declared Operable following a satisfactory channel check.

Evaluator Notes

Plant Response: The crew will continue raising power by pulling control rods in preparation for placing the Mode switch to RUN. Rods pulls will commence at Step 166 (10-23 @ 12) of the A2X sequence. While withdrawing control rods, IRM C will fail upscale causing a rod block and half scram.

Objectives: SRO - Directs and monitor reactor power ascension with control rods
Determine Technical Specification application.
RO - Withdraw control rods to raise reactor power.
Perform actions for IRM C failure

Success Path: Declare IRM A operable by channel check and bypass IRM C with tracking LCO for IRM C.

Event Termination: Go to Event 4 at the Direction of the Lead Evaluator.

EVENTS 3 and 4: RAISE REACTOR POWER, IRM C FAILURE			
Time	**Pos**	**EXPECTED Operator Response**	**Comments**
	SRO	Ensures no other distracting evolutions are in progress while reactivity controls are being manipulated.	
		Directs RO to raise reactor power by withdrawing control rods IAW 0GP-10 Item 10 Step 166. (Continuous withdrawal allowed).	
		Directs APP reference.	
		Contacts I&C for IRM C failure. May contact Shift Manager also.	
		References TS 3.3.1.1 and determines with IRMs A & C inoperable: Condition A is applicable for Function 1a Required Action A.1 is required within 12 hours, or A.2 is required in 12 hours.	
		May enter TRM 3.3 (Control Rod Block Instrumentation) Function 3 Condition A, Tracking LCO.	
		Evaluates IRM A operability following satisfactory channel check . 2OP-09, Attachment 4, 2.3.4 (Operability Guidance).	Channel Checks are a sufficient WO PMT for SRMs and IRMs at power unless a component failure is suspected in which case an I/V curve and TDR trace is desirable Definitions provide guidance as to how.
		Directs IRM A channel check be performed.	Channel Check definition in the RO DSR.

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EVENTS 3 and 4: RAISE REACTOR POWER, IRM C FAILURE

Time	Pos	EXPECTED Operator Response	Comments
	SRO	Determines IRM A is operable	
		Directs removing IRM A from Bypass	
		Directs bypassing IRM C	
		Directs resetting half scram	
	BOP	Monitors the plant	

EVENTS 3 and 4: RAISE REACTOR POWER, IRM C FAILURE

Time	Pos	EXPECTED Operator Response	Comments
	RO	Commence rod withdrawal at step 166 of GP-10 per guidance of OI-01.02	
		<u>2OP-07 Continuous Rod Withdraw</u> 1. ENSURE ROD SELECT POWER control switch is in ON. 2. SELECT desired control rod by depressing its CONTROL ROD SELECT push button. 3. ENSURE the backlighted CONTROL ROD SELECT push button is brightly illuminated AND the white indicating light on the full core display is also illuminated. 4. ENSURE ROD WITHDRAWAL PERMISSIVE indication has illuminated. 5. CONTINUOUSLY WITHDRAW control rod to position designated on GP pull sheets by holding EMERGENCY ROD IN NOTCH OVERRIDE switch to OVERRIDE, while simultaneously holding ROD MOVEMENT switch to NOTCH OUT. 6. MONITOR control rod position AND nuclear instrumentation while withdrawing the control rod. 7. PERFORM the following for control rods to be fully withdrawn: a. WHEN control rod reaches position 48, THEN PERFORM either of the following: - MAINTAIN the continuous withdraw signal for the desired time - APPLY a separate notch withdraw signal. b. ENSURE control rod does NOT retract beyond position 48. (ref. SR 3.1.3.4) c. RELEASE ROD MOVEMENT and EMERGENCY ROD IN NOTCH OVERRIDE switches, if used. d. ENSURE control rod settles at position 48 AND rod settle light extinguishes. e. ENSURE control rod reed switch position indicators agree with FULL OUT indication on full core display.	
		Stops withdrawing control rods when IRM C fails upscale. *ROD OUT BLOCK*	

EVENTS 3 and 4: RAISE REACTOR POWER, IRM C FAILURE

Time	Pos	EXPECTED Operator Response	Comments
	RO	Determines IRM C failed upscale.	
		Responds and reports applicable alarms for IRM C failing upscale. *A-5* *1-7 REACTOR AUTO SCRAM SYS A* *4-7 NEUT MON SYS TRIP* *2-4 IRM UPSCALE* *2-2 ROD OUT BLOCK* *3-4 IRM A UPSCALE/INOP*	
		A-5 *IRM A UPSCALE/INOP* actions: May Reposition range switch for IRM C to bring indicated power to between 15 and 50 on the 0-125 scale. May verify IRM C Drawer Selector switch (Control Panel H12-P606) is in OPERATE. May notify SRO of Tech Spec applicability	
		May inform SRO IRM C cannot be bypassed and half scram cannot be reset due to IRM A being bypassed.	
		Performs channel check of IRM A for operability. RO DSR Item # 9 (IRM channel check) 2OI-03.2, Definition 5.1.	
		Removes IRM A from Bypass	
		Bypasses IRM C per APP guidance.	
		Resets half scram per APP guidance.	

EVENT 5: CIRC WATER PUMP A TRIP	
Simulator Operator Actions	
	At the direction of the Lead Evaluator, **Initiate Trigger 3**, to initiate CW Pump A trip
	Note: At this low a power level, Condenser vacuum will not change. If crew does not start an idle CW pump, **Trigger 10** will trip an additional CW Pump.

Simulator Operator Role Play	
	If asked as Outside AO, acknowledge request to check pump. After 2-3 minutes, call back and report that shear pin on the traveling screens for CW Pump A broke.
	If asked as TBAO, identify that breaker AB8 on 4160 V Switchgear 2C is tripped on overcurrent. No other abnormalities.
	If asked as I&C to investigate, acknowledge the request.

Evaluator Notes	
Plant Response:	Circ Water Pump A will trip and annunciator UA-01, 1-7, CIRC WATER PUMP A TRIP, will alarm. After investigating the cause of the alarm, another Circ Water Pump should be started IAW the APP. At this power level, Condenser vacuum should not be effected.
Objectives:	SRO - Direct actions of APP-UA-01, 1-7, CIRC WATER PUMP A TRIP Direct Emergency Depressurization BOP – Perform action of APP UA-01, 1-7, CIRC WATER PUMP A TRIP RO – Monitor plant parameters
Success Path:	Another Circ Water pump is be started.

EVENT 5: CIRC WATER PUMP A TRIP

Time	Pos	EXPECTED Operator Response	Comments
	SRO	Direct actions of APP-UA-01, 1-7, CIRC WATER PUMP A TRIP	
	RO	Monitor plant parameters	
	BOP	Take actions IAW APP-A-07, 1-7, CIRC WATER PUMP A TRIP	
		Direct AOs to investigate pump and pump breaker to determine cause of pump trip.	
		Start a Circ Water pump: ☐ Place SC ISOL VALVES MODE SELECTOR switch to D position ☐ Start CWIP 2C ☐ Place switch to C position	

EVENT 6: SLC PUMP BREAKER TRIP

Simulator Operator Actions	
	When directed by the lead evaluator, **Initiate Trigger 4** to fail SLC Pump B.

Simulator Operator Role Play	
	If asked as I&C to investigate, acknowledge the request.
	If asked as RBAO report acrid smell in the area of 2XH and the breaker is tripped for 2B SLC Pump (no fire/smoke in the area).

Evaluator Notes	
Plant Response:	SLC Pump Breaker will trip. TS 3.1.7, Condition A With one SLC subsystem inoperable restore the SLC subsystem to OPERABLE status in 7 days.
Objectives:	SRO – Determine Technical Specifications applications RO – Respond to a trip of SLC Pump Breaker BOP – Monitor plant parameters
Success Path:	Determines TS 3.1.7, Condition A applies.
Event Termination:	Go to Event 7 at the direction of the lead evaluator.

EVENT 6: SLC PUMP BREAKER FAILURE

Time	Pos	EXPECTED Operator Response	Comments
	SRO	Direct actions of APPs	
		Direct I&C to investigate	
		Evaluate Tech Spec 3.1.7, SLC. Condition A, With one SLC subsystem inoperable restore the SLC subsystem to OPERABLE status in 7 days	
	BOP	Plant Monitoring	
	RO	Refer to the appropriate APPs.	
		Diagnose failure of SLC Pump B breaker.	
		Dispatch AO to RB MCC 2XG.	

EVENT 7: SULCV FAILS CLOSED	
	Simulator Operator Actions
	At the direction of the Lead Evaluator, **Initiate Trigger 5** to activate the SULCV failing closed.
	If crew does not respond properly to this event, the Reactor may scram on low water level. If this happens continue to next event. Discuss with Lead Evaluator.

	Simulator Operator Role Play
	If contacted as TBAO to investigate SULCV, acknowledge request.
	If contacted as I&C to investigate failure, acknowledge request.

Evaluator Notes
Plant Response: SULCV fails closed and Reactor water level lowers.
Objectives: SRO - Direct actions for failed SULCV and lowering reactor water level RO - Monitors reactor plant parameters BOP - Take action to respond to a failed SULCV and lowering reactor water level
Success Path: Level restored to normal band by establishing flow through an alternate path
Event Termination: Go to Event 8 at the direction of the Lead Evaluator.

EVENT 7: SULCV FAILS CLOSED

Time	Pos	EXPECTED Operator Response	Comments
	SRO	Direct actions in response to lowering reactor water level. *A-07 2-2, REACTOR WATER LEVEL HIGH/LOW* *A-05 3-3, SRM PERIOD*	
		Direct AOP-23 entry	
		Direct injection to the vessel be established by manually opening one of the following valves: • FW-V120 • FW-V118 • FW-V119	
		Direct manual scram if level control not established and level continues to lower.	
	RO	Recognize and respond to lowering reactor water level (may notice before alarm) APP-A-07 2-2, REACTOR WATER LEVEL HIGH/LOW	
		Diagnose SULCV has failed closed and attempt to OPEN	
		If direct by SRO, insert manual scram	
	BOP	Attempt to establish flow to the vessel by manually opening one of the following valves: • FW-V120 • FW-V118 • FW-V119	
		Monitor plant parameters	

EVENT 8, 9: LOSS OF OFF-SITE POWER / SCRAM / DG3 FAILS TO START / DG4 TRIPS	
Simulator Operator Actions	
	At the discretion of the Lead Evaluator, **Initiate Trigger 6** to initiate a loss of off-site power.
	Trigger 11: 2AB-TB to ALT (ED_ZIEDH14
	Trigger 12: RCC on CSW, Open SW-V146 (SW_VHSW146L)
	If requested to cross-tie of service air, wait 3 minutes and modify Remote Function AI_VHAIV07L, OPEN
Simulator Operator Role Play	
	If asked as load dispatcher, transmission line crews are investigating cause and no current estimates of restoration
	If asked as OAO to investigate E4/DG4, report device 86DP tripped @ E4 switchgear
	If asked as I&C to investigate DG3, DG4 and/or HPCI, acknowledge the request
	If asked to swap Panel 2AB-TB to alternate, notify Sim Operator to activate **Trigger 11** (2 min).
	If requested to align RBCCW to CSW cooling, **Trigger 12** (5 min.)
Evaluator Notes	
Plant Response:	LOOP on Unit 2 (Unit 1 maintains off-site power), Reactor scram, MSIV closure, DG4 auto starts, then trips on differential overcurrent. DG3 fails to start but will start in CR Auto and energize E3. HPCI FIC power failure 2 minutes after LOOP.
Objectives:	SRO - Enter and direct the activities of EOP-01-RSP Direct RPV level and pressure bands Direct entry into AOP-36.1 RO – Perform Scram immediate actions Restore and maintain RPV water level as directed by SRO Stabilize RPV pressure as directed by SRO BOP – Recognize and report loss of 230 kV buses Verify auto start of DGs Recognize and report failure of DG3 to auto start, manually start DGs Recognize and report failure of DG4 due to overcurrent Perform AOP-36.1 actions
Success Path:	Scram immediate actions are completed, RPV pressure and level are stabilized and controlled within band, and plant electrical needs are met via DGs.

EVENT 8, 9: LOSS OF OFF-SITE POWER / SCRAM / DG3 FAILS TO START / DG4 TRIPS

Time	Pos	EXPECTED Operator Response	Comments
	SRO	Direct actions Reactor Scram Procedure	
		Direct RPV pressure be maintained800-1000 psig.	
		Direct RPV level be maintained 166-206 inches	
		Direct entry into AOP-36.1	
		Monitor Containment parameters	
		Direct start of DG3.	***Critical Step***
		Contact I&C/Maintenance to for DG4 and HPCI failures	
	RO	Perform Scram Immediate Actions	
		Stabilize pressure as directed by the SRO	
		Restore and maintain RPV water level 166-206 inches	

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EVENT 8, 9: LOSS OF OFF-SITE POWER / SCRAM / DG3 FAILS TO START / DG3 TRIPS

Time	Pos	EXPECTED Operator Response	Comments
	BOP	Recognize and report loss of 230 kV busses	
		Verify start of DGs	
		Recognize & report failure of DG3 to auto start, auto starts DG3.	***Critical Step***
		Recognize and report failure of DG4 due to overcurrent trip	
		Dispatch OAO to investigate DG4	
		Enter and perform AOP-36.1	
		Start Battery Room HVAC	
		Start Control Building HVAC	
		Start available SW pumps	
		Manually closing SW-V106, NSW supply to RCC	
		Direct RCC cooling water restored to CSW header	
		Direct cross-tie of Service Air	
		Start CRD per OP-08	
		Restore RPS	

EVENT 10, 11, 12: SMALL BREAK LOCA / LOSS OF HP INJECTION / ED

	Simulator Operator Actions
	At the direction of the Lead Evaluator, **Initiate Trigger 7** to initiate a Recirc Pump suction line rupture and failure of the RCIC flow controller.
	Trigger 13: EP_IACS993P, EP_IACS994P
	Trigger 14: SL_IASLCTS to ALT, SL-IASLCSRC to ALT
	Trigger 15: ED_IARJAX5 - IN, ED_IARKA10 - IN

	Simulator Operator Role Play
	If requested to monitor DGs, acknowledge alarms using DG Local Alarm Panel(Instructor Aids), report alarms if requested
	If requested to defeat drywell cooler LOCA lockout, wait 2 minutes and then initiate **Trigger 13** (EP_IACS993P, EP_IACS994P). Report actions in the back panels are complete.
	If asked as I&C to investigate and RCIC, acknowledge the request
	If asked to investigate HPCI Aux Oil Pump, acknowledge request.
	If requested to transfer SLC suction to demin water, wait two minutes then modify Remote SL_IASLCTS to ALT, SL-IASLCSRC to ALT (**Trigger 14)**
	If requested to rack in E7-E8 crosstie breakers, **initiate Trigger 15**, ED_IARJAX5, IN, ED_IARKA10, IN

Evaluator Notes	
Plant Response:	RPV level lowers below LL2, HPCI fails to start and is unavailable, RCIC fails to start, but can be started manually. After a couple minutes, RCIC goes into low speed oscillation.
Objectives:	SRO - Enter and direct the activities of EOP-01-RVCP Direct ED when LL4 is reached. RO – Restore and maintain RPV water level as directed by SRO Stabilize RPV pressure as directed by SRO Maximize CRD flow using Hard Card, then SEP-09 Open 7 ADS valves as directed by CRS BOP – Diagnose failure of HPCI Aux Oil Pump Perform Alternate Coolant Injection using SLC IAW LEP-01 Diagnose failure of RCIC speed control Trip RCIC to avoid prolonged low speed operation
Success Path:	Operation of all high pressure injection system is attempted. Reactor is depressurized when LL4 cannot be maintained. Reactor is re-flooded, and Containment parameters are addressed.
Scenario Termination:	***Reactor is depressurized, level is being restored to normal band, Containment and Drywell Sprays are being placed in service.***

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EVENT 10, 11, 12: SMALL BREAK LOCA / LOSS OF HP INJECTION / ED

Time	Pos	EXPECTED Operator Response	Comments
	SRO	Direct actions RVCP	
		Direct ED when LL4 is reached.	***Critical Task***
		Monitor Containment parameters	
		Enters and Directs actions of PCCP:	
		Direct Spraying Torus.	
		Direct Spraying Drywell.	
		Direct Cross-Tie actions IAW 0AOP-36.1	
	RO	Ensure LL3 actuations have occurred.	
		Identify that LP ECCS Systems (RHR/CS) fail to auto start – manually start pumps and open discharge valves.	
		When directed by SRO, Open 7 ADS valves	***Critical Task***
	BOP	Monitor plant parameters	
		Operate LP ECCS as required to restore and maintain level as directed by the SRO.	
		Identify that LP ECCS Systems (RHR/CS) fail to auto start – manually start pumps and open discharge valves.	
		Place Suppression Pool Sprays in service IAW SEP-03	See Enclosure 2
		Place Drywell Sprays in service IAW SEP-02.	See Enclosure 3
		Perform cross-tie actions IAW0AOP-36.1 to cross-tie busses E7 and E8.	

Enclosure 1 **Placing RFPT in Auto**

Page 1 of 2

46. **<u>WHEN</u>** reactor feed pump discharge pressure is greater than 900 psig, **<u>THEN</u> place** C32-SIC-R600 (Mstr RFPT Sp/Rx Lvl Ctl) in A (automatic) as follows: .... ________

a. **Ensure** C32-SIC-R600 (Mstr RFPT Sp/Rx Lvl Ctl), in M (manual) .... ________

b. **Ensure** Feedwater Control Mode Select in 1 ELEM .... ________

c. **Depress** SEL pushbutton on C32-SIC-R601A(B) [RFPT A(B) Sp Ctl] until A(B) BIAS is indicated and **ensure** bias is set to 0% .... ________

d. **Depress** SEL pushbutton on C32-SIC-R601A(B) [RFPT A(B) Sp Ctl] until PMP A(B) DEM is displayed .... ________

e. **Depress** SEL pushbutton on C32-SIC-R600 (Mstr RFPT Sp/Rx Lvl Ctl), until MASTR DEM is displayed .... ________

f. Using the raise and lower pushbuttons on C32-SIC-R600 (Mstr RFPT Sp/Rx Lvl Ctl), **set** MASTR DEM to equal the PMP A(B) DEM value displayed on C32-SIC-R601A(B) [RFPT A(B) Sp Ctl] .... ________

g. **Depress** A/M pushbutton on C32-SIC-R601A(B) [RFPT A(B) Sp Ctl] and **confirm** the following:

- Indicator on control station changes to A (automatic) .... ________
- PMP DEM signal remains unchanged .... ________

h. **Depress** SEL pushbutton on the out-of-service C32-SIC-R601A(B) [RFPT A(B) Sp Ctl] until LVL ERROR is indicated and **confirm** LVL ERROR is approximately 0 inches .... ________

i. **Depress** A/M pushbutton on C32-SIC-R600 (Mstr RFPT Sp/Rx Lvl Ctl) and **confirm** the indicator on the control station changes to A (automatic) .... ________

Enclosure 1

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6.3 Heating And Pressurization Of The Reactor (continued)

j. **Confirm** signals for PMP A(B) DEM on C32-SIC-R601A(B) [RFPT A(B) Sp Ctl] and VALVE DEM on FW-LIC-3269 (SULCV Ctl) remain unchanged .... ________

k. **Depress** A/M pushbutton on FW-LIC-3269 (SULCV Ctl) and **confirm** the indicator on the control station changes to M (manual) .... ________

> **CAUTION**
>
> Momentarily depressing the raise or lower pushbuttons on FW-LIC-3269 (SULCV Ctl) will cause valve demand to change in increments of 0.1%. Continually depressing the raise or lower pushbuttons will cause valve demand to change at an exponential rate. .... ☐

l. Using raise pushbutton on FW-LIC-3269 (SULCV Ctl), slowly **open** the SULCV until VALVE DEM is 100% .... ________

m. **Confirm** reactor water level is being maintained between 182 and 192 inches .... ________

Enclosure 2

Page 1 of 2

1.0 ENTRY CONDITIONS

- As directed by Emergency Operating Procedures (EOPs)

2.0 INSTRUCTIONS

2.1 Torus Spray

2.1.1 Manpower Required

- 1 Reactor Operator

2.1.2 Special Equipment

None

2.1.3 Torus Spray Actions

1. **Confirm** torus pressure above 2.5 psig.......... ☐ RO

2. **IF** Loop A RHR will be used,
   THEN:

   a. **Place** E11-CS-S18A (2/3 Core Height LPCI Initiation Override Switch) to MANUAL OVERRD.......... ☐ RO

   b. Momentarily **place** E11-CS-S17A (Containment Spray Valve Control Switch) to MANUAL.......... ☐ RO

   c. **Ensure** one Loop A RHR Pump running.......... ☐ RO

   d. **Ensure** E11-F028A (Torus Discharge Isol Vlv) OPEN.......... ☐ RO

   e. **Open** E11-F027A (Torus Spray Isol Vlv).......... ☐ RO

   f. **Ensure** operation in LPCI, Torus Cooling or Drywell Spray mode.......... ☐ RO

Enclosure 2

Page 2 of 2

2.1.3 Torus Spray Actions (continued)

3. **IF** Loop B RHR will be used,
   THEN:

   a. **Place** E11-CS-S18B (2/3 Core Height LPCI Initiation Override Switch) to MANUAL OVERRD.................. ☐ RO

   b. Momentarily **place** E11-CS-S17B (Containment Spray Valve Control Switch) to MANUAL.................. ☐ RO

   c. **Ensure** one Loop B RHR Pump running.................. ☐ RO

   d. **Ensure** E11-F028B (Torus Discharge Isol Vlv) OPEN.................. ☐ RO

   e. **Open** E11-F027B (Torus Spray Isol Vlv).................. ☐ RO

   f. **Ensure** operation in LPCI, Torus Cooling **OR** Drywell Spray mode.................. ☐ RO

4. **WHEN** torus pressure drops to 2.5 psig **OR** directed to terminate sprays,
   THEN ensure CLOSED:

   - E11-F027A (Torus Spray Isol Vlv).................. ☐ RO
   - E11-F027B (Torus Spray Isol Vlv).................. ☐ RO

5. **IF** re-initiation of sprays required,
   THEN return to Section 2.1.3 Step 1.................. ☐ RO

6. **WHEN** sprays **NO** longer required,
   THEN go to Section 2.2.................. ☐ RO

Enclosure 3

2.1.3 Drywell Spray Actions

1. **Ensure** both reactor recirculation pumps tripped. .... ☐ RO

2. **IF** E-bus load stripping has occurred,
   THEN:

   a. **Confirm** electrical power has been aligned per EOP-01-SBO-14. .... ☐ RO

   b. **Secure** drywell coolers per Attachment 1 and **continue** at Section 2.1.3 Step 2.c. .... ☐ RO

   c. **IF** RHR Loop A will be used for sprays,
      THEN go to Section 2.1.3 Step 9. .... ☐ RO

   d. **IF** RHR Loop B will be used for sprays,
      THEN go to Section 2.1.3 Step 10. .... ☐ RO

3. **Place** all drywell cooler control switches to OFF (L/O). .... ☐ RO

Enclosure 3

Page 2 of 6

2.1.3 Drywell Spray Actions (continued)

4. **Unit 1 Only: IF** drywell coolers continue to run,
   THEN:
   - In Panel XU-27, west side, **place** VA-CS-5993 (D/W Clr A&D Override Switch) in STOP. ☐ RO
   - In Panel XU-28, west side, **place** VA-CS-5994 (D/W Clr B&C Override Switch) in STOP. ☐ RO

5. **Unit 2 Only: IF** drywell coolers continue to run,
   THEN:
   - In Panel XU-27, west side, **place** VA-CS-5993 (D/W Clr A&D Override Switch) in STOP. ☐ RO
   - In Panel XU-28, east side, **place** VA-CS-5994 (D/W Clr B&C Override Switch) in STOP. ☐ RO

6. **IF** drywell coolers continue to run,
   THEN secure drywell coolers per Attachment 1 and **continue** at Section 2.1.3 Step 7. ☐ RO

7. **Ensure** SW-V141 (Well Water to Vital Header Vlv) CLOSED. ☐ RO

8. **Ensure** one valve OPEN:
   - SW-V111 (Conv SW To Vital Header Vlv) ☐ RO
   - SW-V117 (Nuc SW To Vital Header Vlv) ☐ RO

Enclosure 3

2.1.3 Drywell Spray Actions (continued)

9. **IF** Loop A RHR will be used for drywell spray,
THEN:

> **NOTE**
> E11-F017A will remain OPEN for five minutes following a LOCA signal. ☐

a. **IF** E11-F015A (Inboard Injection Vlv) OPEN,
THEN close E11-F017A (Outboard Injection Vlv). ☐ RO

b. Place E11-CS-S18A (2/3 Core Height LPCI Initiation Override Switch) to MANUAL OVERRD. ☐ RO

c. Momentarily place E11-CS-S17A (Containment Spray Valve Control Switch) to MANUAL. ☐ RO

d. Ensure E11-F024A (Torus Cooling Isol Vlv) CLOSED. ☐ RO

e. Ensure one Loop A RHR Pump running. ☐ RO

f. Confirm requirements for Drywell Spray Initiation met:

- Safe region of Drywell Spray Initiation Limit ☐ RO
- Torus level below +21 inches ☐ RO

g. Open E11-F021A (Drywell Spray Inbd Isol Vlv). ☐ RO

h. Throttle open E11-F016A (Drywell Spray Otbd Isol Vlv) to obtain between 8,000 gpm and 10,000 gpm flow. ☐ RO

i. **IF** E-bus load stripping has occurred,
THEN go to Section 2.1.3 Step 11. ☐ RO

Enclosure 3

2.1.3 Drywell Spray Actions (continued)

j. **IF** additional flow required,
THEN start the other RHR pump and limit flow to less than or equal to 11,500 gpm. .... ☐ RO

k. **Ensure** RHRSW Loop A operating:

(1) **Place** E11-S19A (RHR SW Booster Pumps A & C LOCA Override Switch) in MANUAL OVERRD. .... ☐ RO

(2) **Align** RHRSW to the heat exchanger (OP-43). .... ☐ RO

l. **Establish** RHR flow through the heat exchanger:

(1) **Ensure** E11-F047A (Hx A Inlet Vlv) OPEN. .... ☐ RO

(2) **Ensure** E11-F003A (Hx A Outlet Vlv) OPEN. .... ☐ RO

NOTE

E11-F048A will remain OPEN for three minutes following a LOCA signal. .... ☐

(3) **Close** E11-F048A (Hx A Bypass Vlv). .... ☐ RO

10. **IF** Loop B RHR will be used for drywell spray,
THEN:

NOTE

E11-F017B will remain OPEN for five minutes following a LOCA signal. .... ☐

a. **IF** E11-F015B (Inboard Injection Vlv) OPEN,
THEN close E11-F017B (Outboard Injection Vlv). .... ☐ RO

b. **Place** E11-CS-S18B (2/3 Core Height LPCI Initiation Override Switch) to MANUAL OVERRD. .... ☐ RO

Enclosure 3

2.1.3 Drywell Spray Actions (continued)

c. Momentarily **place** E11-CS-S17B (Containment Spray Valve Control Switch) to MANUAL. .... ☐ RO

d. **Ensure** E11-F024B (Torus Cooling Isol Vlv) CLOSED. .... ☐ RO

e. **Ensure** one Loop B RHR Pump running. .... ☐ RO

f. **Confirm** requirements for Drywell Spray Initiation are met:

- Safe region of the Drywell Spray Initiation Limit .... ☐ RO
- Torus level below +21 inches .... ☐ RO

g. **Open** E11-F021B (Drywell Spray Inbd Isol Vlv). .... ☐ RO

h. **Throttle open** E11-F016B (Drywell Spray Otbd Isol Vlv) to obtain between 8,000 gpm and 10,000 gpm flow. .... ☐ RO

i. **<u>IF</u>** E-bus load stripping has occurred,
<u>THEN</u> go to Section 2.1.3 Step 11. .... ☐ RO

j. **<u>IF</u>** additional flow required,
<u>THEN</u> **start** the other RHR pump and **limit** flow to less than or equal to 11,500 gpm. .... ☐ RO

k. **Ensure** RHRSW Loop B operating:

(1) **Place** E11-S19B (RHR SW Booster Pumps B & D LOCA Override Switch) in MANUAL OVERRD. .... ☐ RO

(2) **Align** RHRSW to the heat exchanger (OP-43). .... ☐ RO

Enclosure 3

Page 6 of 6

2.1.3 Drywell Spray Actions (continued)

l. **Establish** RHR flow through the heat exchanger:

(1) **Ensure** E11-F047B (Hx B Inlet Vlv) OPEN. .... ☐ RO

(2) **Ensure** E11-F003B (Hx B Outlet Vlv) OPEN.......................... ☐ RO

NOTE
E11-F048B will remain OPEN for three minutes following a LOCA signal. .... ☐

(3) **Close** E11-F048B (Hx B Bypass Vlv). .... ☐ RO

11. **<u>WHEN</u>** drywell pressure drops to 2.5 psig **<u>OR</u>** directed to terminate drywell spray,
<u>THEN</u> ensure CLOSED:

a. E11-F016A (Drywell Spray Otbd Isol Vlv) .... ☐ RO

b. E11-F021A(Drywell Spray Inbd Isol Vlv). .... ☐ RO

c. E11-F016B (Drywell Spray Otbd Isol Vlv) .... ☐ RO

d. E11-F021B (Drywell Spray Inbd Isol Vlv). .... ☐ RO

12. **Ensure** <u>either</u>:

- RHR operated in LPCI mode .... ☐ RO
- RHR operated in Torus Cooling. .... ☐ RO
- RHR pumps are secured .... ☐ RO

13. **<u>IF</u>** re-initiation of drywell spray required,
<u>THEN</u> return to Section 2.1.3 Step 9. .... ☐ RO

ATTACHMENT 1 - Scenario Quantitative Attribute Assessment

Category	NUREG 1021 Rev. 2 Supp. 1 Req.	Scenario Content
Total Malfunctions	5-8	8
Malfunctions after EOP Entry	1-2	2
Abnormal Events	2-4	3
Major Transients	1-2	2
EOPs Used	1-2	2
EOP Contingency	0-2	2
Run Time	60-90 min	90
Crew Critical Tasks	2-3	2
Tech Specs	2	2
Instrument / Component Failures before Major	2 – RO 2 - BOP	4
Instrument / Component Failures after Major	2	2
Normal Operations	1	1
Reactivity manipulation	1	1

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R
Reference
Use

Neutron Monitoring Spiking Troubleshooting Form

1. Initiator's name Unit Two SRO

2. Check all instruments that are spiking and the associated Unit:

☐ Unit 1	☐ SRM A	☒ IRM A	☐ IRM E
☒ Unit 2	☐ SRM B	☐ IRM B	☐ IRM F
	☐ SRM C	☐ IRM C	☐ IRM G
	☐ SRM D	☐ IRM D	☐ IRM H

3. Time and date of event Today - Previous Shift

4. What is the duration of the spiking (duration of individual spike)? Add additional information below to characterize spiking event.

☐ Seconds ☒ Minutes ☐ Hours

5. Ensure all required observations to support operability are appropriately documented.

6. Has a WO or AR been initiated? If yes, list number(s): 00345765	☒ Yes	☐ No
7. Has a log entry been made?	☒ Yes	☐ No
8. Is there any welding occurring in the plant?	☒ Yes	☐ No
9. Are there any personnel under-vessel?	☐ Yes	☒ No
10. Are there any plant evolutions in progress?	☒ Yes	☐ No
11. Is there any electrical switching occurring?	☐ Yes	☒ No
12. Are any control rods being moved or selected?	☒ Yes	☐ No
13. Has there been a recent change in the mode switch?	☐ Yes	☒ No
14. Is there any major equipment being started?	☒ Yes	☐ No
15. Has there been any observed relay chatter?	☐ Yes	☒ No
16. Is there any refuel bridge movement?	☐ Yes	☒ No
17. Are the rod interlocks being affected?	☒ Yes	☐ No
18. Completed copy of this attachment sent to engineer	☒ Yes	☐ No

Please note below any additional information that may aid troubleshooting (such as 2 instruments spiking but not in the same manner):

Multiple upscale and downscale alarms during startup over a 15 minute period.
All other IRMs responded normally.

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ATTACHMENT 2 – Shift Turnover

Brunswick Unit 2 Plant Status					
Station Duty Manager:				Workweek Manager:	
Mode:	2	Rx Power:	2%	Gross*/Net MWe*:	NA
Plant Risk: Current EOOS Risk Assessment is:				Green	
SFP Time to 200 Deg F:	65 hrs			Days Online:	0 days
Turnover:	Complete Step 6.3.46 of 0GP-02, Approach to Criticality and Pressurization of the Reactor. Raise power to 6-10%. A2X sequence at step 166. Permission for continuous withdrawal has been granted for rods going from 12-48.				
Protected Equipment:	ADHR / FPC Loop A / Demin Transfer Pump				
Comments:	IRM A was bypassed due to spiking and the paperwork is being evaluated for its return to service.				

Facility Brunswick Date of Exam: December 2015

Tier	Group	RO K/A Category Points												SRO-Only Points		
		K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G*	Total	A2	G*	Total
1. Emergency & Abnormal Plant Evolutions	1	4	3	4	N/A			3	3	N/A		3	20	4	3	7
	2	2	1	1				1	1			1	7	2	1	3
	Tier Totals	6	4	5				4	4			4	27	6	4	10
2. Plant Systems	1	2	3	3	2	3	1	2	3	3	2	2	26	3	2	5
	2	1	2	1	1	1	1	1	1	1	1	1	12	0 2	1	3
	Tier Totals	3	5	4	3	4	2	3	4	4	3	3	38	5	3	8

3. Generic Knowledge and Abilities Categories	1	2	3	4	Total	1	2	3	4	Total
	2	3	3	2	10	2	2	1	2	7

1. Ensure that at least two topics from every applicable K/A category are sampled within each tier of the RO and SRO-only outlines (i.e., except for one category in Tier 3 of the SRO-only outline, the ATier Totals@ in each K/A category shall not be less than two).
2. The point total for each group and tier in the proposed outline must match that specified in the table. The final point total for each group and tier may deviate by ∀1 from that specified in the table based on NRC revisions. The final RO exam must total 75 points and the SRO-only exam must total 25 points.
3. Systems/evolutions within each group are identified on the associated outline; systems or evolutions that do not apply at the facility should be deleted and justified; operationally important, site-specific systems that are not included on the outline should be added. Refer to ES-401, Attachment 2, for guidance regarding the elimination of inappropriate K/A statements.
4. Select topics from as many systems and evolutions as possible; sample every system or evolution in the group before selecting a second topic for any system or evolution.
5. Absent a plant-specific priority, only those K/As having an importance rating (IR) of 2.5 or higher shall be selected. Use the RO and SRO ratings for the RO and SRO-only portions, respectively.
6. Select SRO topics for Tiers 1 and 2 from the shaded systems and K/A categories.
7. *The generic (G) K/As in Tiers 1 and 2 shall be selected from Section 2 of the K/A Catalog, but the topics must be relevant to the applicable evolution or system.
8. On the following pages, enter the K/A numbers, a brief description of each topic, the topics= importance ratings (IRs) for the applicable license level, and the point totals (#) for each system and category. Enter the group and tier totals for each category in the table above; if fuel handling equipment is sampled in other than Category A2 or G* on the SRO-only exam, enter it on the left side of Column A2 for Tier 2, Group 2 (Note # 1 does not apply). Use duplicate pages for RO and SRO-only exams.
9. For Tier 3, select topics from Section 2 of the K/A catalog, and enter the K/A numbers, descriptions, IRs, and point totals (#) on Form ES-401-3. Limit SRO selections to K/As that are linked to 10 CFR 55.43.

KA	NAME / SAFETY FUNCTION:	IR RO	IR SRO	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	TOPIC:
295001AK1.04	Partial or Complete Loss of Forced Core Flow Circulation / 1 & 4	2.5	3.3	✔											Limiting cycle oscillation: Plant-Specific...........
295003G2.4.50	Partial or Complete Loss of AC / 6	4.2	4.0											✔	Ability to verify system alarm setpoints and operate controls identified in the alarm response manual.
295004AK3.02	Partial or Total Loss of DC Pwr / 6	2.9	3.3			✔									Ground isolation/fault determination..................
295005AA1.07	Main Turbine Generator Trip / 3	3.3	3.3							✔					A.C. electrical distribution..........................
295006AA2.02	SCRAM / 1	4.3	4.4								✔				Control rod position.................................
295016AK2.01	Control Room Abandonment / 7	4.4	4.5		✔										Remote shutdown panel: Plant-Specific................
295018AK3.02	Partial or Total Loss of CCW / 8	3.3	3.4			✔									Reactor power reduction..............................
295019AA1.04	Partial or Total Loss of Inst. Air / 8	3.3	3.2							✔					Service air isolations valves: Plant-Specific........
295021AK3.02	Loss of Shutdown Cooling / 4	3.3	3.4			✔									Feeding and bleeding reactor vessel..................
295023AA2.01	Refueling Acc Cooling Mode / 8	3.6	4.0								✔				Area radiation levels................................
295024G2.4.50	High Drywell Pressure / 5	4.2	4.0											✔	Ability to verify system alarm setpoints and operate controls identified in the alarm response manual.

KA	NAME / SAFETY FUNCTION:	IR RO	IR SRO	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	TOPIC:
295025EK2.05	High Reactor Pressure / 3	4.1	4.2		✔										Safety/relief valves: Plant-Specific................
295026EK1.02	Suppression Pool High Water Temp. / 5	3.5	3.8	✔											Steam condensation..................................
295028EK1.01	High Drywell Temperature / 5	3.5	3.7	✔											Reactor water level measurement......................
295030EK1.02	Low Suppression Pool Wtr Lvl / 5	3.5	3.8	✔											Pump NPSH...
295031G2.2.37	Reactor Low Water Level / 2	3.6	4.6											✔	Ability to determine operability and/or availability of safety related equipment
295037EK3.07	SCRAM Condition Present and Power Above APRM Downscale or Unknown / 1	4.2	4.3			✔									Various alternate methods of control rod insertion: Plant-Specific......................................
295038EA2.02	High Off-site Release Rate / 9	2.5	3.3								✔				Total number of curies released........................
600000AK2.01	Plant Fire On Site / 8	2.6	2.7		✔										Sensors / detectors and valves
700000AA1.02	Generator Voltage and Electric Grid Distrurbancecs	3.8	3.7							✔					Turbine / generator controls

KA	NAME / SAFETY FUNCTION:	IR RO	IR SRO	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	TOPIC:
295007G2.4.11	High Reactor Pressure / 3	4.0	4.2											✔	Knowledge of abnormal condition procedures.
295008AK1.02	High Reactor Water Level / 2	2.8	2.8	✔											Component erosion/damage.............................
295010AK2.02	High Drywell Pressure / 5	3.3	3.5		✔										Drywell/suppression chamber differential pressure: Mark I&II
295014AA2.04	Inadvertent Reactivity Addition / 1	4.1	4.4								✔				Violation of fuel thermal limits......................
295017AK1.03	High Off-site Release Rate / 9	2.7	3.4	✔											Meteorological effects on off-site release............
295020AK3.02	Inadvertent Cont. Isolation / 5 & 7	3.3	3.5			✔									Drywell/containment pressure response..................
500000EA1.02	High CTMT Hydrogen Conc. / 5	3.3	3.2							✔					Primary containment oxygen instrumentation

KA	NAME / SAFETY FUNCTION:	IR RO	IR SRO	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	TOPIC:
203000A4.06	RHR/LPCI: Injection Mode	3.9	3.9										✔		System reset following automatic initiation: Plant-Specific
205000K3.02	Shutdown Cooling	3.2	3.3			✔									Reactor water level: Plant-Specific
205000K5.03	Shutdown Cooling	2.8	3.1					✔							Heat removal mechanisms
206000G2.2.36	HPCI	3.1	4.2											✔	Ability to analyze the effect of maintenance activities, such as degraded power sources, on the status of limiting conditions of operations
209001A3.03	LPCS	3.5	3.5									✔			System pressure
211000G2.4.9	SLC	3.8	4.2											✔	Knowledge of low power / shutdown implications in accident (e.g. LOCA or loss of RHR) mitigation strategies.
212000K1.03	RPS	3.4	3.6	✔											Recirculation system
215003K2.01	IRM	2.5	2.7		✔										IRM channels/detectors
215004K5.01	Source Range Monitor	2.6	2.6					✔							Detector operation
215005K1.10	APRM / LPRM	3.3	3.3	✔											Reactor manual control system: Plant-Specific
217000K6.01	RCIC	3.4	3.5						✔						Electrical power

KA	NAME / SAFETY FUNCTION:	IR RO	IR SRO	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	TOPIC:
218000A3.03	ADS	3.7	3.8									✔			ADS valve acoustical monitor noise: Plant-Specific
218000K2.01	ADS	3.1	3.3		✔										ADS logic
223002A2.09	PCIS/Nuclear Steam Supply Shutoff	3.6	3.7								✔				System initiation
223002A4.05	PCIS/Nuclear Steam Supply Shutoff	2.5	2.8										✔		SPDS/ERIS/CRIDS/GDS: Plant-Specific
239002K4.04	SRVs	3.4	3.6				✔								Ensures even distribution of heat load to suppression pool,and adequate steam condensing
259002A2.06	Reactor Water Level Control	3.3	3.4								✔				Loss of controller signal output
261000K3.04	SGTS	3.1	3.1			✔									High pressure coolant injection system: Plant- Specific
262001A1.05	AC Electrical Distribution	3.2	3.5							✔					Breaker lineups
262002A2.01	UPS (AC/DC)	2.6	2.8								✔				Under voltage
262002A3.01	UPS (AC/DC)	2.8	3.1									✔			Transfer from preferred to alternate source
263000A1.01	DC Electrical Distribution	2.5	2.8							✔					Battery charging/discharging rate

KA	NAME / SAFETY FUNCTION:	IR RO	IR SRO	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	TOPIC:
264000K3.03	EDGs	4.1	4.2	☐	☐	☑	☐	☐	☐	☐	☐	☐	☐	☐	Major loads powered from electrical buses fed by the emergency generator(s)
264000K5.05	EDGs	3.4	3.4	☐	☐	☐	☐	☑	☐	☐	☐	☐	☐	☐	Paralleling A.C. power sources
300000K2.01	Instrument Air	2.8	2.8	☐	☑	☐	☐	☐	☐	☐	☐	☐	☐	☐	Instrument air compressor
400000K4.01	Component Cooling Water	3.4	3.9	☐	☐	☐	☑	☐	☐	☐	☐	☐	☐	☐	Automatic start of standby pump

KA	NAME / SAFETY FUNCTION:	IR RO	IR SRO	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	TOPIC:
201001K4.03	CRD Hydraulic	2.7	2.7	☐	☐	☐	☑	☐	☐	☐	☐	☐	☐	☐	Control rod drive mechanism cooling water flow
201002A1.04	RMCS	3.6	3.5	☐	☐	☐	☐	☐	☐	☑	☐	☐	☐	☐	Overall reactor power
201003A2.01	Control Rod and Drive Mechanism	3.4	3.6	☐	☐	☐	☐	☐	☐	☐	☑	☐	☐	☐	Stuck rod
204000A4.01	RWCU	3.1	3.0	☐	☐	☐	☐	☐	☐	☐	☐	☐	☑	☐	System pumps
215002K2.03	RBM	2.8	2.9	☐	☑	☐	☐	☐	☐	☐	☐	☐	☐	☐	APRM channels: BWR-3,4,5
219000G2.2.37	RHR/LPCI: Torus/Pool Cooling Mode	3.6	4.6	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	☑	Ability to determine operability and/or availability of safety related equipment
233000K2.02	Fuel Pool Cooling/Cleanup	2.8	2.9	☐	☑	☐	☐	☐	☐	☐	☐	☐	☐	☐	RHR pumps
241000K6.10	Reactor/Turbine Pressure Regulator	3.6	3.7	☐	☐	☐	☐	☐	☑	☐	☐	☐	☐	☐	Bypass valves
256000A3.07	Reactor Condensate	2.9	2.9	☐	☐	☐	☐	☐	☐	☐	☐	☑	☐	☐	Feedwater heater level
271000K1.03	Offgas	2.7	3.0	☑	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	Elevated release point
290002K5.05	Reactor Vessel Internals	3.1	3.3	☐	☐	☐	☐	☑	☐	☐	☐	☐	☐	☐	Brittle fracture

KA	NAME / SAFETY FUNCTION:	IR RO	IR SRO	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	TOPIC:
290003K3.02	Control Room HVAC	3.3	3.6	☐	☐	☑	☐	☐	☐	☐	☐	☐	☐	☐	Computer/instrumentation: Plant-Specific

KA	NAME / SAFETY FUNCTION:	IR RO	IR SRO	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	TOPIC:
G2.1.30	Conduct of operations	4.4	4.0	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	☑	Ability to locate and operate components, including local controls.
G2.1.42	Conduct of operations	2.5	3.4	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	☑	Knowledge of new and spent fuel movement procedures
G2.2.25	Equipment Control	3.2	4.2	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	☑	Knowledge of the bases in Technical Specifications for limiting conditions for operations and safety limits.
G2.2.40	Equipment Control	3.4	4.7	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	☑	Ability to apply technical specifications for a system.
G2.2.43	Equipment Control	3.0	3.3	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	☑	Knowledge of the process used to track inoperable alarms
G2.3.11	Radiation Control	3.8	4.3	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	☑	Ability to control radiation releases.
G2.3.15	Radiation Control	2.9	3.1	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	☑	Knowledge of radiation monitoring systems
G2.3.7	Radiation Control	3.5	3.6	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	☑	Ability to comply with radiation work permit requirements during normal or abnormal conditions
G2.4.17	Emergency Procedures/Plans	3.9	4.3	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	☑	Knowledge of EOP terms and definitions.
G2.4.42	Emergency Procedures/Plans	2.6	3.8	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	☑	Knowledge of emergency response facilities.

KA	NAME / SAFETY FUNCTION:	IR RO	IR SRO	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	TOPIC:
295003AA2.04	Partial or Complete Loss of AC / 6	3.5	3.7	☐	☐	☐	☐	☐	☐	☐	☑	☐	☐	☐	System lineups.......................................
295016G2.1.7	Control Room Abandonment / 7	4.4	4.7	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	☑	Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior and instrument interpretation.
295018G2.1.23	Partial or Total Loss of CCW / 8	4.3	4.4	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	☑	Ability to perform specific system and integrated plant procedures during all modes of plant operation.
295021AA2.06	Loss of Shutdown Cooling / 4	3.2	3.3	☐	☐	☐	☐	☐	☐	☐	☑	☐	☐	☐	Reactor pressure ....
295023G2.4.11	Refueling Acc Cooling Mode / 8	4.0	4.2	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	☑	Knowledge of abnormal condition procedures.
295025EA2.05	High Reactor Pressure / 3	3.4	3.6	☐	☐	☐	☐	☐	☐	☐	☑	☐	☐	☐	Decay heat generation................................
295037EA2.07	SCRAM Condition Present and Power Above APRM Downscale or Unknown / 1	4.0	4.2	☐	☐	☐	☐	☐	☐	☐	☑	☐	☐	☐	Containment conditions/isolations....................

KA	NAME / SAFETY FUNCTION:	IR RO	IR SRO	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	TOPIC:
295015G2.4.5	Incomplete SCRAM / 1	3.7	4.3	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	☑	Knowledge of the organization of the operating procedures network for normal, abnormal, and emergency evolutions.
295022AA2.01	Loss of CRD Pumps / 1	3.5	3.6	☐	☐	☐	☐	☐	☐	☐	☑	☐	☐	☐	Accumulator pressure.................................
295034EA2.02	Secondary Containment Ventilation High Radiation / 9	3.7	4.2	☐	☐	☐	☐	☐	☐	☐	☑	☐	☐	☐	Cause of high radiation levels.......................

KA	NAME / SAFETY FUNCTION:	IR RO	IR SRO	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	TOPIC:
203000G2.4.6	RHR/LPCI: Injection Mode	3.8	4.5											✔	Knowledge of EOP mitigation strategies.
209001A2.07	LPCS	2.6	2.8								✔				Loss of room cooling
212000A2.06	RPS	4.1	4.2								✔				High reactor power
217000G2.4.47	RCIC	4.2	4.2											✔	Ability to diagnose and recognize trends in an accurate and timely manner utilizing the appropriate control room reference material.
261000A2.05	SGTS	3.0	3.1								✔				Fan trips

KA	NAME / SAFETY FUNCTION:	IR RO	IR SRO	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	TOPIC:
219000G2.1.25	RHR/LPCI: Torus/Pool Cooling Mode	3.9	4.2	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	☑	Ability to interpret reference materials such as graphs, monographs and tables which contain performance data.
256000A2.07	Reactor Condensate	2.9	2.9	☐	☐	☐	☐	☐	☐	☐	☑	☐	☐	☐	High hotwell level
290002A2.04	Reactor Vessel Internals	3.7	4.1	☐	☐	☐	☐	☐	☐	☐	☑	☐	☐	☐	Excessive heatup/cooldown rate

KA	NAME / SAFETY FUNCTION:	IR RO	IR SRO	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	TOPIC:
G2.1.26	Conduct of operations	3.4	3.6	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	☑	Knowledge of industrial safety procedures (such as rotating equipment, electrical, high temperature, high pressure, caustic, chlorine, oxygen and hydrogen).
G2.1.38	Conduct of operations	3.7	3.8	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	☑	Knowledge of the stations requirements for verbal communication when implamenting procedures
G2.2.17	Equipment Control	2.6	3.8	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	☑	Knowledge of the process for managing maintenance activities during power operations.
G2.2.22	Equipment Control	4.0	4.7	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	☑	Knowledge of limiting conditions for operations and safety limits.
G2.3.6	Radiation Control	2.0	3.8	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	☑	Ability to aprove release permits
G2.4.13	Emergency Procedures/Plans	4.0	4.6	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	☑	Knowledge of crew roles and responsibilities during EOP usage.
G2.4.46	Emergency Procedures/Plans	4.2	4.2	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	☑	Ability to verify that the alarms are consistent with the plant conditions.

1. 201001 1
Which one of the following completes the statement below?

On a loss of air, the C11-F002A(B), CRD Flow Control Valve, will fail ___(1)___, causing ___(2)___ cooling water to the CRD Mechanism.

A. (1) open
(2) minimum

B. (1) open
(2) maximum

C. (1) closed
(2) minimum

D. (1) closed
(2) maximum

Answer: C

K/A:
201001 Control Rod Drive Hydraulic System
K4 Knowledge of CONTROL ROD DRIVE HYDRAULIC SYSTEM design feature(s) and/or interlocks which provide for the following: (CFR 41.7)
03 Control rod drive mechanism cooling water flow

RO/SRO Rating: 2.7/2.7

Pedigree: New

Objective: LOI-CLS-008, Objective 8e
Given plant conditions, predict the effect that a loss or malfunction of the following will have on the CRDH System: Instrument Air

Reference: None

Cog Level: High

Explanation: The Flow Control Valve is air to open, spring to close. With a loss of air to the flow controller, the CRD Flow Control Valve will fail closed robbing cooling water flow from the CRD mechanism. If cooling water flow is not restored, the mechanism will overheat.

Distractor Analysis:

Choice A: Plausible because if the FCV was an air to close, spring to open valve, which is feasible, then part 1 would be correct. An example is the Scram Valves which will open on loss of air. Student must know where cooling water taps off to the CRD System to answer part 2. If upstream of the FCV, then cooling water to the CRDM would be reduced and make part 2 correct.

Choice B: Plausible because if the FCV was an air to close, spring to open valve, which is feasible, then part 1 would be correct, which would make part 2 also correct.

Choice C: Correct Answer, see explanation.

Choice D: Plausible because part 1 is correct. Student must know where cooling water taps off to the CRD System to answer part 2. If upstream of the FCV, then cooling water to the CRDM would be maximized and make this combination correct. The drive water pressure control valve is an example of closing the valve to raise pressure and flow.

SRO Basis: N/A

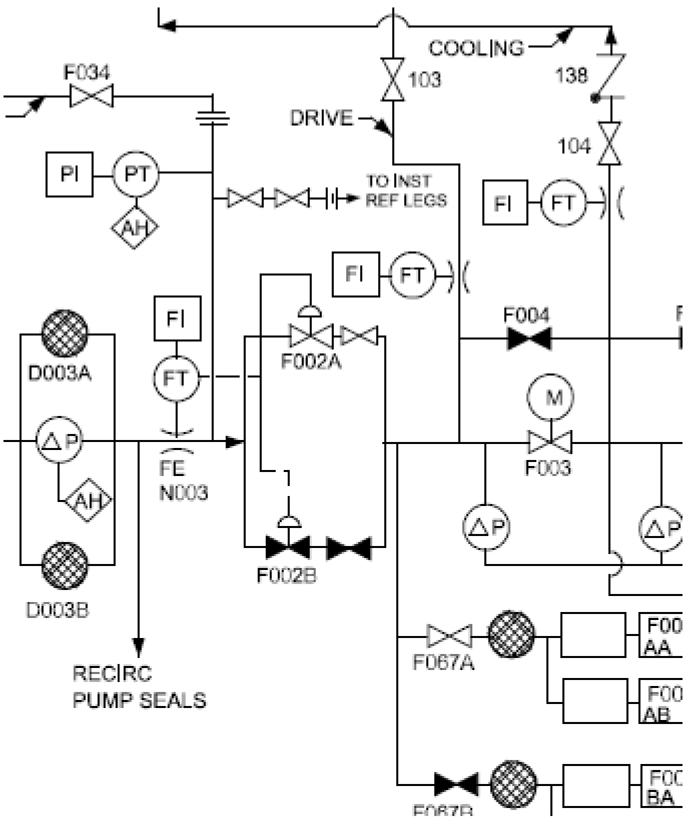
SD-8:

4.6.5 **Loss of Instrument Air**

Interruptible Instrument Air supplies air to the condensate pressure control valve, CO-PCV-4105, supplying condensate to the CRD pump suction. Loss of this air supply will cause the PCV to close. CRD suction will then be supplied from the CST.

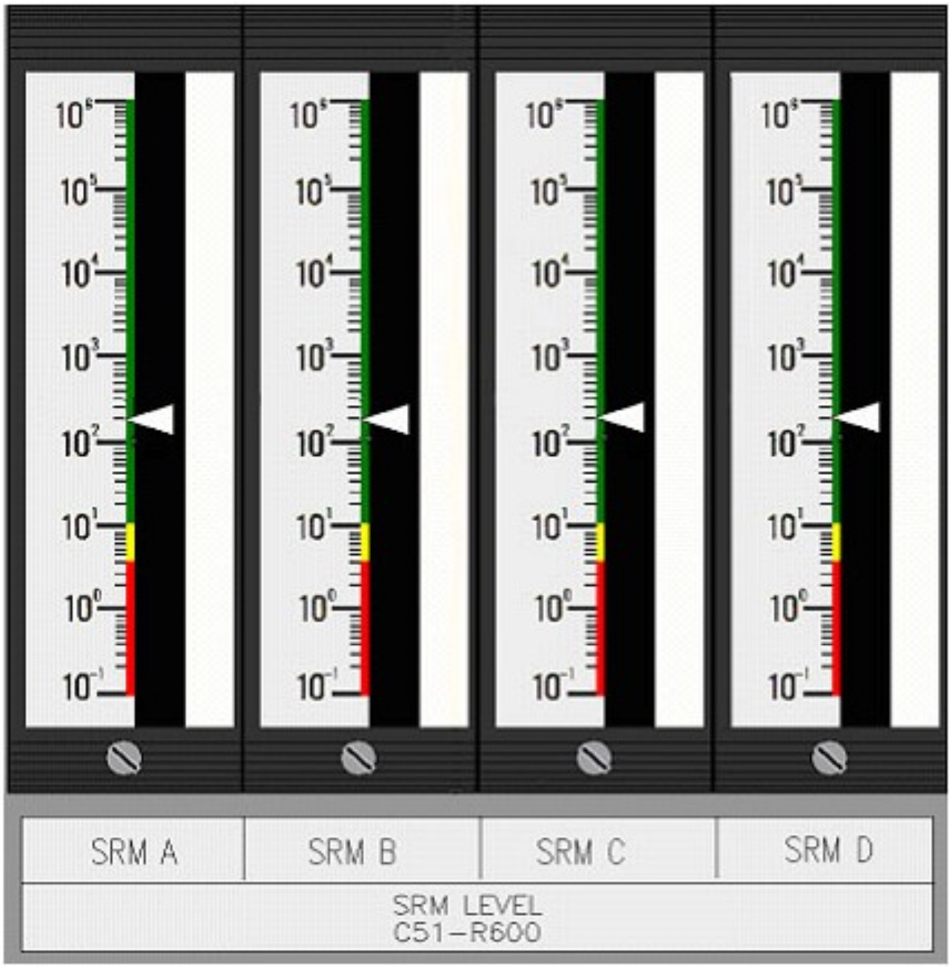
Non-Interruptible Instrument Air supplies positioning and control air to the flow control valves. Loss of the air supply will result in the in-service flow control valve closing. Normal rod movement cannot be performed in this condition due to the loss of drive pressure. This condition would also cause CRD temperatures to increase.

Non-Interruptible Instrument Air supplies the Scram Air Header. Loss of the air supply will result in the scram valves opening and the SDV vent and drain valves closing (i.e., a scram condition). Scram valves may begin opening if pressure approaches 40 psig, causing control rods to start drifting into the core.



2. 201002 1

The initial SRM count rates are as observed below.



The Unit Two control room staff is ready to withdraw control rods for a reactor startup.

Which one of the following identifies when criticality is expected to be achieved IAW 0GP-02, Approach To Criticality and Pressurization of the Reactor?

A. At ~800 cpm

B. At ~1000 cpm

C. At ~3200 cpm

D. At ~6400 cpm

Answer: D

K/A:
201002 Reactor Manual Control System
A1 Ability to predict and/or monitor changes in parameters associated with operating the REACTOR MANUAL CONTROL SYSTEM controls including: (CFR: 41.5 / 45.5)
04 Overall reactor power

RO/SRO Rating: 3.6/3.5

Pedigree: 2014 NRC Exam

Objective: LOI-CLS-LP-307-A, Objective B6
GP-02, Approach to Criticality and Pressurization of the Reactor: List the indications that the reactor is critical in accordance with GP-02.

Reference: None

Cog Level: High

Explanation:
As a rule of thumb, five "doubles" in the neutron count rate will yield criticality.
Initial count rate 200 cpm
1st double = 400 cpm
2nd double = 800 cpm
3rd double = 1600 cpm
4th double = 3200 cpm
5th double = 6400 cpm

Distractor Analysis:

Choice A: Plausible because a common error is to count the initial readings as one of the doubling values with that logic this would be three doublings which is when single notching of control rods is required as the operators approach criticality.

Choice B: Plausible because this value is the current reading times 5.

Choice C: Plausible because a common error is to count the initial readings as one of the doubling values with that logic this would be five doublings.

Choice D: Correct Answer, see explanation

SRO Basis: N/A

NOTE:	When performing 'doubling' calculations, start with the values indicated on the SRM s. Example: If the indicated SRM value is 100 cpm then (2 X 100 =200) 200 cpm is the FIRST 'doubling', then 800 cpm is the third 'doubling' recorded in Step 5.2.7.1, and 3200 cpm is the fifth 'doubling' value to be recorded in Step 5.2.7.2.

0GP-02	Rev. 106	Page 9 of 54

NOTE:	As a rule of thumb, five "doubles" in the neutron count rate will yield criticality; however, this rule may not always hold true due to initial core conditions and time between control rod withdrawals.

3. 201003 1

During a Reactor startup on Unit Two, a control rod is stuck at position 24.

Which one of the following completes the statements below?

Shutdown Margin is ___(1)___.

IAW 2OP-07, Reactor Manual Control System Operating Procedure, Drive Header DP is raised by throttling ___(2)___ C12-PCV-F003, Drive Pressure Valve.

A. (1) maintained, provided all other control rods insert to position 00 on a scram,
   (2) open

B. (1) maintained, provided all other control rods insert to position 00 on a scram,
   (2) closed

C. (1) NOT maintained, even if all other control rods insert to position 00 on a scram,
   (2) open

D. (1) NOT maintained, even if all other control rods insert to position 00 on a scram,
   (2) closed

Answer: B

K/A:
201003 Control Rod and Drive Mechanism
A1 Ability to (a) predict the impacts of the following on the CONTROL ROD AND DRIVE MECHANISM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: (CFR: 41.5 / 45.6)
01 Stuck rod

RO/SRO Rating: 3.4/3.6

Pedigree: New

Objective: LOI-CLS-LP-302-B, Objective 04
Given plant conditions, determine the required supplementary actions IAW 2OP-02.0, Control Rod Malfunction/Misposition.

Reference: None

Cog Level: Fund

Explanation: Shutdown Margin definition assumes rod of single highest worth is full out, therefore shutdown margin maintained provided all other rods scram. Drive Header dp is raised by throttling closed the pressure control valve.

Distractor Analysis:

Choice A: Plausible because part 1 is correct. Opening the PCV would lower dp because it is a back pressure control valve. This would be correct for cooling water pressure.

Choice B: Correct Answer, see explanation.

Choice C: Plausible because trainee must know the definition of SDM for part 1. Opening the PCV would lower dp because it is a back pressure control valve. This would be correct for cooling water pressure.

Choice D: Plausible because trained must know the definition of SDM for part 1. Part 2 is correct.

SRO Basis: N/A
Shutdown Margin definition assumes rod of single highest worth is full out, therefore shutdown margin maintained provided all other rods scram.

CAUTION

Pressure on the C12-PI-R006 (Drive Header Gauge) is **NOT** to exceed 1450 psig. ☐

b. **Raise** CRD drive water pressure up to 450 psid drive header DP without exceeding 1450 psig drive header pressure limit using the above calculation. ☐

c. Momentarily **place** Emergency Rod In Notch Override in EMERGENCY ROD IN. ☐

d. Simultaneously **place** Emergency Rod In Notch Override in OVERRIDE and Rod Movement in NOTCH OUT. ☐

14. **Lower** control rod drive pressure to between 260 and 275 psid drive header DP. ☐

4. 203000 1
Which one of the following identifies the correct sequence for resetting a Core Spray initiation signal IAW 1OP-17, Residual Heat Removal System Operating Procedure?

A. Reset both Divisions of Core Spray logic, then reset both Divisions of LPCI logic within 10 seconds.

B. Reset both Divisions of LPCI logic, then reset both Divisions of Core Spray logic within 10 seconds.

C. Reset Division I Core Spray Logic then Division I LPCI Logic within 10 seconds, and then reset Division II Core Spray Logic then Division II LPCI Logic within 10 seconds.

D. Reset Division I LPCI Logic then Division I Core Spray Logic within 10 seconds, and then reset Division II LPCI Logic then Division II Core Spray Logic within 10 seconds.

Answer: A

K/A:
203000 RHR/LPCI: Injection Mode
A4 Ability to manually operate and/or monitor in the control room: (CFR: 41.7 / 45.5 to 45.8)
06 System reset following automatic initiation: Plant-Specific

RO/SRO Rating: 3.9/3.9

Pedigree: Bank

Objective: LOI-CLS-LP-017, Objective 13
Given plant conditions, determine the operator actions required to reset a LPCI initiation signal.

Reference: None

Cog Level: Fund

Explanation: The sequence of resetting Core Spray and RHR logics is fundamental to the logic. See logic prints in Notes Section.

Distractor Analysis:

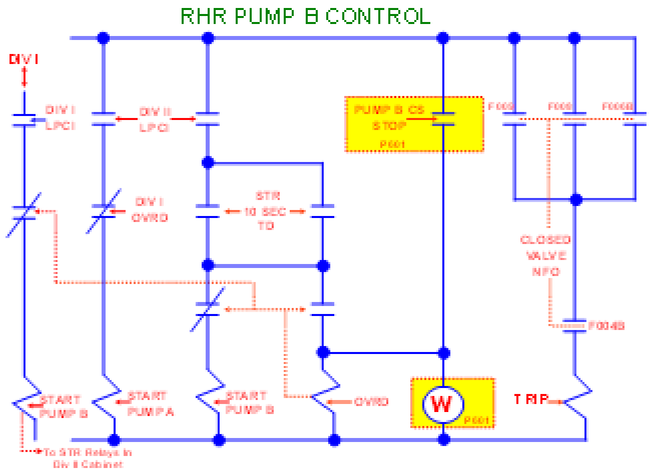
Choice A: Correct Answer, see explanation.

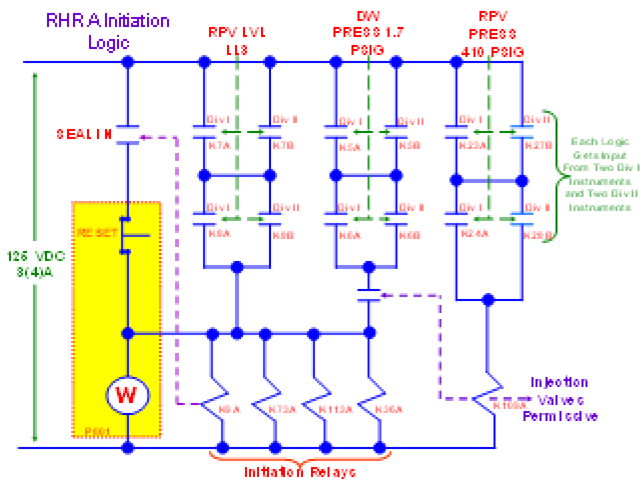
Choice B: Plausible because it is opposite of correct answer, which is feasible, but not correct according to the logics.

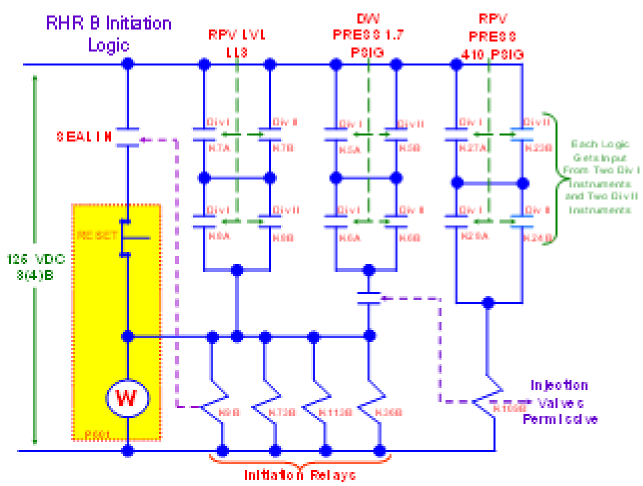
Choice C: Plausible because without knowing that the Core Spray logics feeds the opposite loop of RHR, this would appear feasible.

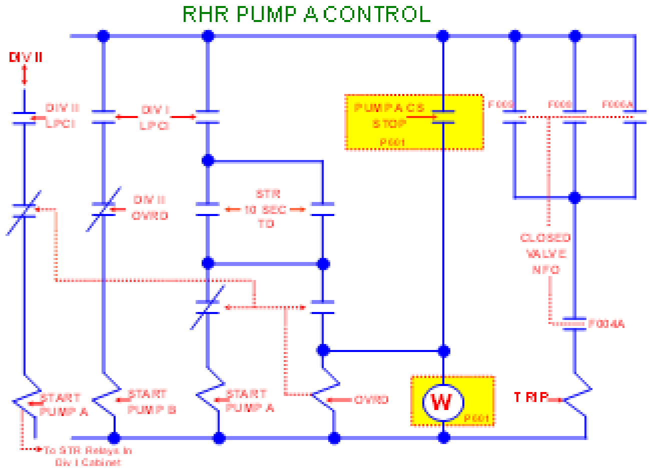
Choice D: Plausible because it is a combination of Choice B and C.

SRO Basis: N/A


RHR PUMP B CONTROL
DIV I
DIV I
LPCI
DIV II
LPCI
PUMP B CS
STOP
P601
DIV I
OVRD
STR
10 SEC
TD
CLOSED
VALVE
INFO
START
PUMP B
START
PUMP A
START
PUMP B
OVRD
W
P601
TRIP
To STR Relays in
Div II Cabinet


RHR A Initiation
Logic
RPV LVL
LL3
DW
PRESS 1.7
PSIG
RPV
PRESS
410 PSIG
SEAL IN
RESET
125 VDC
3(4)A
W
P601
Injection
Valves
Permissive
Each Logic
Gets Input
From Two Div I
Instruments
and Two Div II
Instruments
Initiation Relays


RHR B Initiation
Logic
RPV LVL
LL3
DW
PRESS 1.7
PSIG
RPV
PRESS
410 PSIG
SEAL IN
RESET
125 VDC
3(4)B
W
P601
Injection
Valves
Permissive
Each Logic
Gets Input
From Two Div I
Instruments
and Two Div II
Instruments
Initiation Relays



NOTE:	Steps 7.1.2.2.a **AND** 7.1.2.2.b to be **EXECUTED** within 10 seconds of each other.

a. **DEPRESS** *CORE SPRAY INITIATION SIGNAL/RESET, E21-CS-15A* **AND** *CORE SPRAY INITIATION SIGNAL/RESET, E21-CS-15B*, push buttons **AND CONFIRM** both white *INITIATION SIGNAL SEALED-IN* lights go out. ☐

7.1.2 Procedural Steps

b. **DEPRESS** *LOOP A LPCI INITIATION SIGNAL RESET, E11-CS-S62A*, **AND** *LOOP B LPCI INITIATION SIGNAL RESET, E11-CS-S62B*, push buttons, **AND CONFIRM** both white *INITIATION SIGNAL SEALED-IN* lights go out ☐

5. 204000 1

Following a reactor scram on Unit One, a reject flow path had been established to control reactor water level.

Subsequently, the following conditions exist:

Reactor water level	198 inches
RWCU differential flow	35 gpm
RWCU System discharge pressure	130 psig
RWCU room temperature	125ºF
RWCU system flow	80 gpm
RWCU Pump cooling water temp	145ºF

Based on these conditions, which one of the following identifies the status of the RWCU System?

The RWCU Pump(s) will ___(1)___.

The 1-G31-F033, RWCU Reject Flow Control Valve, will ___(2)___.

A. (1) trip
(2) close

B. (1) trip
(2) remain open

C. (1) continue to run
(2) close

D. (1) continue to run
(2) remain open

Answer: B

K/A:
204000 Reactor Water Cleanup System
A4 Ability to manually operate and/or monitor in the control room: (CFR: 41.7 / 45.5 to 45.8)
01 System Pumps

RO/SRO Rating: 3.1/3.0

Pedigree: New

Objective: LOI-CLS-LP-014, Objective 7
Given plant conditions, determine if the RWCU pump(s) should TRIP.

Reference: None

Cog Level: Higher

Explanation: A RWCU pump trip signal is high Pump Cooling Water (RBCCW) Temperature 140^{o} F. The F033 will close at a RWCU discharge pressure of 140 psig increasing or 5 psig decreasing.

Distractor Analysis:

Choice A: Plausible because the pump would be tripped due to high cooling water temperature. The F033 would close at a system pressure of 140 psig.

Choice B: Correct Answer, see explanation

Choice C: Plausible because the only trip signal is RWCU Pump cooling water temperature. The F033 will close at a RWCU System discharge pressure of 140 psig.

Choice D: Plausible because the only trip signal is RWCU Pump cooling water temperature. The F033 would be open.

SRO Basis: N/A

TABLE 14-3
Page 1 of 7
Instrument and Control Setpoints - Reactor Water Cleanup System

INSTRUMENT TRIP FUNCTION	INSTRUMENT DESIGNATION	INDICATOR/ RECORDER	TRIP SETPOINT AND FUNCTION	
Filter/Demineralizer Influent Conductivity	RXS-CE-N009	RXS-CRS-R601	1.0 µmhos/cm ± .01% of reading Pnl. A-4. Increasing	-Cleanup Sys Hi Conductivity Alarm on Ann.
Filter/Demineralizer A or B Effluent Conductivity	RXS-CE-N010A or RXS-CE-N010B	RXS-CRS-R601	0.1 µmhos/cm ± .01% of reading Pnl. A-4. Increasing	-Cleanup Sys Hi Conductivity Alarm on Ann.
RWCU System High Differential Flow	B21-XY-5949B *	G31-FDI-R615 B21-XY-5949B	40 gpm increasing	-"Cleanup Leak Hi" annunciator (A-04 4-4) Inverse video "RWCU ΔF HI" on B21-XY-5949B
RWCU System High Differential Flow		G31-FDI-R615 B21-XY-5949B	43 gpm-"Cleanup Leak HI-HI/Isol increasing	Timer Start" Annunciator (A-04 5-4) -Initiates RWCU system high differential flow time delay

TABLE 14-3
Page 3 of 7
Instrument and Control Setpoints - Reactor Water Cleanup System

INSTRUMENT TRIP FUNCTION	INSTRUMENT DESIGNATION	INDICATOR/ RECORDER	TRIP SETPOINT	AND FUNCTION
Precoat Tank Level	G31-Z002-LS-75A	-	6" from top of tank ± 1.0" Increasing	-Cleanup Filt Demin Failure Alarm on Ann. Pnl. A-4. -Alarm on Pnl. G31-Z002-25.
	and G31-Z002-LS-75B	-	6" from bottom of tank ± 1.0" Decreasing	-Cleanup Filt Demin Failure Alarm on Ann. Pnl. A-4. -Alarm on Pnl. G31-Z002-25.
RWCU System Pumps Flow	G31-PDIS-N025	-	60 gpm ± 5 gpm Decreasing	-Cleanup Pmps Flow Low Alarm on Ann. Pnl. A-4. -Trips both RWCU pumps
Filter/Demineralizer A or B Differential Pressure	G31-Z002-PDSH-87A	-	25 psid ± 1.00 psid Increasing	-Cleanup Filt Demin Failure Alarm on Ann. Pnl. A-4.
	or G31-Z002-PDSH-87B	-	30 psid ± 1.00 psid Increasing	-Filter-Demineralizer placed in Hold after a 5 second delay.
Filter/Demineralizer A or B Resin Trap Differential	G31-Z002-PDSH-88A	-	5 psid ± 0.5 psid Increasing	-Cleanup Filt Demin Failure on Alarm on Ann. Pnl. A-4
	or G31-Z002-PDSH-88B	-	10 psid ± 0.1 psid Increasing	-Filter-Demineralizer placed in Hold after a 5 second delay.

TABLE 14-3
Page 5 of 7
Instrument and Control Setpoints - Reactor Water Cleanup System

INSTRUMENT TRIP FUNCTION	INSTRUMENT DESIGNATION	INDICATOR/ RECORDER	TRIP SETPOINT	AND FUNCTION
RWCU System Pumps RBCCW Outlet Temperature	G31-TS-N002A or G31-TS-N002B	- -	140° ± 2°F Increasing	-Cleanup Pumps Cooling Wtr Temp Hi Alarm on Ann. Pnl. A-4. -RWCU Pumps Trip.
Non Regenerative Heat Exchanger Outlet	G31-TS-N020	-	130°F ± 2°F Increasing	-Cleanup Filt Inlet Temp Hi Alarm on Ann. Pnl. A-4.
Temperature	G31-TS-N008	-	135°F ± 3°F Increasing	-Nonregen Hx Disch High Temp on Ann. Pnl. A-2. -Closes Valve F004 which trips the RWCU Pumps.
Standby Liquid Control Initiation	C41A-CS-S1 *	-	System Initiation	-Isolates G31-F004 System Outboard Isolation Valve (VLV. GRP. 3) (See SD-12).

* Tech. Spec. Related

TABLE 14-3
Page 6 of 7
Instrument and Control Setpoints - Reactor Water Cleanup System

INSTRUMENT TRIP FUNCTION	INSTRUMENT DESIGNATION	INDICATOR/ RECORDER	TRIP SETPOINT AND FUNCTION	
RWCU Pump/Hx Room Ambient Temperature	B21-XY-5949A CH. A1-1 CH. A2-1 CH. A3-1 B21-XY-5949B CH. A1-1 CH. A2-1 CH. A3-1 *	B21-XY-5949A or B21-XY-5949B	140°F Increasing	-"STM Leak Det Ambient Temp HI" Annunciator (A-02 5-7) -Inverse video "A" alarm flag and "I" isolate flag present on B21-XY-5949A or B21-XY-5949B Ch. A1-1, A2-1, or A3-1. -Isolates RWCU System (VLV. GRP. 3) (See SD-12).
RWCU Pump/Hx Room Differential Temperature (In/Out)	B21-XY-5949A CH. A4-1 CH. A5-1 CH. A6-1 B21-XY-5949B CH. A4-1 CH. A5-1 CH. A6-1 *	B21-XY-5949A or B21-XY-5949B	47°F Increasing	-"STM Leak Det • •Temp HI" Annunciator (A-02 6-7) -Inverse video "A" alarm flag and "I" isolate flag present on B21-XY-5949A or B21-XY-5949B Ch. A4-1, A5-1, or A6-1. -Isolates RWCU System (VLV. GRP. 3) (See SD-12).

* Tech. Spec. Related

TABLE 14-3
Page 4 of 7
Instrument and Control Setpoints - Reactor Water Cleanup System

INSTRUMENT TRIP FUNCTION	INSTRUMENT DESIGNATION	INDICATOR/ RECORDER	TRIP SETPOINT AND FUNCTION	
RWCU System Discharge Pressure	G31-PSH-N014	-	140 psig ± 3 psig Increasing	-Cleanup Disch Press Hi/Lo Alarm on Ann. Pnl. A-4. -Closes Valve F033.
	or			
	G31-PSL-N013	-	5 psig ± 3 psig Decreasing	-Cleanup Disch Press Hi/Lo Alarm on Ann. Pnl. A-4. -Closes Valve F033.
Filter/Demineralizer A or B Effluent Pressure (Effluent Low Flow)	G31-Z002-PSL-74A or G31-Z002-PSL-74B	- -	6 psig ± 0.65 psig (Equivalent to 60 gpm ± 5 gpm Decreasing)	-Cleanup Filt Demin Failure Alarm on Ann. Pnl. A-4. -F/D goes into HOLD. -Local Alarm on Pnl. G31-Z002-25.

* Tech. Spec. Related

6. 205000 1

Unit Two is in MODE 4. Shutdown Cooling mode of RHR is lost and cannot be reestablished.

Which one of the following identifies the reactor water level band directed by 0AOP-15.0, Loss of Shutdown Cooling, unless otherwise directed by the CRS, and the reason for this level band?

A. 182-200 inches to ensure forced circulation.

B. 182-200 inches to ensure natural circulation.

C. 200-220 inches to ensure forced circulation.

D. 200-220 inches to ensure natural circulation.

Answer: D

K/A:
205000 Shutdown Cooling System (RHR Shutdown Cooling Mode)
K3 Knowledge of the effect that a loss or malfunction of the SHUTDOWN COOLING SYSTEM (RHR SHUTDOWN COOLING MODE) will have on following: (CFR: 41.7 / 45.4)
02 Reactor water level: Plant-Specific

RO/SRO Rating: 3.2/3.3

Pedigree: Bank

Objective: LOI-CLS-LP-302-L, Objective 3
Given plant conditions and AOP-15.0, Loss of Shutdown Cooling, determine the required supplementary actions.

Reference: None

Cog Level: Fund

Explanation: With both loops of RHR in Shutdown Cooling, no Reactor Recirculation pumps would be in operating. When Shutdown Cooling is lost due to the Group 8 isolation, forced circulation is lost. Therefore level must be raised to 200-220 inches to promote natural circulation.

Distractor Analysis:

Choice A: Plausible because if Recirculation Pumps or other means of forced circulation were present, then this would be correct. Trainee must know that both Recirc pumps are tripped under the given conditions.

Choice B: Plausible because this is the level required for forced circulation, but not natural circulation. Trainee must know that both Recirc pumps are tripped under the given conditions.

Choice C: Plausible because this is the level for natural circulation, but not needed for forced circulation.

Choice D: Correct Answer, see explanation.

SRO Basis: N/A

LOSS OF SHUTDOWN COOLING	0AOP-15.0
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4.2 Supplementary Actions (continued)

2. **IF** forced circulation has been lost,
AND natural circulation has **NOT** been established,
THEN ensure reactor vessel water level is being maintained between 200 inches and 220 inches as read on B21-LI-R605A(B) (RPV Water Level),
OR as directed by the Unit CRS based on plant conditions until forced circulation is restored... ☐

7. 205000 2

Unit Two is in MODE 3 with RHR Loop A in Shutdown Cooling. RHR pump 2A is running and RHR pump 2C is in standby. A small steam leak results in the following conditions:

RPV water level	176 inches
RPV pressure	40 psig
Drywell pressure	1.9 psig

Which one of the following identifies how RHR Loop A will respond?

A. Group 8 isolates and RHR Pump 2A trips.

B. RHR Pump 2C auto starts and cooldown rate rises.

C. RHR Pump 2C auto starts but decay heat removal is lost.

D. RHR Pump 2A remains running and RHR Pump 2C remains off.

Answer: C

K/A:
205000 Shutdown Cooling System (RHR Shutdown Cooling Mode)
K5 Knowledge of the operational implications of the following concepts as they apply to SHUTDOWN COOLING SYSTEM (RHR SHUTDOWN COOLING MODE): (CFR: 41.5 / 45.3)
03 Heat removal mechanisms

RO/SRO Rating: 2.8/3.1

Pedigree: Bank

Objective: LOI-CLS-LP-043, Objective 16c
Given plant conditions, predict the effect that the following will have on the Service Water System: LOCA

Reference: None

Cog Level: High

Explanation: There is no Group 8 isolation signal, so RHR Pump A will continue to run. RHR Pump C will start on a LOCA signal, and the RHR SW pumps will trip on a LOCA signal, removing the decay heat removal mechanism.

Distractor Analysis:

Choice A: Plausible because trainee must know Group 8 isolation signals and effect on system.

Choice B: Plausible because RHR Pump C will auto start, and if RHR SW is running, the heat removal rate will rise.

Choice C: Correct Answer, see explanation.

Choice D: Plausible because if it is not determined that a LOCA signal exists, this would be correct.

SRO Basis: N/A
SD-43:

RHR Service Water Pumps

Each RHRSW Booster Pump has its own STOP/START spring return to neutral switch on Panel P601. Placing the control switch to START will start the pump if:

- The motor operated isolation Valve F002 is open
- Pump suction pressure is greater than 15.5 psig.
- An undervoltage condition does not exist on the E bus

A second two position NORMAL/OVERRIDE keylock switch on P601 for the respective division RHRSW Booster Pumps exists. These pumps will trip if a LOCA signal exists. To start the RHRSW Booster Pump with a LOCA signal present, the keylock switch must be placed in OVERRIDE.

8. 206000 1

With Unit Two operating at rated power, the following HPCI pump parameters are recorded IAW 0PT-09.2, HPCI System Operability Testing:

Suction (stopped) pressure	6 psig
Suction (running) pressure	4 psig
Lubricant Level Normal	Visible in sight glass
Discharge pressure	335 psig
Flow rate	4550 GPM
Turbine speed	2490 RPM (using a portable speed indicator)
Vibration Position 4H	0.224 in/s peak
Vibration Position 4V	0.275 in/s peak
Vibration Position 9H	0.315 in/s peak
Vibration Position 9V	0.504 in/s peak
Vibration Position 10A	0.248 in/s peak
Vibration Position 10H	0.367 in/s peak
Vibration Position 10V	0.212 in/s peak

Which one of the following identifies the status of HPCI IAW 0PT-09.2 based on these readings?

(Reference provided)

A. HPCI meets the Acceptance Criteria and no additional action is required.

B. HPCI data falls within the Alert Range, and testing frequency must be doubled.

C. HPCI does not meet the Acceptance Criteria based on low Pump Dp. The instruments may be recalibrated and the pump retested.

D. HPCI does not meet the Acceptance Criteria based on vibration data and shall not be returned to service until the condition is corrected.

Answer: D

K/A:
206000 High Pressure Coolant Injection System
G2.2.36 Ability to analyze the effect of maintenance activities, such as degraded power sources, on the status of limiting conditions for operations. (CFR: 41.10 / 43.2 / 45.13)

RO/SRO Rating: 3.1/4.2

Pedigree: 2008 Makeup

Objective: LOI-CLS-LP-019, Objective 24
Given plant conditions, determine whether minimum Technical Specification requirements associated with the HPCI System are met.

Reference: 0PT-09.2, Section 6.0 and attachment 2

Cog Level: High

Explanation: Per attachment 2, pump dp is discharge pressure minus suction pressure (running). This is 335 - 4 = 331 psid. Since minimum dp for unit 2 is 330.3, acceptance criteria is met. If the wrong suction pressure is used or the Unit One data is used, the pump would not be within the acceptance range. Flow rate is the reference value. Speed is within acceptable value. Vibration data for Position 9V is outside REQUIRED ACTION RANGE. and, therefore, the system must be declared inoperable IAW Acceptance Criteria 6.1.4.

Distractor Analysis:

Choice A: Plausible because most of the data is within the required acceptable range. Data must be analyzed to determine values outside the ALERT or REQUIRED ACTION RANGE.

Choice B: Plausible because this action would be required if within the ALERT Range, and not in the REQUIRED ACTION RANGE.

Choice C: Plausible because Pump dp must be calculated and this action would be appropriate if outside the ALLOWABLE ACCEPTANCE VALUE, but it is not.

Choice D: Correct Answer, see explanation.

SRO Basis: N/A

6.0 ACCEPTANCE CRITERIA

This test may be considered satisfactory when the following criteria are met:

6.1 Pump Tests

6.1.1 The HPCI pump develops a flow rate of greater than or equal to 4250 gpm with a pump discharge pressure of greater than or equal to 1110 psig when reactor pressure is between 945 psig and 1045 psig.

6.1.2 The pump test data shall be compared to the allowable ranges identified in Test Information Attachment 2.

6.0 ACCEPTANCE CRITERIA

6.1.3 **IF** deviations fall within the ALERT RANGE of Attachment 2, **THEN**:

1. An NCR shall be initiated to evaluate component per 0OI-01.01, BNP Conduct of Operations Supplement, and OPS-NGGC-1305, Operability Determinations, for degraded or nonconforming conditions.

2. IST Program Engineer shall be notified to initiate a Surveillance Testing Request (STR) to double the frequency of testing.

6.1.4 If the deviations fall within the REQUIRED ACTION RANGE of Attachment 2, then the pump shall be declared inoperable and **NOT** returned to service until the condition has been corrected.

6.1.5 When completed test results show deviations outside the allowable ACCEPTANCE VALUE, the instruments involved may be recalibrated and the test rerun. However, this shall **NOT** preclude declaring the pump inoperable as required.

HPCI Pump Data Sheet

Unit 2

Discharge pressure - suction pressure (running) = delta P (dP)

335 - 4 = 331

Lubricant level normal X

NOTES: 1. Pump vibration is required only during CPT and is measured at the test point marked on the pump for the correct bearing number and direction as indicated by the Test Position number as follows:
- the number indicates the bearing number from Attachment 6
- for direction, A = Axial, H = Horizontal, V = Vertical

2. The magnetic holder is to be used with the accelerometer probe for all vibration readings.
3. Should quarterly pump test data exceed the CPT limits, the pump remains operable and the test results will be evaluated as part of the BNP IST trending program.

UNIT 2 HPCI PUMP TEST DATA

TEST PARAMETER	ACTUAL VALUE	REFERENCE VALUE	ACCEPTABLE VALUE	ALERT RANGE		REQUIRED ACTION RANGE	
				LOW	HIGH	LOW	HIGH
Suction Press. (Stopped) psig	6	N/A	≥ 4	N/A	N/A	< 4	N/A
Suction Press. (Running) psig	4	N/A	N/A	N/A	N/A	N/A	N/A
Discharge Press. psig	335	N/A	N/A	N/A	N/A	N/A	N/A
Flow Rate gpm	4550	4550	N/A	N/A	N/A	N/A	N/A
Turbine Speed rpm	2940	2500	2485 to 2515*	N/A	N/A	N/A	N/A
Pump DP psid	331	367	330.3 to 403.7	N/A	N/A	< 330.3	> 403.7
Vibration-vel(in/s) peak Position 4 H	0.224	0.056	0 to 0.140	N/A	> 0.140 to 0.336	N/A	> 0.336
Vibration-vel(in/s) peak Position 4 V	0.275	0.163	0 to 0.325	N/A	> 0.325 to 0.700	N/A	> 0.700
Vibration-vel(in/s) peak Position 9 H	0.315	0.087	0 to 0.217	N/A	> 0.217 to 0.522	N/A	> 0.522
Vibration-vel(in/s) peak Position 9 V	0.504	0.080	0 to 0.200	N/A	> 0.200 to 0.480	N/A	> 0.480
Vibration-vel(in/s) peak Position 10 A	0.248	0.108	0 to 0.270	N/A	> 0.270 to 0.648	N/A	> 0.648
Vibration-vel(in/s) peak Position 10 H	0.367	0.131	0 to 0.325	N/A	> 0.325 to 0.700	N/A	> 0.700
Vibration-vel(in/s) peak Position 10 V	0.212	0.100	0 to 0.250	N/A	> 0.250 to 0.600	N/A	> 0.600

* Range given in Acceptable Value only applicable when using a portable speed indicator.

7.4 Restoration

1. **Perform** review of completed procedure sections to verify Section 5.0, Acceptance Criteria, for tests performed, have been met. .... _________

2. **IF** Acceptance Criteria is **NOT** met, **THEN perform** following: .... _________

   a. **Report** any equipment found INOPERABLE or **NOT** meeting Acceptance Criteria to Unit CRS. .... _________

   b. **Ensure** CR has been initiated. .... _________

3. **Notify** Unit CRS when this test is complete or found to be unsatisfactory. .... _________

9. 209001 1

A line rupture has occurred in the Unit One drywell. Plant conditions are:

Drywell pressure	18 psig
Reactor water level	60 inches
Reactor pressure	350 psig

Which one of the following identifies the expected status of the Core Spray System?

The Core Spray System injection valves are ___(1)___.

The shutoff head of the Core Spray pumps is approximately ___(2)___ psig.

A. (1) closed
   (2) 200

B. (1) closed
   (2) 300

C. (1) open
   (2) 200

D. (1) open
   (2) 300

Answer: D

K/A:
209001 Low Pressure Core Spray System
A3 Ability to monitor automatic operations of the LOW PRESSURE CORE SPRAY SYSTEM including: (CFR: 41.7 / 45.7)
03 System pressure

RO/SRO Rating: 3.5/3.5

Pedigree: New

Objective: LOI-CLS-LP-018, Objective 7
Given plant conditions, determine if the Core Spray System should automatically initiate.

Reference: None

Cog Level: High

Explanation: Below 410 psig, injection valves open. With Reactor pressure above 300 psig, the Core Spray Pumps do not inject to the vessel.

Distractor Analysis:

Choice A: Plausible because for part 1, the student must know when the injection valves come open, and part 2 is the shutoff head for RHR.

Choice B: Plausible because for part 1, the student must know when the injection valves come open, and part 2 is correct.

Choice C: Plausible because part 1 is correct, and part 2 is the shutoff head for RHR.

Choice D: Correct Answer, see explanation

SRO Basis: N/A

SD-18 (Core Spray):

Water will only inject into the reactor vessel when reactor vessel pressure is less than the pump shutoff head (~300 psig).

- The Outboard Injection Valve (E21-F004) receives an automatic "open" signal (opens if closed after 10-second time delay and reactor pressure < 410 psig).
- The Inboard Injection Valve (E21-F005) receives an automatic "open" signal (opens after 10 second time delay and reactor pressure < 410 psig).
- System injection flow begins when reactor pressure is reduced below the shutoff head of the Core Spray pump.

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SD-17 (RHR)

- Loop injection flow begins when reactor pressure is reduced below the shutoff head of the RHR Pumps (approximately 202 psig). At approx. 185 psig, flow will be observed.

10. 211000 1
Which one of the following completes the statements below concerning the ATWS Control Procedure?

When SLC Tank level is 30%, ___(1)___ Shutdown Boron Weight has been injected into the Reactor.

Raising the reactor water level band increases ___(2)___ circulation for boron mixing.

A. (1) Cold
(2) natural

B. (1) Cold
(2) forced

C. (1) Hot
(2) natural

D. (1) Hot
(2) forced

Answer: C

K/A:
211000 Standby Liquid Control System
G2.4.09 Knowledge of low power/shutdown implications in accident (e.g., loss of coolant accident or loss of residual heat removal) mitigation strategies. (CFR: 41.10 / 43.5 / 45.13)

RO/SRO Rating: 3.8/4.2

Pedigree: Bank

Objective: LOI-CLS-LP-300-E, Objective 11b, c
Given plant conditions and the ATWS Control Procedure, determine the following:
b. If Hot or Cold Boron Weight has been injected into the reactor
c. When boron mixing is required after being injected into the reactor

Reference: None

Cog Level: Fund

Explanation: See Notes Section. In addition, Reactor Recirculation pumps are tripped based on Reactor level and the ATWS Control Procedure. Therefore, natural circulation is the only means available for mixing.

Distractor Analysis:

Choice A: Plausible because student must differentiate between Hot and Cold Boron Weight. Cold Shutdown boron Weight also uses tank level. Second part is correct.

Choice B: Plausible because student must differentiate between Hot and Cold Boron Weight. Student must know that both Recirc Pumps would be tripped under the given conditions, and only natural circulation is available for mixing.

Choice C: Answer, see explanation

Choice D: Plausible because first part is correct. Student must know that both Recirc Pumps would be tripped under the given conditions, and only natural circulation is available for mixing.

SRO Basis: N/A

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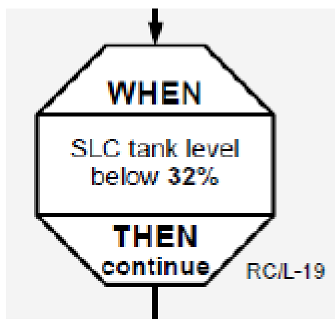
5.15 Step RC/L-19



When Hot Shutdown Boron Weight (HSBW) has been injected the operator continues in the RC/L flow path.

5.16 Step RC/L-20

Restore and **maintain** RPV level between **+166 inches** and **+206 inches** using Table L-2 and L-3 systems.

IF	THEN
RPV level **CANNOT** be restored and maintained between **+166 inches** and **+206 inches**	**Restore** and **maintain** above **LL-4**
RPV level **CANNOT** be restored and maintained above **LL-4**	**Proceed to** 6

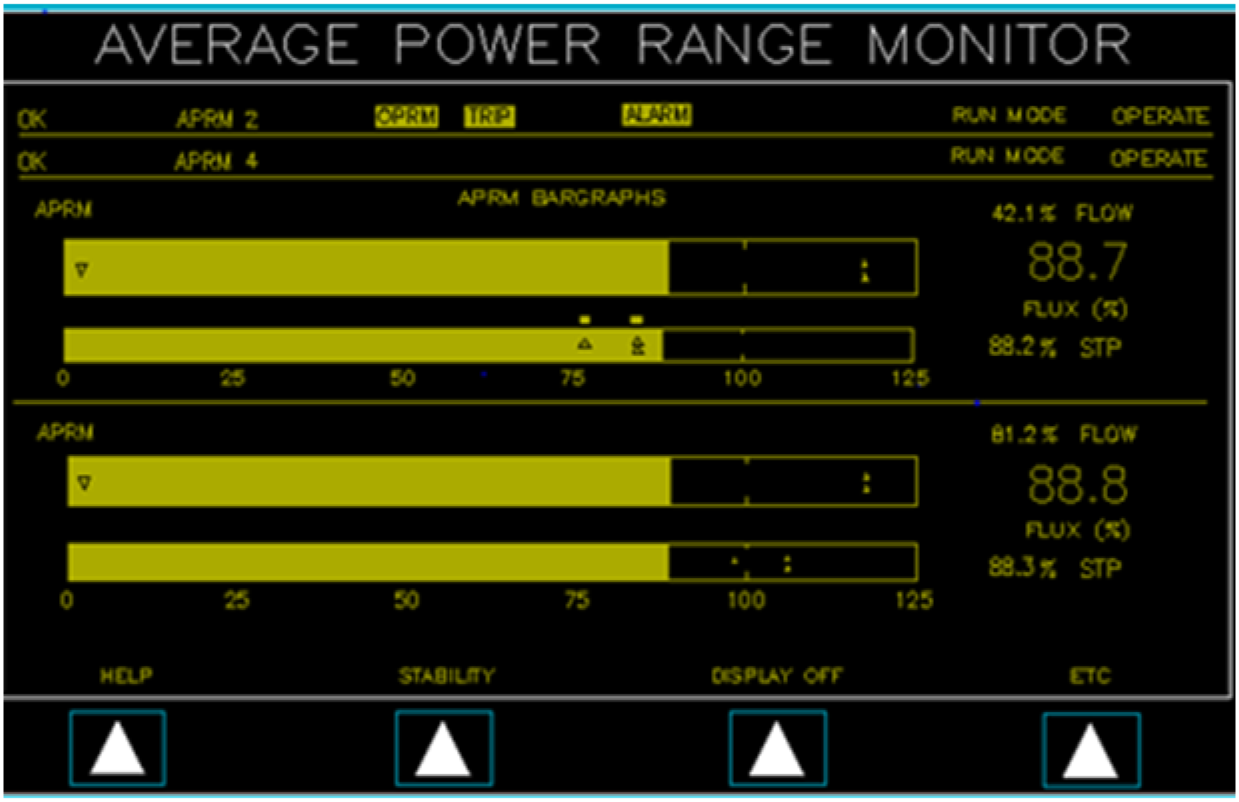
RC/L-20

Step RC/L-20 is a complex action step. The preferred strategy is to restore and maintain RPV level between +166 inches and +206 inches using Table L-2 and Table L-3.

When an amount of boron equivalent to HSBW has been injected, RPV level is restored to and maintained within the normal operating range. As RPV level is increased, natural circulation flow increases and boron which has accumulated in the lower plenum is quickly mixed and distributed throughout the core region. This phenomenon is known as "boron remixing," thereby distinguishing it from any mixing which may have occurred in the early phase of the transient when some core flow was present.

11. 212000 1

A Unit Two APRM ODA shows the following indications:



Which one of the following completes the statements below?

The cause of the indications and alarms for APRM 2 is a ____(1)____.

As a result of this condition, Voter input status lights will show a trip on ____(2)____.

A. (1) Critical Self-Test Fault
   (2) Voter 2 **ONLY**

B. (1) Critical Self-Test Fault
   (2) all 4 Voters

C. (1) Recirculation flow unit failed downscale
   (2) Voter 2 **ONLY**

D. (1) Recirculation flow unit failed downscale
   (2) all 4 Voters

Answer: D

K/A:
212000 Reactor Protection System
K1 Knowledge of the physical connections and/or cause-effect relationships between REACTOR PROTECTION SYSTEM and the following: (CFR: 41.2 to 41.9 / 45.7 to 45.8)
03 Recirculation system

RO/SRO Rating: 3.4/3.6

Pedigree: New

Objective: LOI-CLS-LP-09.6, Objective 12g
Given plant conditions, predict the response of the PRNMS to a malfunction/failure of the following systems/components: Recirc Flow Module

Reference: None

Cog Level: Higher

Explanation: A flow signal failure downscale will generate a flow-biased upscale trip for the effected APRM. An upscale trip will input to all 4 Voters. This question meets the intent of the K/A because the Voters feed into the RPS trip systems. RPS trip logic is 2/4 Voters to cause a full scram. A half scram will not be indicated with a Recirc flow unit failure.

Distractor Analysis:

Choice A: Plausible because an upscale failure of a flow unit would indicate an alarm on the APRM, but not a trip. A trip would be indicated on Voter 2, but also on the other 3 Voters

Choice B: Plausible because an upscale failure of a flow unit would indicate an alarm on the APRM, but not a trip. Part 2 is correct.

Choice C: Plausible because part 1 is correct. A trip would be indicated on Voter 2, but also on the other 3 Voters.

Choice D: Correct Answer, see explanation.

SRO Basis: N/A

The Recirculation Flow Monitor System provides a signal representative of total recirculation flow rate for use by the APRMs in determining flow-biased upscale trips and alarms for reactor power.

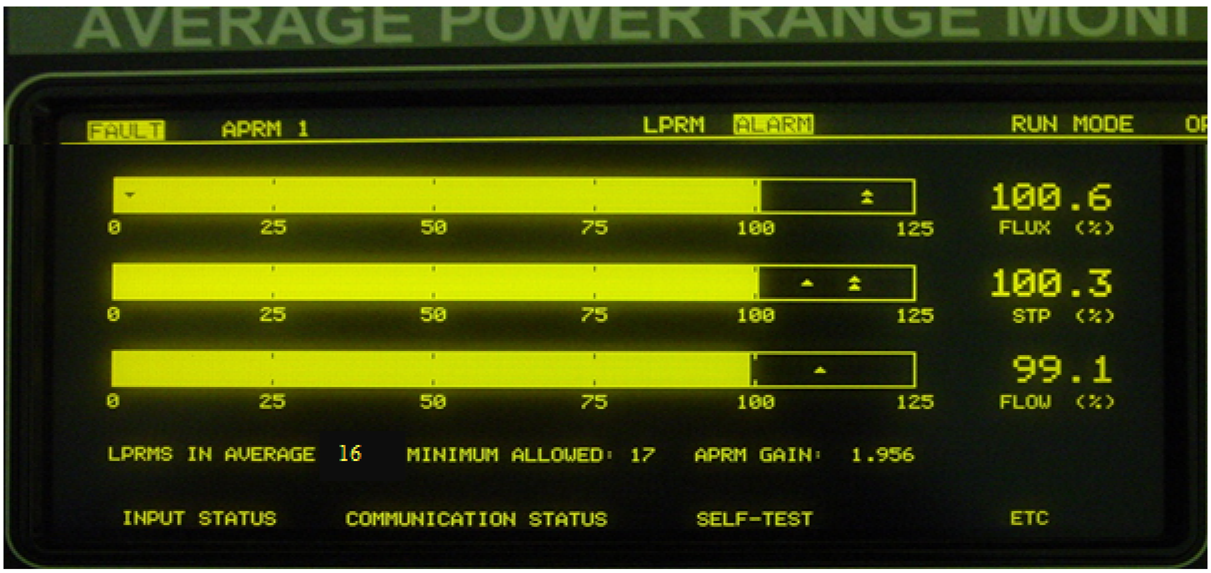
The Two-Out-of-Four Logic System receives safety parameter information from each of the four APRM/OPRM channels. This logic scheme supplies trip inputs to RPS, as necessary, based upon plant conditions.

Total Recirculation Flow Rate is used by the APRM System to determine the flow-biased upscale trips and alarms for Simulated Thermal Power (STP). Recirc Flow Rate is also used to define a region of power and flow in which the OPRM System is enabled. Eight differential pressure transmitters monitor the flow in each recirculation loop. Each APRM NUMAC processes the signals from one sensor in Loop A and one sensor in Loop B and averages the signals to obtain the total Recirc flow rate.

The VOTERS serve as the interface between the APRM/OPRM channels, which generate safety trips, and the RPS. Each of the four VOTERS corresponds to a channel of the A1, A2, B1, and B2 RPS logic. The VOTER outputs to the RPS logic are: A1 (VOTER 1), A2 (VOTER 3), B1 (VOTER 2), and B2 (VOTER 4). VOTERS cannot be bypassed.

The VOTER logic is arranged so that if any one APRM channel loses power, becomes inoperative, or indicates a trip, no trip outputs to RPS will result, and no half-scrams will occur. However, a trip in any two non-bypassed APRM channels (for either the APRM function or the OPRM function) will result in trip outputs to RPS from all four VOTER modules and result in a scram.

Failed Recirc Flow transmitters could result in control rod blocks or trip signals to be generated by the associated APRM, depending on the direction of failure and the initial reactor power level. For example, if one of the two Recirc flow signals to an APRM failed to a zero signal with reactor power at 100%, its OPRM becomes enabled because the calculated flow is reduced to one half of its initial value, and its STP rod block and trip set point will be exceeded because the flow used to calculate the STP rod block and trip set points is also reduced to one half of its initial value. The other APRM channels are not affected since they use separate record flow signals.



12. 215002 1

Which one of the following is the power supply to APRM Channel 4 NUMAC?

A. RPS

B. UPS

C. Div I DC

D. Div II DC

Answer: A

K/A:
215002 Rod Block Monitor System
K2 Knowledge of electrical power supplies to the following: (CFR: 41.7)
03 APRM channels: BWR-3,4,5

RO/SRO Rating: 2.8/2.9

Pedigree: 2008 NRC Makeup

Objective: LOI-CLS-LP-09.6, Objective 7a
Describe the operational relationships between the PRNMS and the following:
Reactor Protection System

Reference: None

Cog Level: Fundamental

Explanation: Each APRM channel NUMAC is equipped with a dual power supply arrangement with one supply from RPS Bus A and the other supply from RPS Bus B. All four APRM channels maintain power on loss of either supply as long as the other supply is available

Distractor Analysis:

Choice A: Correct Answer, see explanation.

Choice B: Plausible since UPS does provide power to the APRM recorders and the Operator Display Assemblies on the main control room panel. Without UPS, APRM channels still have power but monitoring capability on the main control room panel is lost.

Choice C: Plausible since this is where the other ranges of nuclear instrumentation (SRM and IRM) receive their power.

Choice D: Plausible since this is where the other ranges of nuclear instrumentation (SRM and IRM) receive their power.

SRO Basis: N/A

2.8.8 PRNMS Power Supplies

The Power Range Neutron monitoring System uses one Quadruple Voltage Power Supply (QLVPS) chassis and four Dual Low Voltage Power Supplies (DLVPS), one for each bay of the PRNMS panel, to provide redundant power to the NUMAC instruments. These LVPS convert 120 VAC to low voltage DC. See Figure 09.6-14.

Each APRM instrument receives power from two power supplies, LVPS 1 and LVPS 4. LVPS 1 is fed from RPS Bus A while LVPS 4 is fed from RPS Bus B. Therefore, a loss of an RPS Bus will not affect operation of the APRM NUMACS. Each RBM instrument also

4.3.1 Reactor Protection System

APRM channels provide signals to open contacts in the scram trip logic of the RPS System under various conditions discussed previously.

The RPS System provides power to each of the four APRM instruments, which in turn provide power to all subsystems driven from the APRM instruments or NUMAC. Both RPS busses, A and B, provide power to each APRM instrument, as well as, each RBM. Therefore, a loss of one RPS bus will not affect operation of the PRNMS.

The reactor mode switch provides input to each APRM instrument to determine when to enforce the fixed or flow biased scram trip and rod block settings. OPRM circuitry is enabled only when power/flow conditions are met and the mode switch in RUN.

4.3 Interrelationships With Other Systems

Unit (2) in parenthesis

4.3.1 ± 24 Volt DC Distribution System

The SRM and IRM Systems receive power from the ± 24 VDC System. Panels 21A(23A) supply channels A/C; and Panels 22B(24B) supply channels B/D to power the detector HVPS, instrumentation and trip units. Channels A and B use the ± 24 VDC power supply which is regulated to a smooth ± 15 VDC prior to entering into the high volts power supply.

Loss of 24 VDC to the SRM or IRM channel components will initiate protective functions to both RMCS and RPS Systems resulting in rod withdraw blocks and scram signals being generated. Flux monitoring capabilities for the affected channels will also be lost as well as annunciator power. Failure of the SRM or IRM System will have no adverse effect on the ± 24 VDC Distribution System.

13. 215003 1

Which one of the following distribution systems identifies the power supply to the Intermediate Range Monitor (IRM) detectors?

A. 24/48 VDC

B. 125/250 VDC

C. 120 VAC UPS

D. 120 VAC from Emergency Power

Answer: A

K/A:
215003 Intermediate Range Monitor System
K2 Knowledge of electrical power supplies to the following: (CFR: 41.7)
01 IRM channels/detectors

RO/SRO Rating: 2.5/2.7

Pedigree: Bank

Objective: LOI-CLS-LP-009-A, Objective 2o
State the purpose and/or function of the following components pertaining to the SRM and IRM systems as applicable: Power Supplies

Reference: None

Cog Level: Fundamental

Explanation: The SRM and IRM Systems receive power from the ± 24 VDC System. Panels 21A(23A) supply channels A/C; and Panels 22B(24B) supply channels B/D to power the detector HVPS, instrumentation and trip units. IRM channels provide input to the RPS System to generate a reactor trip signal to its respective RPS channel. The 120 VAC Emergency Power System supplies power to the SRM & IRM drive motor and motor control circuits through Panel 1(2)AB-RX. Power is also supplied to the IRM alarm indicating lights on P603 through this distribution panel. The 120 VAC UPS System supplies power to the SRM recorders and indicators on P603, and IRM, APRM/RBM Recorders on P603 through Panel V7(8)A.

Distractor Analysis:

Choice A: Correct Answer, see explanation.

Choice B: Plausible because it is another DC system. Student must know which DC system supplies the IRM detectors.

Choice C: Plausible because 120 VAC UPS system supplies power to the IRM recorders on P603.

Choice D: Plausible because the 120 VAC Emergency Power System supplies power to the IRM drive motor and motor control circuits through Panel 1(2)AB-RX.

SRO Basis: N/A

4.3.1 ± 24 Volt DC Distribution System

The SRM and IRM Systems receive power from the ± 24 VDC System. Panels 21A(23A) supply channels A/C; and Panels 22B(24B) supply channels B/D to power the detector HVPS, instrumentation and trip units. Channels A and B use the ± 24 VDC power supply which is regulated to a smooth ± 15 VDC prior to entering into the high volts power supply.

Loss of 24 VDC to the SRM or IRM channel components will initiate protective functions to both RMCS and RPS Systems resulting in rod withdraw blocks and scram signals being generated. Flux monitoring capabilities for the affected channels will also be lost as well as annunciator power. Failure of the SRM or IRM System will have no adverse effect on the ± 24 VDC Distribution System.

IRM channels provide input to the RPS System to generate a reactor trip signal to its respective RPS channel.

Assignment of IRMS to the RPS System is as follows:

RPS A receives signals from IRMs A, C, E, G
RPS B receives signals from IRMs B, D, F, H

The reactor MODE SWITCH provides input to IRM circuitry to bypass or enforce protective functions of the IRM System dependent upon MODE SWITCH position. Loss of any one IRM will generate a scram signal to its respective RPS channel.

Loss of the RPS system will have no direct effect on the IRM System.

4.3.4 120 VAC Emergency Power System

The 120 VAC Emergency Power System supplies power to the SRM & IRM drive motor and motor control circuits through Panel 1(2)AB-RX. Power is also supplied to the IRM alarm indicating lights on P603 through this distribution panel. Loss of the power supply to the detector drive and controls will render the detectors unmovable. Loss of alarm light power will require the operator to observe the drawer lights to determine IRM status.

4.3.5 UPS

The 120 VAC UPS System supplies power to the SRM recorders and indicators on P603, and IRM, APRM/RBM Recorders on P603 through Panel V7(8)A.

Loss of the UPS power supply will required monitoring of the SRM or IRM channels from the control room back panels. Loss of the SRM or IRM System will have no effect on the UPS System.

4.3.6 Average Power Range Monitoring (APRM) &

14. 215004 1

A plant startup is in progress. A control rod block has occurred. The following nuclear instrument indications are noted:

SRM	Counts	Position	IRM	Counts	Range
A	$3x10^5$	Full In	A	25/125	3
B	190	Mid Position	B	65/125	2
C	$6x10^4$	Full In	C	35/125	3
D	125	Mid Position	D	15/125	3
			E	12/125	2
			F	55/125	2
			G	30/125	3
			H	25/125	3

Which one of the following is the <u>minimum</u> required action(s) that will clear the control rod block?

A. Withdrawing SRM A ONLY.

B. Ranging IRM E to range 3.

C. Withdrawing SRM A and C.

D. Ranging IRM B and F to range 3.

Answer: A

K/A:
215004 Source Range Monitor System
K5 Knowledge of the operational implications of the following concepts as they apply to SOURCE RANGE MONITOR (SRM) SYSTEM: (CFR: 41.5 / 45.3)
01 Detector operation

RO/SRO Rating: 2.6/2.6

Pedigree: NRC Exam 10-1

Objective: CLS-LP-09.1 Objective 9a
Describe the insertion/withdrawal of the SRM detectors, including the following:
Reason for maintaining counts between 125 and $2x10^5$.

Reference: None

Cog Level: High

Explanation: To clear the rod block SRM must be below $2x10^5$ or IRMs must be > range 7. The retract permit is bypassed with IRMs ≥ range 3. Withdrawing SRM A will cause the rod block to clear when less than $2x10^5$.

Distractor Analysis:

Choice A: Correct Answer, see explanation.

Choice B: Plausible because IRM E is the only Div I IRM below range 3. If all Div I IRMs are above range 3 then the rod block from SRM Retract Permissive in would be bypassed, not the signal from SRM upscale. Also ranging IRM E to range 3 will cause an IRM downscale which is a rod block.

Choice C: Plausible because SRM A does need to be withdrawn and C is above the old setpoint for the upscale alarm. (recent change, old setpoint was $5x10^4$).

Choice D: Plausible because IRM B & F are the only Div II IRMs below range 3 and these do meet the requirements for ranging them to 3. If all Div II IRMs are above range 3 then a rod block from SRM Retract Permissive would be bypassed on Div II, not the signal from SRM upscale.

SRO Basis: N/A

TABLE 09.1- 1
INSTRUMENT AND CONTROL SETPOINTS
STARTUP RANGE NEUTRON MONITORING SYSTEM

INSTRUMENT DESIGNATION AND TRIP FUNCTION	TRIP SETPOINT AND FUNCTION	FUNCTION, ADDITIONAL CONDITIONS AND COMMENTS
SRM Inop Trip C51-SRM-K600 (A-D) TRM Annunciator "SRM UPSCALE/INOP" (A-05 2-3)	HVPSS - 10% ± 1%* Switch not in OPERATE or SRM Module unplugged	Initiates a **rod block** if the following conditions are met: •Reactor MODE SWITCH is <u>not</u> in RUN •ANY divisional IRM < Range 8 and <u>NOT</u> bypassed. <u>Note:</u> Bypassed if all divisional IRMs are above Range 7
SRM Downscale Trip C51-SRM-K600 (A-D) TRM Annunciator "SRM DOWNSCALE" (A-05 1-3)	5 ± 1.5 cps	Initiates a **rod block** if the following conditions are met: •Reactor MODE SWITCH is <u>not</u> in RUN •ANY divisional IRM < Range 3 and <u>NOT</u> bypassed. Note: Bypassed if all divisional IRMs are above Range 2
SRM Retract Permissive C51-SRM-K600 (A-D) TRM Annunciator "SRM RETRACT NOT PERMITTED" (A-05 4-3)	125 cps (101 to 150)	Initiates a **rod block** if the following conditions are met: •SRM detector not FULL IN •ANY divisional IRM < Range 3 and <u>NOT</u> bypassed. Note: Bypassed if all divisional IRMs are above Range 2
SRM Upscale Alarm C51-SRM-K600 (A-D) TRM Annunciator "SRM UPSCALE/INOP" (A-05 2-3)	$2 X 10^5$ cps ($1.3 X 10^5 – 3.0 X 10^5$)	Initiates **a rod block** if the following conditions are met: •Reactor MODE SWITCH is <u>not</u> in RUN •ANY divisional IRM < Range 8 and <u>NOT</u> bypassed. <u>Note:</u> Bypassed if all divisional IRMs are above Range 7
SRM Upscale Trip C51-SRM-K600 (A-D) TRM	$5 X 10^5$ cps ($3.3 X 10^5 - 7.5 X 10^5$)	**Full Scram** if refueling shorting links removed
SRM Period C51-SRM-K600 (A-D) TRM	50 seconds -10, +16 sec	Annunciator "SRM PERIOD" (A-05 3-3)

TRM Technical Requirement Manual (TRM) related (SRM Instrumentation is Technical Specification related however trips listed are in the TRM)

* HVPSS is the high voltage power supply setting (350-600 Vdc range) and the percentages are of this value. <u>Note</u>: A complete loss of power will produce an apparent trip of all trip units (i.e. Full scram if shorting links are removed due to SRM Upscale Trip)

15. 215005 1

A reactor startup is being performed on Unit Two. Reactor power is currently 18% with the Reactor Mode Switch in Run. APRM Channels 1 and 2 have the following number of operable LPRM inputs:

Level	A	B	C	D
APRM 1	5	3	4	4
APRM 2	6	4	2	5

Which one of the following identifies the effect on the Reactor Manual Control System (RMCS)?

A. Rod Block. APRM 1 **ONLY** INOPERABLE.

B. Rod Block. APRM 2 **ONLY** INOPERABLE.

C. Rod Block. **BOTH** APRM 1 **AND** 2 INOPERABLE.

D. No Rod block. **NEITHER** APRM 1 **NOR** 2 INOPERABLE.

Answer: C

K/A:
215005 Average Power Range Monitor/Local Power Range Monitor System
K1 Knowledge of the physical connections and/or cause-effect relationships between AVERAGE POWER RANGE MONITOR/LOCAL POWER RANGE MONITOR SYSTEM and the following: (CFR: 41.2 to 41.9 / 45.7 to 45.8)
10 Reactor manual control system: Plant-Specific

RO/SRO Rating: 3.3/3.3

Pedigree: New

Objective: LOI-CLS-LP-09.6, Objective 5a
List the PRNMS system signals/conditions that will cause the following actions:
APRM / RBM Rod Blocks

Reference: None

Cog Level: Higher

Explanation: An APRM Trouble alarm will be generated by < 17 LPRM inputs or < 3 detectors per level. APRM A has < 17 operable inputs, and APRM B has < 3 detectors per level, therefore both are inoperable and will generate rod blocks in RMCS.

Distractor Analysis:

Choice A: Plausible because APRM 1 is inoperable, but APRM 2 must also be analyzed.

Choice B: Plausible because APRM 2 is inoperable, but APRM 1 must also be analyzed.

Choice C: Correct Answer, see explanation.

Choice D: Plausible because conditions must be analyzed to determine operability.

SRO Basis: N/A

SD-7:

3.1.3 Rod Motion Inhibits

Any APRM Trouble alarm if initiated by too few LPRM detectors per level or too few LPRM detectors in flux average will generate a rod block. This assures that no control rod is withdrawn unless the average power range neutron monitoring channels are either in service or properly bypassed.

SD-9.6:

A loss of LPRMs such that < 17 inputs or < 3 detectors per level to an APRM exist renders the APRM inoperable causing an APRM trouble alarm.

16. 217000 1

Which one of the following identifies the RCIC functions that will remain operable following a loss of 125 VDC Panel 4A on Unit Two?

A. automatic initiation and inboard isolation logic.

B. automatic initiation and outboard isolation logic.

C. automatic shutdown on high RPV water level and inboard isolation logic.

D. automatic shutdown on high RPV water level and outboard Isolation logic.

Answer: B

K/A:
217000 Reactor Core Isolation Cooling System
K6 Knowledge of the effect that a loss or malfunction of the following will have on the REACTOR CORE ISOLATION COOLING SYSTEM (RCIC): (CFR: 41.7 / 45.7)
01 Electrical power

RO/SRO Rating: 3.4/3.5

Pedigree: New

Objective: LOI-CLS-LP-016, Objective 7
Identify the power supply (bus and voltage) for the following RCIC components:
b. RCIC Logic (initiation, isolation, trip)

Reference: None

Cog Level: Higher

Explanation: RCIC initiation logic powered from Div II DC. Portion of isolation logic and high RPV level shutdown powered from Div I DC. High RPV level input in Div I powered logic energizes on high level to provide input to Div II circuitry. Loss of Division I DC power will make the inboard isolation logic inoperable and resulting failure of the Turbine Steam Supply Valve to automatically close on a high vessel level.

Distractor Analysis:

Choice A: Plausible because part 1 is correct, but part 2 is incorrect.

Choice B: Correct Answer, see explanation.

Choice C: Plausible because part 1 is incorrect, but part 2 is correct.

Choice D: Plausible because although incorrect, a differentiation between Div 1 and Div 2 DC is required to answer the question.

SRO Basis: N/A

4.3.11 DC Electrical Distribution

NOTE:	Unit 2 components, if different, are in parentheses.

The majority of the RCIC System components are powered from Division II 125/250 Vdc Electrical Distribution via MCC 1-XDB (2-XDB) and are summarized below:

<u>Component Description</u>

Barometric Condenser Vacuum Pump

Barometric Condenser Condensate Pump

Turbine Trip & Throttle Valve, E51-V8

CST Suction Valve, E51-F010

Cooling Water Supply Valve, E51-F046

Pump Discharge Valve, E51-F012

Injection Valve, E51-F013

Bypass to CST Valve, E51-F022

Steam Supply Outboard Isolation Valve, E51-F008

Turbine Steam Supply Valve, E51-F045

Suppression Pool Inboard Suction Valve, E51-F031

Suppression Pool Outboard Suction Valve, E51-F029

Minimum Flow Bypass to Suppression Pool Valve, E51-F019

RCIC Steam Supply Inboard Isolation Valve, E51-F007, ASSD Feed

The RCIC Relay Logic B (which includes the Initiation, Trip, and Isolation Logic B), the Remote Turbine Trip, the 48 Vdc Power Supply for the EGM (Control Panel H12-P621) are powered from 125 Vdc Distribution Panel 3B (4B). Control power to the Condensate Pump

Discharge Inboard Drain Valve (E51-F004), the Supply Drain Pot Outboard Drain Valve (E51-F026), the Supply Drain Pot Drain Bypass Valve (E51-F054), Turbine Supervisory Lights, the 24 Vdc Power Supply for the FIC, and the 52.5 Vdc Power Supply for the RTGB indications is from 125 Vdc Distribution Panel 3B (4B).

The RCIC Relay Logic A, which includes Isolation Logic A and one of the required high level inputs to the high vessel level closure of the RCIC Turbine Steam Supply Valve, E51-F045, is powered from 125 Vdc Distribution Panel 3A (4A). Control power to the Condensate Pump Discharge Outboard Drain Valve (E51-F005) and the Supply Drain Pot Inboard Drain Valve (E51-F025) is from 125 Vdc Distribution Panel 3A (4A). The Remote Shutdown Panel RCIC Turbine EGM Control Box is powered from 125 Vdc Distribution Panel 1B (2B).

A loss of Division I DC power will make the inboard isolation logic (Isolation Logic A) inoperative, and result in the failure of the Turbine Steam Supply Valve to automatically close on a high vessel level condition. RCIC operation would be otherwise unaffected.

A loss of Division II DC power will render the RCIC System totally inoperative for normal use.

17. 218000 1

Which one of the following completes the statements below concerning Annunciation of A-03 (1-10) *Safety / Relief Valve Open?*

This alarm is activated by ___(1)___ .

When the alarm clears, the amber light for the effected SRV ___(2)___ be illuminated on the apron section of RTGB Panel P601.

A. (1) a SRV sonic detector
   (2) will

B. (1) a SRV sonic detector
   (2) will NOT

C. (1) high temperature on recorder B2I-TR-6l4, Safety Relief Vlv Temp recorder
   (2) will

D. (1) high temperature on recorder B2I-TR-6l4, Safety Relief Vlv Temp recorder
   (2) will NOT

Answer: A

K/A:
218000 Automatic Depressurization System
A3 Ability to monitor automatic operations of the AUTOMATIC DEPRESSURIZATION SYSTEM including: (CFR: 41.7 / 45.7)
03 ADS valve acoustical monitor noise: Plant-Specific

RO/SRO Rating: 3.7/3.8

Pedigree: 2014 NRC Exam

Objective: LOI-CLS-LP-020, Objective 5
Describe the operation of the SRVs for both an overpressure condition and a manual/ADS actuation.

Reference: None

Cog Level: High

Explanation: This alarm input is from the Sonic detectors and the alarm A-03 (1-1) Safety or Depress Vlv Leaking is from the temperature recorder. The red light indicates the valve is open and the amber light is a memory light. The amber light is reset on the sonic detector panel in the Reactor Building.

Distractor Analysis:

Choice A: Correct Answer, see explanation.

Choice B: Plausible because the sonic detector does cause the alarm but the amber light must be manually reset.

Choice C: Plausible because high temperature causes a different alarm and the amber light is illuminated.

Choice D: Plausible because high temperature causes a different alarm but the amber light must be manually reset.

SRO Basis: N/A

Unit 2
APP A-03 1-10
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DEVICE/SETPOINTS

SRV Sonic Detector Relay B21-74X-F	0.08 - 0.12 (volts)
SRV Sonic Detector Relay B21-74X-E	0.08 - 0.12 (volts)
SRV Sonic Detector Relay B21-74X-D	0.08 - 0.12 (volts)
SRV Sonic Detector Relay B21-74X-C	0.08 - 0.12 (volts)
SRV Sonic Detector Relay B21-74X-B	0.08 - 0.12 (volts)
SRV Sonic Detector Relay B21-74X-A	0.08 - 0.12 (volts)
SRV Sonic Detector Relay B21-74X-G	0.08 - 0.12 (volts)
SRV Sonic Detector Relay B21-74X-H	0.08 - 0.12 (volts)
SRV Sonic Detector Relay B21-74X-J	0.08 - 0.12 (volts)
SRV Sonic Detector Relay B21-74X-K	0.08 - 0.12 (volts)
SRV Sonic Detector Relay B21-74X-L	0.08 - 0.12 (volts)

Unit 2
APP A-03 1-1
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DEVICE/SETPOINTS

Temperature Recorder B21-TR-614(SW1)	287-293°F (337-343°F for B21-F013C only)

18. 218000 2
Which one of the following completes the statement below concerning the 125 VDC power supply to the Unit One ADS logic?

The normal power supply is from Distribution Panel ___(1)___,

The backup power supply is from Distribution Panel ___(2)___.

A. (1) 3A
   (2) 3B

B. (1) 3B
   (2) 3A

C. (1) 3A
   (2) 4A

D. (1) 3B
   (2) 4B

Answer: B

K/A:
218000 Automatic Depressurization System
K2 Knowledge of electrical power supplies to the following: (CFR: 41.7)
01 ADS logic

RO/SRO Rating: 3.1/3.3

Pedigree: Bank

Objective: LOI-CLS-LP-020, Objective 14a,
State the power supplies to the following: a. ADS Logic

Reference: None

Cog Level: Fundamental

Explanation: The automatic transfer of the logic power from 125V DC 3(4)B Panel to 125V DC 3(4)A Panel on a loss of normal power for the B logic will still allow the B logic to initiate.

Distractor Analysis:

Choice A: Plausible because power to ADS Logic is from 125 VDC power and it would not be illogical to assume Div I would be the normal and Div II the backup, but it is just the opposite.

Choice B: Correct Answer, see explanation.

Choice C: Plausible because transfer to same division is not uncommon.

Choice D: Plausible because transfer to same division is not uncommon.

SRO Basis: N/A

4.3.5 DC Power

ADS valves require 125V DC to operate in the ADS mode or to be manually controlled from Panel P601 or the Remote Shutdown Panel. No single failure of DC power will prevent ADS from performing its intended function, up to and including the loss of an entire division of DC power. The automatic transfer of the logic power from 125V DC 3(4)B Panel to 125V DC 3(4)A Panel on a loss of normal power for the B logic will still allow the B logic to initiate. The non-ADS valves located at the remote Shutdown Panel (B21-F013B, E, G) would be inoperable on a loss of the 125V DC MCC-1XDB (Panel 2B). A loss of the 125V DC 3(4)A Panel would prevent the B logic from initiating due to the level transmitters being powered from 125V DC 3(4)A Panel with no automatic transfer. Therefore a loss of 125 VDC Panel 3(4)A renders Logic "B" inop.

19. 219000 1

Unit Two is operating at rated power. 2B RHR and 2B RHR Service Water pumps have been placed in Torus Cooling in preparation for a HPCI surveillance. Torus Cooling has been maximized with the 2E11-F048B, RHR Heat Exchanger Bypass Valve, fully closed.

A subsequent transient occurs with the following plant conditions:

Drywell Pressure	18.1 psig
Torus Pressure	13.7 psig
Reactor Pressure	885 psig
Reactor water level	36 inches

Which one of the following completes the statements below?

2B RHR SW Pump ___(1)___.

2E11-F048B ___(2)___.

A. (1) has tripped
   (2) has auto opened

B. (1) has tripped
   (2) remains closed

C. (1) remains running
   (2) has auto opened

D. (1) remains running
   (2) remains closed

Answer: A

K/A:
219000 RHR/LPCI: Torus/Suppression Pool Cooling Mode
G2.2.37 Ability to determine operability and/or availability of safety related equipment.
(CFR: 41.7 / 43.5 / 45.12)

RO/SRO Rating: 3.6/4.6

Pedigree: 2008 NRC Exam

Objective: LOI -CLS-LP-017, Objective 9
Given an RHR pump or valve, list the interlocks, permissives and/or automatic actions associated with the RHR pump or valve, including setpoints.

Cog Level: Higher

Explanation: The availability of RHR in Torus Cooling is affected by a LPCI signal. The RHR SW Pumps will trip, and the E11-F048 will get an open signal for 3 minutes. The RHR SW Pumps can be restarted by taking a Keylock LOCA override switch to the override position.

Distractor Analysis:

Choice A: Correct answer, see explanation.

Choice B: Plausible because part 1 is correct, part 2 is the opposite.

Choice C: Plausible because student must identify a LOCA signal and know the logic for RHR SW, part 2 is correct.

Choice D: Plausible because student must identify a LOCA signal and know the logic for RHR SW, part 2 is opposite.

SRO Basis: N/A

SD-17:

The LPCI Outboard Injection Valve, F017A(B), is a throttle valve which may be adjusted to control flow into the vessel, whereas the Inboard Injection Valve, E11-F015A(B), is designed for either full open or full close service. E11-F017A(B) is normally open, but with an automatic open signal present, this valve cannot be closed or throttled for 5 minutes to ensure a discharge path exists from the pumps to the vessel. E11-F015A(B), cannot be closed as long as the LPCI initiation signal is present. In addition, the RHR heat exchanger is automatically bypassed via the RHR Heat Exchanger Bypass Valve, E11-F048A(B), for the first three minutes to ensure that flow gets to the reactor through the most direct route. During the interval of time when the RHR pumps are operating to restore the reactor vessel level, heat removal is not necessary.

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- The RHR Heat Exchanger Bypass Valve, E11-F048A(B), receives an open signal. A "closure inhibit" time delay relay prevents the valve from being closed during the first three minutes of LPCI injection. After the three minute timer times out, this valve may be throttled to control flow through the heat exchanger, as desired.

- A trip signal is sent to the RHRSW Booster Pump Breakers since heat removal is not an immediate concern. To start the RHRSW Booster Pumps with a LOCA signal present, the P601 keylocked AUTO/Manual override switch must be placed in Manual OVERRIDE for the desired Loop.

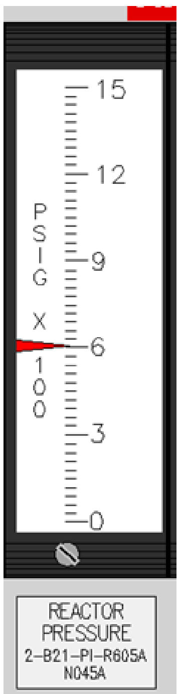
20. 223002 1

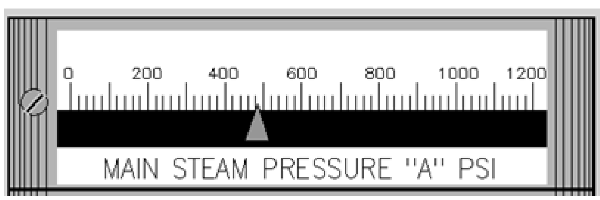
Unit Two Mode Switch is in Startup with reactor pressure at 600 psig.
Main Steam Line A steam flow instrument fails upscale.

Which one of the following completes the statements below?

The MSIVs ____(1)____ receive an automatic close signal.

If a Group 1 isolation occurs, the MSIVs ____(2)____ be opened for a rapid recovery of the Main Condenser, IAW 2OP-25, Reopening the MSIVs Following a Scram, given the following indications:





A. (1) will
(2) may

B. (1) will
(2) may NOT

C. (1) will NOT
(2) may

D. (1) will NOT
(2) may NOT

Answer: C

K/A:
223002 Primary Containment Isolation System/Nuclear Steam Supply Shut-Off
A2 Ability to (a) predict the impacts of the following on the PRIMARY CONTAINMENT ISOLATION SYSTEM/NUCLEAR STEAM SUPPLY SHUT-OFF; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: (CFR: 41.5 / 45.6)
09 System initiation

RO/SRO Rating: 3.6/3.7

Pedigree: New

Objective: LOI-CLS-LP-012, Objective 6
Given plant conditions, determine if a Group Isolation should occur.

Reference: None

Cog Level: Higher

Explanation: Four differential pressure transmitters sense flow through the main steam line flow elements to provide input of Main Steam line flow to the 4 RPS trip cabinets (one steam line flow per trip cabinet). It takes high steam flow sensed in at least 2 lines to close the MSIVs, and in all 4 steam lines for a full Group 1 isolation. The maximum dp to open the MSIVs is 200 psid to preclude equipment damage.

Distractor Analysis:

Choice A: Plausible because any high steam flow sensed in any 1 line completes the logic for high steam flow in RUN isolations. 50 psid is the normal dp allowed to open the MSIVs after isolation. 200 psid is the allowed dp for a more rapid recovery of the condenser. Therefore second part is correct.

Choice B: Plausible because any high steam flow sensed in any 1 line completes the logic for high steam flow in RUN isolations. 50 psid is the normal dp allowed to open the MSIVs after isolation.

Choice C: Correct Answer, see explanation.

Choice D: Plausible because logic is correct. 50 psid is the normal dp allowed to open the MSIVs after isolation.

SRO Basis: N/A

5. **Main Steam Line Flow High not in RUN (Unit-2 Only), Setpoint 30% (6.8 psid from calc) (See Table 12-2)**

Four differential pressure transmitters sense flow through the main steam line flow elements to provide input of main steam line flow to the 4 RPS trip cabinets (one steam line flow per trip cabinet). Unlike the high steam flow above, it takes high steam flow sensed in at least 2 lines to close the MSIVs, and in all 4 steam lines for a full Group 1 isolation. This isolation is intended to protect against a EHC malfunction causing bypass valve opening during plant operation when the Steam Line Low Pressure isolation is bypassed (not in RUN). This protective action is required on Unit 2 only due

4. **Main Steam Line Flow High, Setpoint 137% (176.3 psid from calc) (See Table 12-2)**

To detect steam line ruptures, sixteen differential pressure transmitters, four per main steam line provide input to the RPS trip cabinets. Each RPS cabinet has a trip unit for each steam line flow, such that if the setpoint is exceeded in any one steam line, all four RPS trip cabinets will send a trip signal to their respective Group 1 logic trip channel.

2OP-25:

5.2.2 Procedural Steps

NOTE:	It is preferable to obtain a differential pressure less than 50 psid before opening the MSIVs. The MSIVs may be opened when the differential is less than 200 psid to allow for more rapid recovery of the main condenser after a Group 1 isolation.

8.2.2 Procedural Steps

NOTE:	It is preferable to obtain a differential pressure less than 50 psid before opening the MSIVs. If desired, for a more rapid recovery, and approved by the Unit CRS, the MSIVs may be opened up to a differential of 200 psid.

CAUTION

Opening MSIVs at differential pressures greater than 200 psid will cause equipment damage.

21. 223002 2

Reactor Recirculation pumps have tripped due to a low reactor water level condition.

G31-F001, RWCU Inboard Isol Vlv, is Closed.
G31-F004, RWCU Outboard Isol Vlv, is Open.

Which one of the following identifies what the Group 3 Isolation Status Box on ERFIS will display in five minutes?

A. A green GROUP ISOL

B. A red NO GROUP ISOL

C. A yellow GROUP ISOL CMND

D. A green NO GROUP ISOL CMND

Answer: A

K/A:
223002 Primary Containment Isolation System/Nuclear Steam Supply Shut-Off
A4 Ability to manually operate and/or monitor in the control room: (CFR: 41.7 / 45.5 to 45.8)
05 SPDS/ERIS/CRIDS/GDS: Plant-Specific

RO/SRO Rating: 2.5/2.8

Pedigree: 2010-1 NRC Exam

Objective: LOI-CLS-LP-060-A, Objective 4d
Describe the methods used to do the following on the ERFIS/SPDS Computer:
Obtain Group Isolation status including valve position

Reference: None

Cog Level: Higher

Explanation: ERFIS relies on the isolation signal to determine if an isolation is required. Since RWCU did receive a signal, ERFIS will recognize a valid isolation signal with at least one valve closed in the penetration path and remain Green and display GROUP ISOL.

Distractor Analysis:

Choice A: Correct Answer, see explanation

Choice B: Plausible because this is what would be expected with an isolation signal and no valves closed.

Choice C: Plausible because the isolation signal and valve closure time has not expired and can be confused with an incomplete isolation of the penetration flow path (both valves not closed).

Choice D: Plausible because if the candidate does not recognize Recirc pump trip is LL2 (same as RWCU), then this would be indicated if no isolation signal present.

SRO Basis: N/A

SD-60:

Event Status	Display Message	Color Code	Condition
Inactive	NO GROUP ISOL CMND	Green	1. No isolation signal
Safe	GROUP ISOL	Green	1. Isolation signal 2. Valve closure time exceeded 3. At least one valve in each path closed
Caution	GROUP ISOL CMND	Yellow	1. Isolation signal 2. Valve closure time not exceeded
Alarm	NO GROUP ISOL	Red	1. Isolation signal 2. Valve closure time exceeded 3. No valve closed in a path

22. 233000 1

RHR is operating in the Fuel Pool Cooling Assist Mode with Fuel Pool Gates Removed IAW 1OP-17, Section 8.11.

Which one of the following identifies the power supply to RHR Pump 1B?

RHR Pump 1B is powered from Bus ________.

A. E1

B. E2

C. E3

D. E4

Answer: D

K/A:
233000 Fuel Pool Cooling and Clean-up
K2 Knowledge of electrical power supplies to the following: (CFR: 41.7)
02 RHR pumps

RO/SRO Rating: 2.8/2.9

Pedigree: New

Objective: LOI-CLS-LP-017, Objective 17a
List the normal and emergency power source for the following:
a. RHR Pumps

Reference: None

Cog Level: Fund

Explanation: Power supplies for RHR Pumps is listed in the Notes Section. RHR Pump 1B is powered from E4.

Distractor Analysis:

Choice A: Plausible because the E busses power the RHR pumps. Student must know the scheme for RHR pump power.

Choice B: Plausible because the E busses power the RHR pumps. Student must know the scheme for RHR pump power.

Choice C: Plausible because the E busses power the RHR pumps. Student must know the scheme for RHR pump power.

Choice D: Correct Answer, see explanation.

SRO Basis: N/A

The power supply for the pumps, along with the associated diesel generator and power supply division, is listed below (also see Figure 17-2B):

RHR Pump	1A/2A	1B/2B	1C/2C	1D/2D
Power Source	E3	E4	E1	E2
Diesel	#3	#4	#1	#2
Division	I	II	I	II

23. 239002 1

Which one of the following completes the statement below?

SRVs K and L are not included in the opening sequence to stabilize reactor pressure in RVCP due to the close proximity of their discharges to:

A. each other

B. HPCI and RCIC steam exhausts

C. RHR Loop A and B pump suctions

D. HPCI and RCIC pump torus suctions

Answer: B

K/A:

239002 Safety Relief Valves

K4 Knowledge of RELIEF/SAFETY VALVES design feature(s) and/or interlocks which provide for the following: (CFR: 41.7)

04 Ensures even distribution of heat load to suppression pool, and adequate steam condensing

RO/SRO Rating: 3.4/3.6

Pedigree: Bank

Objective: LOI-CLS-LP-300-D, Objective 7
Given plant conditions, the Reactor Vessel Control Procedure, and which steps have been completed, determine the required operator actions.

Reference: None

Cog Level: Fund

Explanation: IAW discussion with NRC, write to manual/procedural actions or bases for actions. See Notes Section for basis.

Distractor Analysis:

Choice A: Plausible because this would be a concern if they were close to each other because they would heat up the Torus unevenly. Examinee must know relative locations of SRV discharge.

Choice B: Correct Answer, see explanation.

Choice C: Plausible because heating of Torus in locations of RHR suction could cause NPSH concerns.

Choice D: Plausible because heating of Torus in locations of HPCI and RCIC suctions could cause NPSH concerns.

SRO Basis: N/A

4.3.2 Suppression Pool

The suppression pool serves as the heat sink for the steam discharged by the ADS valves. Long term operation of relief valve(s) should be avoided as suppression pool temperature will become elevated (approximately 2-3°F/min. with one valve open). The periodic pulsation of the steam jet at the relief valve discharge pipe may cause severe continuous vibration of the suppression pool, possibly resulting in structural damage. The distribution of the SRV tailpipes is designed to minimize localized heat loads in the suppression pool and takes into consideration the steam exhaust from the HPCI and RCIC turbines.

Reactor Vessel Parameters

SRV Opening Sequence

Parameter Value

B, F, E, G, A, C, J, H, D

Source for Value

Prints D-27092 (02792)

Specific Usage

PSTG/PSSAG (specific steps):

- RC/P-2
- C3-1

EOPs/SAMGs:

- 1(2)EOP-01-RSP
- 1(2)EOP-01-RVCP
- 1(2)EOP-01-LPC
- 0EOP-01-STCP

Basis for Value Selection

An opening sequence for SRVs is specified for the operator to use in order to more evenly distribute the heat load in the suppression pool. SRVs "K" and "L" are not listed due to the close proximity of their discharges to the HPCI and RCIC steam exhausts. In situations where prompt reduction of reactor pressure is required, adherence to this opening sequence would be unwarranted.

24. 241000 1

During a Unit Two shutdown, the Turbine Bypass Valves failed to open following a Scram. The Turbine Bypass Valve Opening Jack was positioned at 20% open to facilitate a cooldown.

Which one of the following identifies how the Bypass Valves and Bypass Valve Opening Jack are expected to respond if condenser vacuum is broken?

The Bypass Valves will close when condenser vacuum lowers to ___(1)___ inches Hg.

The Bypass Valve Opening Jack will ___(2)___.

A. (1) 7
   (2) run back to 0% with no operator action.

B. (1) 7
   (2) remain at 20% until lowered by the operator.

C. (1) 10
   (2) run back to 0% with no operator action.

D. (1) 10
   (2) remain at 20% until lowered by the operator.

Answer: B

K/A:
241000 Reactor/Turbine Pressure Regulating System
K6 Knowledge of the effect that a loss or malfunction of the following will have on the REACTOR/TURBINE PRESSURE REGULATING SYSTEM: (CFR: 41.7 / 45.7)
10 Bypass valves

RO/SRO Rating: 3.6/3.7

Pedigree: Bank

Objective: LOI-CLS-LP-026.3, Objective 10h
For each of the following plant conditions, analyze the condition and predict the effect it will have on the main Turbine EHC electrical system:
h. Failed open/closed Bypass Valves

Reference: None

Cog Level: Fund

Explanation: The Main Turbine will trip at 7 inches Hg vacuum. The Bypass Valve Jack is a potentiometer which does not change automatically. This question is based on actual plant events when the BPV Jack was left at 20% and during a subsequent startup, when condenser vacuum was established, the BPVs came open.

Distractor Analysis:

Choice A: Plausible because part 1 is correct. Part 2 is the way the automatic Bypass Valve demand signal works, but not the Jack.

Choice B: Correct Answer, see explanation.

Choice C: Plausible because the MSIVs close at 10 inches Hg for part 1, which is the closest Automatic Action preceding Turbine Tip, and part 2 is the way the automatic Bypass Valve demand signal works, but not the Jack.

Choice D: Plausible because the MSIVs close at 10 inches Hg for part 1, which is the closest Automatic Action preceding Turbine Tip, and part 2 is correct.

SRO Basis: N/A

SD-26.3:

The bypass valve flow signal from the Bypass Valve Summing Junction is gated with a motor driven potentiometer Bypass Valve Jack at a high value gate. This allows manual opening of bypass valves. (During reactor shutdown, it may be desirable to open a bypass valve(s) to control vessel cooldown).

From the high value gate, the bypass valve steam demand passes to a low value gate (LVG) which serves two functions:

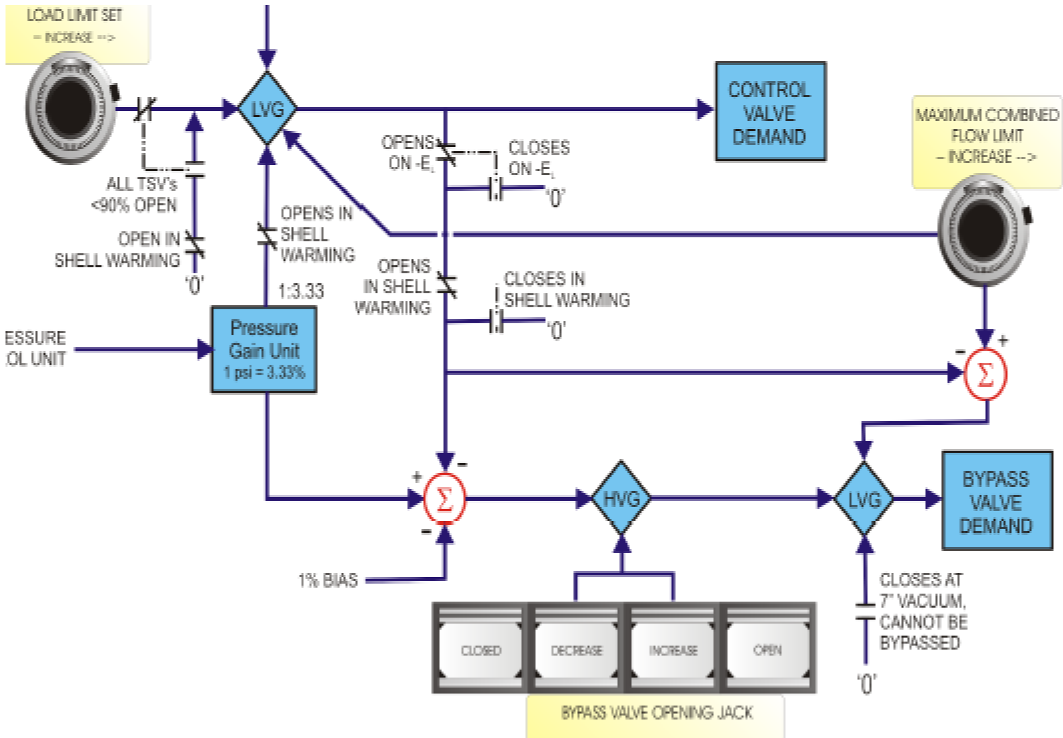
– Prevents opening the bypass valves when condenser vacuum is low which protects the condenser from damage.
– Prevents concurrent opening of the bypass and control valves to a value greater than that permitted by the max combined flow limiter.

OAOP-37.0 (Automatic Actions):

3. **IF** condenser vacuum lowers to 10 inches Hg,
THEN the following valves receive a close signal:

- MSIVs (MSIV closure in RUN mode results in reactor scram) .... ☐
- B21-F016 (Main Steam Line Drain Inbd Isol Vlv) and B21-F019 (Main Steam Line Drain Otbd Isol Vlv) .... ☐
- B32-F019 (Sample Inbd Isol Vlv), and B32-F020 (Sample Otbd Isol Vlv) .... ☐

4. **IF** condenser vacuum lowers to 7 inches Hg,
THEN main turbine bypass valves receive a close signal. .... ☐


LOAD LIMIT SET
-- INCREASE -->
LVG
CONTROL VALVE DEMAND
OPENS ON -E
CLOSES ON -E
'0'
MAXIMUM COMBINED FLOW LIMIT
-- INCREASE -->
ALL TSV's <90% OPEN
OPEN IN SHELL WARMING
'0'
OPENS IN SHELL WARMING
1:3.33
OPENS IN SHELL WARMING
CLOSES IN SHELL WARMING
'0'
ESSURE
OL UNIT
Pressure Gain Unit
1 psi = 3.33%
HVG
LVG
BYPASS VALVE DEMAND
1% BIAS
CLOSES AT 7" VACUUM, CANNOT BE BYPASSED
'0'
CLOSED
DECREASE
INCREASE
OPEN
BYPASS VALVE OPENING JACK

25. 256000 1

With Unit Two operated at rated power, the 2A Feedwater Heater level reaches the Hi Hi Level setpoint due to a failed Feedwater Heater level control valve.

Which one of the following completes the statement below?

The Moisture Removal Valves will open to drain the extraction steam lines to the ___(1)___, and final feedwater temperature to the reactor will ___(2)___.

A. (1) Condenser
   (2) increase

B. (1) Condenser
   (2) decrease

C. (1) Heater Drain Deaerator
   (2) increase

D. (1) Heater Drain Deaerator
   (2) decrease

Answer: B

K/A:
256000 Reactor Condensate System
A3 Ability to monitor automatic operations of the REACTOR CONDENSATE SYSTEM including: (CFR: 41.7 / 45.7)
07 Feedwater heater level

RO/SRO Rating: 2.9/2.9

Pedigree: 2008 NRC Exam

Objective: LOI-CLS-LP-034, Objective 7c
Given plant conditions, describe the automatic feedwater heater level control actions for the following: c. High-High Feedwater Heater/Heater Drain Deaerator level

Reference: None

Cog Level: High

Explanation: A high-high level condition in the 2A FW Heater will cause the associated MRVs to open, directing 11th stage extraction steam to the condenser, and allowing the extraction line check valves to close. Feedwater heating is lost for this heater causing overall feedwater temperature to decrease.

Distractor Analysis:

Choice A: Plausible because part 1 is correct, but feedwater temperature will decrease.

Choice B: Correct Answer, see explanation

Choice C: Plausible because MRVs open to the condenser; other LP FW heaters direct flow to the HDD

Choice D: Plausible because MRVs open to the condenser; other LP FW heaters direct flow to the HDD

SRO Basis: N/A
NRVs open when turbine is reset, MRVs only close when closed by the operator.

SD-34:

In the "auto" mode the signal is maintained based on the requested level (red pointer) and the actual deviation from setpoint (black pointer). The drain valves open and close to maintain level between the high and low level annunciators. If the air signal decreases to ≃ < 3 psig which is indicative of a high level, an annunciator in the Control Room illuminates and the following occurs:

- #1A/1B FW Heater level switch HD-LSHH-216/217 actuates and opens the Emergency Drain valve to the Condenser (LV-216/217)
- #2A/2B FW Heater level transmitter HD-LT-61/64 opens the high level drain to the Condenser HD-LV-61/64.
- #5A/5B FW Heater level transmitter HD-LT-83/87 opens the high level drain to the HDD tank HD-LV-83/87.

The #3 and #4 FW Heaters are not equipped with high level or emergency drains.

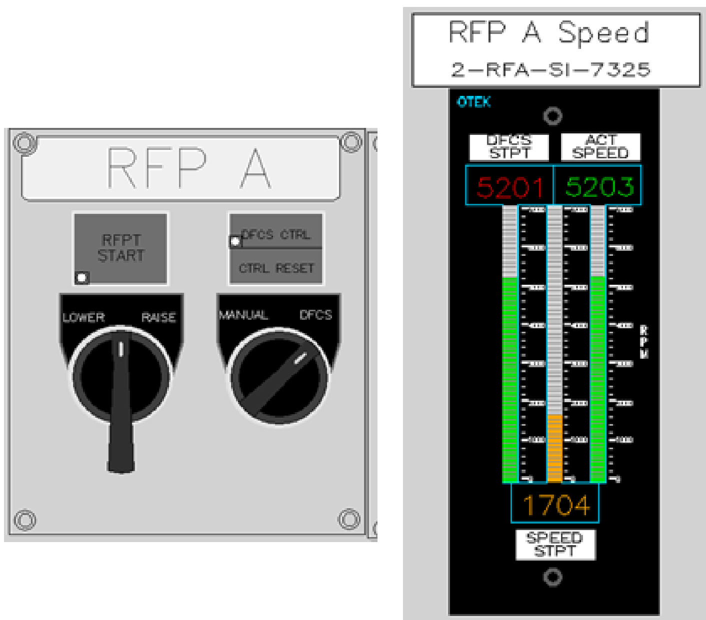
If level continues to increase and the HI-HI level switches actuate a common annunciator (UA-04 1-9 FW Heater Level High Ext Trip) in the Control Room illuminates and the following occur (Figures 34-4 and 8).

26. 259002 1

Unit Two is operating at 100% power. Reactor Feed Pump (RFP) 2A is operating in automatic DFCS control when the following alarm is received:

UA-13 (6-5) *RFP A Control Trouble*

The RO observes the following indications for RFPT 2A on XU-1 and P603 respectively:



Which one of the following identifies how RFP 2A will respond and what actions are required to control RFP 2A under this condition?

RFP 2A will ____(1)____.

The RO can manually change the speed IAW 0AOP-23, Condensate/Feedwater System Failure, by using the LOWER/RAISE ____(2)____.

A. (1) remain at the current speed
   (2) Speed Control Switch on XU-1.

B. (1) remain at the current speed
   (2) speed demand pushbuttons at RFP 2A panel display station on P603.

C. (1) automatically lower to 1704 RPM
   (2) Speed Control Switch on XU-1.

D. (1) automatically lower to 1704 RPM
   (2) speed demand pushbuttons at RFP 2A panel display station on P603.

Answer: A

K/A:
259002 Reactor Water Level Control System

A2 Ability to (a) predict the impacts of the following on the REACTOR WATER LEVEL CONTROL SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: (CFR: 41.5 / 45.6)

06 Loss of controller signal output

RO/SRO Rating: 3.3/3.4

Pedigree: 2004 NRC Exam

Objective: LOI-CLS-LP-032.2, Objective 6d
Given plant conditions, determine the response of the DFCS to the following events:
d. Loss of signal interface between controllers and processor

Reference: None

Cog Level: High

Explanation: Indications are consistent with a loss of DFCS signal (DFCS CTRL white light out). Operator is required to change RFPT speed by using the LOWER/RAISE control switch at XU-1.

Distractor Analysis:

Choice A: Correct Answer, see explanation

Choice B: Plausible because RFP will remain at the current speed. Both Manual and Auto signals on Panel P603 will not function. If the speed controller on P603 had failed, this would be the correct way to control feed pump speed.

Choice C: Plausible because DFCS demand is 1704 rpm, so candidate must know that RFP will stay at last demand signal and not track lowering demand. DFCS is already in manual.

Choice D: Plausible because DFCS demand is 1704 rpm, so candidate must know that RFP will stay at last demand signal and not track lowering demand. Both Manual and Auto signals on Panel P603 will not function. If the speed controller on P603 had failed, this would be the correct way to control feed pump speed.

SRO Basis: N/A

MANUAL/DFCS switch on the XU-1 Panel. Placing the switch in MANUAL places the RAISE/LOWER switch on the XU-1 panel in control of the feed pump.

Whenever DFCS is controlling feed pump speed the white DFCS CTRL light above the MANUAL/DFCS switch on the XU-1 Panel is on.

4.2.7 Signal Failure to the Feed Control System

The normal control signal into the pump control stations and the startup valve demand is between 4 ma to 20 ma. If the signal goes outside this range the redundant signal takes over to control the associated component without any change.

If both of the redundant signals to the pumps or valve demand are outside the normal range a "FW Control Signal Failure" is sensed on detectors C32-K607A/B. The feed pump governor controls will lock the feed pump speed at the last signal called for. The SULCV will respond to the failed signal as if it were valid.

The turbine speed controller will "Lock Out" the reactor feed pump at the speed demand signal last called for. The Woodward controls automatically shift to manual control. The loss of the control signal will be annunciated in the Control Room.

A control signal loss could be a result of Foxboro module failures, or a loss of AC/DC power. This results in the K2A/K2B relays energizing causing the amber lights (DS1A/DS1B) at the Unit 1 P603 to light.

Upon restoration of the control signal, the controllers must be manually reset. On the XU-1 Panel, the MANUAL/DFCS control switch must be placed in MANUAL then the CTRL RESET pushbutton depressed.

27. 261000 1

While venting containment on Unit Two IAW 2OP-10, Section 6.3.2, Venting Containment Via SBGT, reactor water level lowers to 100 inches. SBGT 2B is tagged out for maintenance.

Subsequently SBGT 2A trips and cannot be restarted.

Which one of the following identifies the effect the loss of SBGT will have on HPCI operation?

A. Contaminated steam will be leaked in the vicinity of the HPCI Turbine.

B. SBGT train inlet and outlet dampers must be manually opened to support HPCI operation.

C. Buildup of non-condensable gases in the Barometric Condenser can raise HPCI exhaust pressure.

D. SBGT must be re-aligned from Containment to the Reactor Building to support HPCI operation.

Answer: A

K/A:
261000 Standby Gas Treatment System
K3 Knowledge of the effect that a loss or malfunction of the STANDBY GAS TREATMENT SYSTEM will have on following: (CFR: 41.7 /45.6)
04 High pressure coolant injection system: Plant-Specific

RO/SRO Rating: 3.1/3.1

Pedigree: New

Objective: LOI-CLS-LP-019, Objective 3u
Given plant conditions, predict how the HPCI System will respond to the following events:
u. Standby Gas Treatment System failure

Reference: None

Cog Level: Fund

Explanation: HPCI operations without SBGT result in loss of HPCI Barometric Condenser Vacuum Pump discharge flow path, causing leakage of contaminated steam in the vicinity of the HPCI Turbine.

Distractor Analysis:

Choice A: Correct Answer, see explanation

Choice B: Plausible because dampers associated with Unit 2 SBGT receive an automatic open signal, but not Unit 1.

Choice C: Plausible because although loss of SBGT will cause a buildup of non-condensible gases in the Barometric Condenser, this is not an exhaust path.

Choice D: Plausible because the flow path must be re-aligned, but SBGT will realign itself to the Reactor Building if an initiation signal occurs while venting Containment.

SRO Basis: N/A

4.3 Interrelationships With Other Systems

4.3.1 High Pressure Coolant Injection (HPCI)

The SBGT System provides a flow path for the HPCI Barometric Condenser Vacuum Pump discharge through the post-LOCA vent valves V8 and V9. HPCI can operate without the Barometric Condenser.

Failure of SBGT may result in loss of HPCI Barometric Condenser

Vacuum Pump discharge flow path, causing some leakage of contaminated steam in the vicinity of the HPCI Turbine.

Unit 2 ONLY

The dampers associated with Unit 2 SBGT System will receive automatic open signals when an initiation signal is received.

When the train inlet and Fan discharge dampers are 96% open the Fan will start and the Heater will energize.

The SBGT will realign itself to the Reactor Building if an initiation signal occurs while venting the containment.

28. 262001 1

Unit One is operating at rated power when a Turbine/Generator trip results in tripping the Main Generator Backup Lockout relays.

Which one of the following identifies the expected response of Bus 1D and the Diesel Generators (DGs)?

Bus 1D ___(1)___ fast transfer to the SAT.

All DGs ___(2)___.

A. (1) will
   (2) auto start

B. (1) will
   (2) remain in standby

C. (1) will not
   (2) auto start

D. (1) will not
   (2) remain in standby

Answer: B

K/A:
262001 A.C. Electrical Distribution
A1 Ability to predict and/or monitor changes in parameters associated with operating the A.C. ELECTRICAL DISTRIBUTION controls including: (CFR: 41.5 / 45.5)
05 Breaker lineups

RO/SRO Rating: 3.2/3.5

Pedigree: Bank

Objective: LOI-CLS-LP-050.1, Objective 18b
Describe the major steps and interlocks necessary to perform the following bus transfer operations: b. UAT to the SAT

Reference: None

Cog Level: High

Explanation: Busses C and D are normally aligned to the UAT and have auto transfers on backup lockout. The B Bus is normally lined up to the SAT and has no auto transfer capability. DGs do not auto start on a backup lockout.

Distractor Analysis:

Choice A: Plausible because first part is correct. DGs auto start on Main Generator Differential and Primary lockouts, but not Backup lockout.

Choice B: Correct Answer, see explanation.

Choice C: Plausible because although Bus 1C or Bus 1D would auto transfer, Bus 1B will not. DGs auto start on Main Generator Differential and Primary lockouts, but not Backup lockout.

Choice D: Plausible because although Bus 1C or Bus 1D would auto transfer, Bus 1B will not. Second part is correct.

SRO Basis: N/A

From SD-50.1:

2. **4160 Bus 1B/2B** (Figures 7, 8, and 9)

The B Bus is normally connected to the SAT during all plant conditions. However if power is available, the bus can be connected to the UAT at the discretion of operations management. The bus does not have an automatic bus transfer from the UAT to the SAT, thus if the B Bus is aligned to the UAT and a reactor scram and/or turbine trip occurs, a loss of recirc pumps will occur. The bus does have a scheme that is capable of transferring the bus from the normal source, (the SAT), to the alternate source, (the UAT) and back without interrupting the power to the bus. This scheme is said to be a **manually initiated, automatically executed fast bus transfer**. During this transfer both power sources are paralleled for a period of 23 to 30 milliseconds.

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3. **4160 1C/2C and 1D/2D Buses** (Figures 10, 11, 12, and 13)

The C and D Buses for each unit are normally connected to the SAT during shutdown conditions. When power is available the bus is aligned to the UAT. The bus has a transfer scheme that is capable of transferring the bus from the shutdown source (the SAT), to the alternate source (the UAT), and back without interrupting the power to the bus. This scheme is said to be a **manually initiated**

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automatically executed fast bus transfer. During this transfer both power sources are energized and paralleled for a period of 23 to 30 milliseconds.

The C and D Buses are provided with an **automatically initiated, automatically executed, quick dead bus transfer**. This transfer will take place if the buses are connected to the UAT and the main generator trips and locks out. The buses aligned to the UAT will automatically transfer to the SAT (if no faults are present on the SAT and any pair of 230 KV bus breakers is closed to cross-connect the A and B buses). The bus and its loads are disconnected from the UAT for a period of 30 to 72 milliseconds before the buses connect to the SAT. No loads should be lost due to an undervoltage condition in this short time frame since undervoltage relaying is time delayed.

From SD-50:

(6) Four diesel generators auto start for the Main Generator Differential Lockout or the Generator/Transformer Primary Lockout. They do not auto start for a Generator/Transformer Backup Lockout.

29. 262002 1

During Station Blackout conditions, Emergency Buses cannot be cross-tied. UPS loads are on the primary inverter with the inverter input from 125/250 VDC distribution. DC voltage is slowly lowering due to loss of battery chargers.

Which one of the following completes the statements below?

The inverter DC input breaker will trip when inverter ___(1)___ drops to a predetermined value.

IAW SBO Procedures, UPS loads must be ___(2)___.

A. (1) DC input voltage
   (2) de-energized

B. (1) DC input voltage
   (2) transferred to the alternate source.

C. (1) AC output voltage
   (2) de-energized.

D. (1) AC output voltage
   (2) transferred to the alternate source.

Answer: A

K/A:

262002 Uninterruptable Power Supply (A.C./D.C.)

A2 Ability to (a) predict the impacts of the following on the UNINTERRUPTABLE POWER SUPPLY (A.C./D.C.) ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: (CFR: 41.5 / 45.6)

01 Under voltage

RO/SRO Rating: 2.6/2.8

Pedigree: Bank

Objective: LOI-CLS-LP-052, Objective 7b

Predict the impact(s) of the following on the UPS System: Under Voltage.

Reference: None

Cog Level: High

Explanation: Station Blackout procedure requires UPS be load stripped if E Buses cannot be cross-tied in order to conserve battery capacity for the coping period. Inverter input breaker trips if inverter input (whether from rectifier or directly from DC) drops to 214 VDC.

Distractor Analysis:

Choice A: Correct answer, see explanation.

Choice B: Plausible because part 1 is correct, second part would be true is any power were available.

Choice C: Plausible because this is the output of the inverter which is upstream of the input voltage. Part 2 is correct.

Choice D: Plausible because this is the output of the inverter which is upstream of the input voltage. Second part would be true is any power were available.

SRO Basis: N/A

SD-52:

TABLE 52-1
Vital UPS Power Supplies

	UNIT 1	UNIT 2
480 VAC SUPPLY PRIMARY UNIT STANDBY UNIT	MCC 1CA (E5) MCC 1CB (E6)	MCC 2CA (E7) MCC 2CB (E8)
250 VDC SUPPLY PRIMARY UNIT STANDBY UNIT	DC SWBD 1A DC SWBD 1B	DC SWBD 2A DC SWBD 2B
ALTERNATE AC	MCC 1CB (E6)	MCC 2CB (E8)

3.0 INSTRUMENTATION AND CONTROL

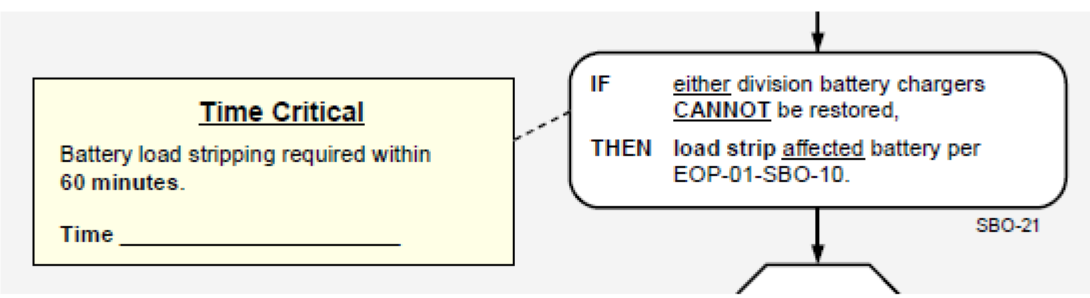
3.1. Component Control

3.1.1. Vital UPS (See Figures 52-9 and 52-10)

The UPS inverter receives the DC power input from the rectifier or the alternate DC source through a DC input breaker and a filter network. The capacitors in the filter network before the DC input breaker must be charged prior to closing the breaker. If the capacitors are not fully charged the DC input breaker (CB 101) may trip on undervoltage. A light (DS101) above the charge/discharge switch indicates when the capacitor is charged. When the unit is shut down it takes several seconds for the capacitor to discharge, and this causes a time delay in the decrease of DC volts to zero.

The inverter converts the DC input into a three-phase 120/208 Vac, 60 hz. output. The inverter performs this function using solid state circuitry. A filter network conditions the output voltage of the inverter.

If inverter output is lost the inverter load is automatically shifted to the alternate AC source. If the Return Mode Switch is in AUTO, (Normal Position) the static transfer switch will retransfer to its normal position when the inverter output recovers. If the Return Mode Switch is in the MAN position, the static transfer switch can be manually transferred back to the inverter using the Reset Switch.



4.2 Supplementary Actions

1. Loss of Battery Chargers:

   a. **Monitor** 125V and 24V DC battery voltages. .... ☐

   b. <u>**IF**</u> power has been removed from the battery chargers for greater than 1 hour,
   <u>**THEN**</u> **remove** selected loads from the battery based on 0OI-50, 125/250 and 24/48 VDC Electrical Load List and Unit CRS direction. .... ☐

   c. Before 125V DC battery voltage reaches the low voltage limit of 105 volts, **remove** loads as directed by the Unit CRS as necessary to maintain battery voltage greater than 105 volts. .... ☐

4.1.4 OWP-51/1 Removal Of 125 VDC Battery/Battery Charger From Service

This procedure provides guidance on transferring DC loads, including distribution panels, control power and UPS power conversion units to an alternate source of power if a battery or battery charger is to be removed from service. Performance of this procedure minimizes loads that are lost if a battery is to be removed from service, or minimizes battery discharge rate when a charger is removed from service. (With a charger out of service, the battery terminal voltage must **NOT** be allowed to drop below 105 VDC or the battery output breaker must be opened to prevent battery cell damage. This is required by the APP for 250 VDC Undervoltage.)

DEVICE/SETPOINTS

Undervoltage Devices

BATT 2A-1 DSLV 120T-01	129V DC
BATT 2A-2 DSLV 120T-01	129V DC

POSSIBLE PLANT EFFECTS

1. The DC equipment supplied from Battery Bus 2A-1(2A-2) may be damaged due to operation at reduced voltage.
2. If the battery charger is inoperable for an extended period of time, the plant will have to be shut down because the batteries have a rating of 1200 ampere hours for 8 hours only.
3. Loss of a battery charger or battery may result in a technical specification LCO.

30. 262002 2

The indications and status of the UPS system at the primary and standby inverters are as follows:

	Primary Inverter	Standby Inverter
Load on UPS light	Off	Off
Load on Inverter light	Off	On
Load on Alternate light	On	Off
Alt Source Failure light	Off	Off
Manual Bypass switch	Norm	Bypass Test

Which one of the following identifies the current status of UPS system loads?

A. De-energized

B. Energized from the primary inverter

C. Energized from the standby inverter

D. Energized from the alternate source

Answer: D

K/A:
262002 Uninterruptable Power Supply (A.C./D.C.)
A3 Ability to monitor automatic operations of the UNINTERRUPTABLE POWER SUPPLY (A.C./D.C.) including: (CFR: 41.7 / 45.7)
01 Transfer from preferred to alternate source

RO/SRO Rating: 2.8/3.1

Pedigree: 2014 NRC Exam

Objective: LOI-CLS-LP-052, Objective 5
Given plant conditions, determine the lineup of the primary UPS, the Standby UPS, and their reserve sources.

Reference: None

Cog Level: High

Explanation: The UPS system is normally aligned such the primary inverter is powering UPS loads. The standby inverter is energized but bypassed with the Manual Bypass switch in Bypass Test. The static transfer switch of the Primary inverter (and also the Standby inverter) is receiving an input from the alternate (hard) source. The indications given for the Standby inverter are normal. The load on inverter light is lit because the static transfer switch is aligned to the inverter output, but this output is bypassed. The indications on the primary inverter are normally Load on UPS and Load on Inverter both lit. If both lights are out and the Manual Bypass switch is in Normal the static transfer switch has transferred to alternate. With the Standby inverter in Bypass Test this alternate source is the hard source. Since the Alt

Source Failure lights are out, UPS loads are energized from the alternate source.

Distractor Analysis:

Choice A: Plausible because this could be correct in the Alt Source Failure lights were lit.

Choice B: Plausible because this is the normal status of UPS and would be correct if the Load on UPS and Load on Inverter lights were lit on the Primary inverter

Choice C: Plausible because this could be correct in the if the Manual Bypass Switch on the Standby Inverter were in Normal instead of Bypass Test

Choice D: Correct Answer, see explanation.

SRO Basis: N/A

Reference: SD-52, Sections 2.2.1, 3.1.1, Table 52-3 and 52-4

The UPS system is normally aligned as follows:

The primary unit is in service with its output connected to the UPS distribution system. Its rectifier receives 480 VAC power from a Division I emergency distribution panel. A 250 VDC from DC Switchboard 1A (2A) is supplied in parallel with the rectifier output to power the inverter should the normal AC source be lost. The alternate AC source from the standby unit is available at the static transfer switch to pick up the loads if the inverter output is lost.

The standby unit is also energized with its 480 VAC input supplied from a Division II emergency distribution panel and its 250 VDC supplied from DC Switchboard 1B (2B); but, its output is bypassed by its manual bypass switch and its alternate AC input is being supplied directly to the primary unit. The standby unit receives its alternate AC source of power from the same Division II distribution panel as its rectifier AC input through a 480-120/208 VAC transformer. The standby units alternate AC input is also referred to as the hard source.

If inverter output is lost the inverter load is automatically shifted to the alternate AC source. If the Return Mode Switch is in AUTO, (Normal Position) the static transfer switch will retransfer to its normal position when the inverter output recovers. If the Return Mode Switch is in the MAN position, the static transfer switch can be manually transferred back to the inverter using the Reset Switch.

31. 263000 1

Which one of the following completes the statements below regarding 125/250 VDC Station Distribution?

During a float charge, the charger output voltage to the battery will be at a ___(1)___ voltage than when in the equalize mode.

The 125 VDC batteries are sized to supply emergency power at a 150 amp rate for ___(2)___ hours.

A. (1) lower
   (2) 8

B. (1) lower
   (2) 10

C. (1) higher
   (2) 8

D. (1) higher
   (2) 10

Answer: A

K/A:
263000 D.C. Electrical Distribution
A1 Ability to predict and/or monitor changes in parameters associated with operating the D.C. ELECTRICAL DISTRIBUTION controls including: (CFR: 41.5 / 45.5)
01 Battery charging/discharging rate

RO/SRO Rating: 2.5/2.8

Pedigree: Mod

Objective: LOI-CLS-LP-051, Objective 2a and b
Define the following terms:
a. Float Charge
b. Equalizing Charge

Reference: None

Cog Level: Fund

Explanation: The float mode voltage for the 125 VDC battery charger is ~135 volts while in equalize the charger output is ~140 volts. The design of the batteries is for 150 amps for 8 hours.

Distractor Analysis:

Choice A: Correct Answer, see explanation

Choice B: Plausible because lower is correct, and the Caswell Beach batteries are rated for 10 hours.

Choice C: Plausible because the examinee must differentiate between float vs. equalize and the 8 hours is correct.

Choice D: Plausible because the examinee must differentiate between float vs. equalize and the Caswell Beach batteries are rated for 10 hours

SRO Basis: N/A

The chargers of the 125/250 VDC system are solid state constant voltage magnetic amplifier controlled. The chargers are of sufficient capacity to allow charging of it's respective battery from a design minimum charged state to the fully charged state in approximately 8 hours while carrying all of it's DC loads associated with normal plant operation. Charger current is limited to 125% of rated to protect the charger in the event of connection to a fully discharged battery. The charger AC input breaker will trip on a high DC output voltage (143 VDC) to prevent overcharging batteries. Battery terminal voltage must remain greater than or equal to 130 VDC on float charge to meet Tech Spec Operability.

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2.0 COMPONENT DESCRIPTION/DESIGN DATA

2.1 Battery Capacity Ratings

All of the battery systems (with the exception of the Caswell Beach Microwave) have a design Ampere-Hour capacity rating which defines the batteries expected lifetime, in hours, based upon a given continuous loading, in amperes. It should be noted that this is merely a reference number and that battery lifetime is shortened if it is discharged at a higher rate or lengthened if discharged at a lower rate. The individual battery capacities are:

BATTERY SYSTEM	AMP-HOUR RATING
125/250 VDC Station (each division)	1200 AMP-HOURS at a 150 amp rate for 8 hours
24/48 VDC Station (each division)	600 AMP-HOURS at a 75 amp rate for 8 hours
125 VDC Caswell Beach	200 AMP-HOURS at a 20 amp rate for 10 hours

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There is no direct indication of the status of the battery charger; i.e., whether it is in the float charge or equalizer charge mode. If in the float charge mode the volt meter should read approximately 135 VDC. If in the equalizer charge mode the meter should read approximately 140 VDC.

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2012 Question:
Modified question changed from equalize to float charge.

Which one of the following completes the statements below regarding 125/250 VDC Station Distribution?

During an equalize charge, the charger output voltage to the battery will be at a ____(1)____voltage than when in the float mode.

The 125 VDC batteries are sized to supply emergency power at a 150 amp rate for ____(2)____ hours.

A. (1) lower
   (2) 8

B. (1) lower
   (2) 10

C. (1) higher - Correct Answer
   (2) 8

D. (1) higher
   (2) 10

32. 264000 1

As a result of a Loss of Offsite Power (LOOP), DG4 is running and tied to Bus E4.

Which one of the following loads loses power as the result of a trip of DG4?

A. RHR Pump 2D

B. Core Spray Pump 2B

C. Nuclear Service Water Pump 2A

D. Conventional Service Water Pump 2C

Answer: B

K/A:
264000 Emergency Generators (Diesel/Jet)
K3 Knowledge of the effect that a loss or malfunction of the EMERGENCY GENERATORS (DIESEL/JET) will have on following: (CFR: 41.7 / 45.4)
03 Major loads powered from electrical buses fed by the emergency generator(s)

RO/SRO Rating: 4.1/4.2

Pedigree: New

Objective: LOI-CLS-LP-039, Objective 3a
Given plant conditions, determine if EDG will trip.
LOI-CLS-LP-050.1, Objective 9g
List the major equipment/loads on each of the following 4160 VAC buses:
g. E4

Reference: None

Cog Level: Comp.

Explanation: DG4 provides power to E4, which powers the loads listed in the Notes section.

Distractor Analysis:

Choice A: Plausible RHR Pumps 1D and 2D are powered from E2, but sequentially may be considered to be E4 and DG4 loads.

Choice B: Correct Answer, see explanation.

Choice C: Plausible because Nuclear Service Water pumps are powered from the E-Busses and DGs on loss of normal power supplies.

Choice D: Plausible because the CSW Pumps are powered from the E-Busses and DGs on loss of normal power supplies.

SRO Basis: N/A

TABLE 50.1-2
Page 3 of 4
4160 VAC Distribution Power Supplies

	E3	**E4**	**E1**	**E2**
ECCS				
RHR Pumps	1A/2A	1B/2B	1C/2C	1D/2D
RHR Injection Valves	1A (E7)	1B (E8)	2A (E5)	2B (E6)
Core Spray Pumps & Valves	2A	2B	1A	1B
Service Water				
RHR SW	1A/2A	1B/2B	1C/2C	1D/2D
NSW	2A	2B	1A	1B
CSW	2A	1A/2B	1B/2C	1C
Power Supplies				
BOP Bus	2D	2C	1D	1C
Diesel Generator	DG3	DG4	DG1	DG2
Cross-Tie	E1	E2	E3 or E2*	E4 or E1*
480 VAC	E7	E8	E5	E6
Other				
CRD Pumps	2A	2B	1A	1B
Fire Pump Feed		Alternate		Normal

* ASSD Only

33. 264000 2

DG1 was running in Control Room Manual for the performance of 0PT-12.2A, No. 1 Diesel Generator Monthly Load Test, and loaded to 2100 KW.

Subsequently off-site power was lost.

Which one of the following completes the statements below after the system has stabilized?

The DG1 governor is currently in ___(1)___ mode of operation.

DG1 frequency is slightly ___(2)___ 60 Hz.

A. (1) droop
   (2) less than

B. (1) droop
   (2) greater than

C. (1) isochronous
   (2) less than

D. (1) isochronous
   (2) greater than

Answer: D

K/A:
264000 Emergency Generators (Diesel/Jet)
K5 Knowledge of the operational implications of the following concepts as they apply to EMERGENCY GENERATORS (DIESEL/JET): (CFR: 41.5 / 45.3)
05 Paralleling A.C. power sources

RO/SRO Rating: 3.4/3.4

Pedigree: Bank

Objective: AOI-CLS-LP-39, Objective 03e
Describe the operation of the below listed EDG components: Voltage adjust rheostat.

Reference: None

Cog Level: Fund

Explanation: Any auto start signal when in the manual mode will cause the EDG controls to automatically revert to isochronous mode. If this occurs during load testing and an E bus undervoltage condition exists the EDG will automatically tie back onto the bus in the isochronous mode. The EDG may tie onto the bus at an elevated frequency.

Distractor Analysis:

Choice A: Plausible because EDG will be in the droop mode during testing. Frequency can vary to 0-3 hertz.

Choice B: Plausible because EDG will be in the droop mode during testing. Second part is correct.

Choice C: Plausible because EDG will transfer to the isochronous mode if an auto start signal occurs.

Choice D: Correct answer, see explanation.

SRO Basis: N/A

SD-39:

Any auto-start signal, when in the manual mode, will cause the EDG controls to automatically revert to an automatic (Isochronous) mode. If this occurs during load testing, the EDG breaker will automatically trip and the Diesel will continue to run while shifting to automatic control. If an E bus undervoltage condition

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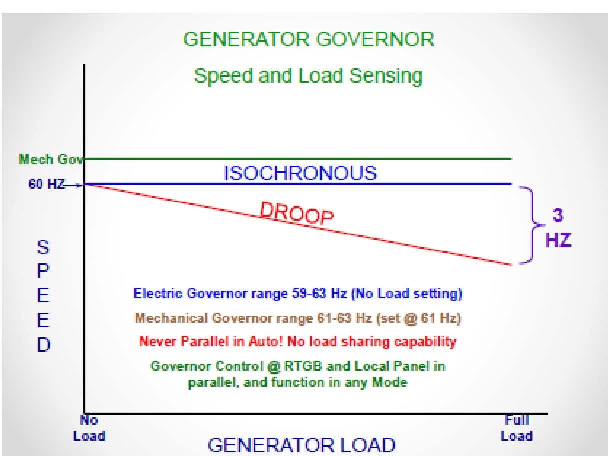
exists, the EDG will automatically tie back onto the bus in the isochronous mode once the bus was stripped. In this case the EDG may tie onto the bus at an elevated frequency, dependent on the amount of EDG load prior to the EDG mode transfer, or the setting of the mechanical governor.

The electric governor has both an ISOCHRONOUS mode and a DROOP mode of operation. ISOCHRONOUS control is constant speed as load varies, and DROOP varies speed with load (increasing load will drop the speed demand setting). The DROOP mode provides a linear change of approximately 0 to 3 Hertz from no load to rated load conditions.



34. 271000 1

While placing the AOG System in service, HCV-102, AOG System Bypass Valve, control switch on XU-80 is placed in AUTO, with the local control switch in the CLOSED position.

Subsequently, UA-45 (2-2) *Discharge H2 Conc High* alarms.

Which one of the following completes the statement below?

The HCV-102, AOG System Bypass Valve, ___(1)___, and the XCV-142, AOG Guard Bed Isolation Valve ___(2)___.

A. (1) auto opens
   (2) auto closes

B. (1) auto opens
   (2) remains open

C. (1) remains closed
   (2) auto closes

D. (1) remains closed
   (2) remains open

Answer: C

K/A:
271000 Offgas System
K1 Knowledge of the physical connections and/or cause-effect relationships between OFFGAS SYSTEM and the following: (CFR: 41.2 to 41.9 / 45.7 to 45.8)
03 Elevated release point

RO/SRO Rating: 2.7/3.0

Pedigree: Bank

Objective: LOI-CLS-LP-030, Objective 5f
Given plant conditions, determine if the following actions should have occurred or the operating condition of the component: (LOCT)
f. AOG System Isolation valve controls, interlocks, and automatic functions.

Reference: None

Cog Level: Fund.

Explanation: Hydrogen downstream of recombiner (>2%) will normally close the AOG isolation valves (including the guard bed isolation valve) and open the AOG bypass valve. Automatic opening of the bypass valve will NOT occur however unless the HCV-102 control switch is in the AUTO position.

Distractor Analysis:

Choice A: Plausible because this would be the response if both local and remote switches were in Auto.

Choice B: Plausible because the 102 would open if both switches were in Auto.

Choice C: Correct Answer, see explanation.

Choice D: Plausible because the 102 would remain closed, but the 142 would not remain open.

SRO Basis: N/A

AUGMENTED OFF GAS CHARCOAL ADSORBER SYSTEM OPERATING PROCEDURE	2OP-33
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6.1.1 AOG Charcoal Adsorber System Startup (continued)

NOTE

Partial off gas flow is processed through AOG Charcoal Adsorber System and the remaining off gas flow is bypassed through AOG-HCV-102 (AOG System Bypass Valve) to the plant stack. ☐

CAUTION

If AOG-HCV-102 (AOG System Bypass Valve) control switch is in CLOSE when the AOG Charcoal Adsorber System is in service, the bypass valve will **NOT** open on:

- High high AOG Charcoal Adsorber System flow (greater than or equal to 150 scfm). ☐
- Low glycol level (14 inches from the top) in the cooler condenser ☐
- Cooler condenser condensate high high level (95 percent). ☐
- High hydrogen (2 percent) concentration in the off gas ☐

e. **Close** AOG-HCV-102 (AOG System Bypass Valve) and **place** its control switch in AUTO. ____/____
IV

CAUTION

The local switches are aligned to match the control room switch position, **NOT** to match the valve position.. ☐

6. At local control panel H2E, **ensure** the following valve switch positions:

   - AOG-HCV-102 (AOG System Bypass Valve) in AUTO. .... ____/____ IV

   - AOG-XCV-142 (Guard Bed Isolation Valve) in OPEN................. ________

DISCHARGE H2 CONC HIGH

AUTO ACTIONS

1. Isolation to AOG System. (Closes XCV-148, 147, 142, 143, 141.)
2. Open AOG-HCV-102.

CAUSES

NOTE: The H_2/O_2 Analyzer takes approximately 145 seconds to complete one analysis cycle on one stream.

1. Hydrogen concentration greater than or equal to 2%.
2. Improper operation of off-gas train.
3. Recombiner failure (temperature ≤ 250°F).
4. Low HWC SJAE oxygen injection flow (while HWC System in service).
5. Preheater drain system failure.
6. Circuit malfunction.
7. H_2/O_2 Analyzer 2-OG-AIT-4284 Stream 2 H_2 **OR** 2-OG-AIT-4324 Stream 1 H_2 fails high.

3.1.4 AOG System Isolation Valves HCV-101, HCV-102, XCV-141, -142, -143, -147, and -148

AOG Bypass Valve, HCV-102, control switches are three position (CLOSE-AUTO-OPEN), maintained contact type switches. HCV-102 is also an air operated valve positioned by the porting of air through a solenoid operated control valve. HCV-102 is designed to shut when air is applied to the valve operator, which is accomplished when the solenoid valve is energized. When positioned to AUTO, HCV-102 will automatically open on any of the following signals:

• High H_2 conc downstream of recombiner (30 sec TD)	≥2%
• High moisture separator level	≥95%
• Low cooler condenser glycol level (from top of tank)	≤14"
• High High off-gas flow (scfm)	≥150
• (All values are procedural)	

Automatic opening of bypass valve HCV-102 will occur with master switch CS-3161 in either CENT or LOCAL.

The opening of HCV-102 automatically bypasses the AOG train and discharges the off-gas directly to the Main Stack. The logic must be reset to close the bypass valve on any of the signals with the exception of the low cooler condenser glycol level which automatically resets. Reset is accomplished by positioning the "controlling" station's control switch to the CLOSE position and then back to AUTO. If left in close, the AUTO opening features become inoperable.

35. 290002 1
A Recirc Pump startup is being performed on Unit One.

Which one of the following completes the statements below concerning the bottom head coolant temperature and the reactor pressure vessel (RPV) coolant temperature?

The maximum temperature difference is limited to ___(1)___ .

These temperature limitations IAW 1OP-02, Reactor Recirculation System, prevent ___(2)___.

A. (1) ≤ 50ºF
(2) cold water injection and resultant power spike

B. (1) ≤ 50ºF
(2) nonductile fracture of the reactor vessel pressure boundary

C. (1) ≤ 145ºF
(2) cold water injection and resultant power spike

D. (1) ≤ 145ºF
(2) nonductile fracture of the reactor vessel pressure boundary

Answer: D

K/A:
290002 Reactor Vessel Internals
K5 Knowledge of the operational implications of the following concepts as they apply to REACTOR VESSEL INTERNALS: (CFR: 41.5 / 45.3)
05 Brittle fracture

RO/SRO Rating: 3.1/3.3

Pedigree: New

Objective: LOI-CLS-LP-001, Objective 13
Given plant conditions and Technical Specifications, including the Bases, TRM, ODCM and COLR, determine whether given plant conditions meet minimum Technical Specifications requirements associated with the Reactor Vessel and Internals system.

Reference: None

Cog Level: Fund

Explanation: Temperature difference permitted between the bottom head and the Recirc loop being started up is $\leq$ 145ºF. The basis for this limit is to prevent nonductile fracture.

Distractor Analysis:

Choice A: Plausible because 50ºF is the heatup and cooldown limit between loops, and cold water injection is the basis for jogging open the Recirc discharge valve.

Choice B: Plausible because 50ºF is the heatup and cooldown limit between loops, and second part is correct.

Choice C: Plausible because 145ºF is correct, and cold water injection is the basis for jogging open the Recirc discharge valve.

Choice D: Correct Answer, see explanation.

SRO Basis: N/A

RCS P/T Limits
B 3.4.9

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.9 RCS Pressure and Temperature (P/T) Limits

BASES

BACKGROUND	All components of the RCS are designed to withstand effects of cyclic loads due to system pressure and temperature changes. These loads are introduced by startup (heatup) and shutdown (cooldown) operations, power transients, and reactor trips. This LCO limits the pressure and temperature changes during RCS heatup and cooldown, within the design assumptions and the stress limits for cyclic operation.

APPLICABLE SAFETY ANALYSES

The P/T limits are not derived from Design Basis Accident (DBA) analyses. They are prescribed during normal operation to avoid encountering pressure, temperature, and temperature rate of change conditions that might cause undetected flaws to propagate and cause nonductile failure of the RCPB, a condition that is unanalyzed. Reference 8 provides the curves and limits in this Specification. Since the P/T limits are not derived from any DBA, there are no acceptance limits related to the P/T limits. Rather, the P/T limits are acceptance limits themselves since they preclude operation in an unanalyzed condition.

RCS P/T limits satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii) (Ref. 10).

LCO

The elements of this LCO are:

a. RCS pressure and temperature are within the applicable limits specified in Figures 3.4.9-1 and 3.4.9-2, and heatup or cooldown rates are ≤ 100°F in any 1 hour period, during RCS heatup and cooldown;

b. RCS pressure and temperature are within the applicable limits in Figures 3.4.9-3, 3.4.9-4, or 3.4.9-5 and heatup or cooldown rates are ≤ 30°F in any 1 hour period, during RCS inservice leak and hydrostatic testing;

c. The temperature difference between the reactor vessel bottom head coolant and the RPV coolant is ≤ 145°F during recirculation pump startup;

d. The temperature difference between the reactor coolant in the respective recirculation loop and in the reactor vessel is ≤ 50°F during recirculation pump startup;

e. RCS pressure and temperature are within the criticality limits specified in Figure 3.4.9-2, prior to achieving criticality; and

f. The reactor vessel flange and the head flange temperatures are ≥ 70°F when tensioning the reactor vessel head bolting studs.

(continued)

36. 290003 1

A Station Blackout occurs on Unit One. DG4 is the only Diesel available. No cross-tie actions are able to be performed. Control Building HVAC is unavailable.

Which one of the following Unit One time-critical actions is required to be performed <u>within 30 minutes</u>, IAW 1-EOP-01-SBO, Station Blackout Procedure?

A. Load strip the batteries.

B. Cooldown to 150-300 psig.

C. Open Reactor Building doors.

D. Open the Control Room panel doors.

Answer: D

K/A:
290003 Control Room HVAC
K5 Knowledge of the effect that a loss or malfunction of the CONTROL ROOM HVAC will have on following: (CFR: 41.7 /45.6)
05 Computer/instrumentation: Plant-Specific

RO/SRO Rating: 3.3/3.6

Pedigree: New

Objective: LOI-CLS-LP-037, Objective 8
Given plant conditions, determine the effect that a loss or malfunction of the CB HVAC and EAF System will have on the following: a. Control Room Panels

Reference: None

Cog Level: Fund.

Explanation: Without Control Room ventilation, Control Room panel doors must be opened within 30 minutes. Cooldown and load stripping batteries are required within 60 minutes. Reactor Building doors must be opened within 6 hours.

Distractor Analysis:

Choice A: Plausible because load stripping batteries is required within 60 minute if power cannot be restored to the battery chargers.

Choice B: Plausible because cooldown to 150-300 psig is required within 60 minutes.

Choice C: Plausible because Reactor Building doors must be open within 6 hours if E1 or E2 cannot be energized within 60 minutes.

Choice D: Correct Answer, see explanation.

SRO Basis: N/A

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5.2.2 Step 2.3.2

Control Building equipment area temperatures could increase until component reliability is threatened. If one unit is not blacked out, AC power may be available to start a Control Building supply fan and associated air conditioner. This is not assured since both Control Building air compressors, required to align dampers for ventilation operation, are supplied by Division I power sources. In addition, Unit 1, Division II does not power a Control Building supply fan and associated air conditioner. In an ELAP neither unit will have AC power to operate ventilation. If cooling cannot be established, opening both units panel doors is required within 30 minutes from the SBO start time. Panels of concern include both back panels and RTGB panels.

STATION BLACKOUT PROCEDURE BASIS DOCUMENT	0OI-37.14
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If any division's battery chargers cannot be restored, batteries are load stripped to conserve DC power. Load stripping includes UPS inverters, lighting inverters and major emergency DC oil pumps (Reactor Feed Pump Turbine, Main Turbine Emergency Seal Oil and Bearing Oil Pumps).

Control Room ventilation for at least one unit is restored so that the safety system instrumentation and controls function properly during the SBO. The SBOCA Report requires Control Room panel doors must be opened within 30 minutes of the start of the SBO if Control Building ventilation is not reestablished. In addition, contingency plans for alternate Control Room ventilation, using portable generators and fans, have been established.

Time Critical

Battery load stripping required within **60 minutes**.

Time ______________________

IF <u>either</u> division battery chargers <u>CANNOT</u> be restored,

THEN **load strip** <u>affected</u> battery per EOP-01-SBO-10.

SBO-21

WHEN

- Security

SBO-12

Time Critical

Control Room panel doors required open within **30 minutes** if A/C <u>NOT</u> available.

Time ______________________

Perform EOP-01-SBO-02 to:

- **Start** Control Room A/C <u>OR</u> **open** panel doors
- **Align** RCIC and HPCI to torus
- **Align** for recovery:
  - ◊ Switchyard
  - ◊ EDGs
  - ◊ RBCCW
  - ◊ S/D cooling

SBO-13

Time Sensitive

Torus suction required within **60 minutes.**

Time ______________________

Time Critical

Cooldown to **150-300 psig** required to start within **60 minutes**

Time ______________________

Control RPV pressure per EOP-01-RVCP.

SBO-10

Open reactor building doors per EOP-01-SBO-04.

SBO-25

Time Sensitive

RB roof hatch required open within **2 hours**.

Time ______________________

Stage alternate fuel pool makeup/spray equipment per EOP-01-SEP-12.

SBO-26

37. 295001 1

Unit One is operating at 94% power with OPRMs Inoperable, when Recirculation Pump 1A trips due to a fault.

The following conditions exist:

Total Core Flow (P603)	31.0 Mlbm/hr.
Total Core Flow (U1CPWTCF)	32.2 Mlbm/hr.
APRMs:	44%

Which one of the following identifies the operating location on the Power to Flow map, and the required operator action?

(Reference provided)

A. Region B, insert manual reactor scam.

B. Region B, raise core flow or insert control rods to exit.

C. Scram Avoidance Region, insert manual reactor scam.

D. Scram Avoidance Region, raise core flow or insert control rods to exit.

Answer: B

K/A:
295001 Partial or Complete Loss of Forced Core Flow Circulation
AK1 Knowledge of the operational implications of the following concepts as they apply to PARTIAL OR COMPLETE LOSS OF FORCED CORE FLOW CIRCULATION: (CFR: 41.8 to 41.10)
04 Limiting cycle oscillation: Plant-Specific

RO/SRO Rating: 2.5/3.3

Pedigree: New

Objective: LOI-CLS-LP-302-C, Objective 7
Given plant conditions and AOP-3.0, determine the required supplementary actions.

Reference: Unit 1 Power to Flow Maps (OPRM Operable , Two Loop Operation and Single Loop Operations, OPRM Inoperable Two Loop and Single Loop Operation, .

Cog Level: High

Explanation: See plot of location on Power to Flow map in Notes Section.

Distractor Analysis:

Choice A: Plausible because Region B is correct. Reactor Scram would be correct for Region A.

Choice B: Correct Answer, see explanation.

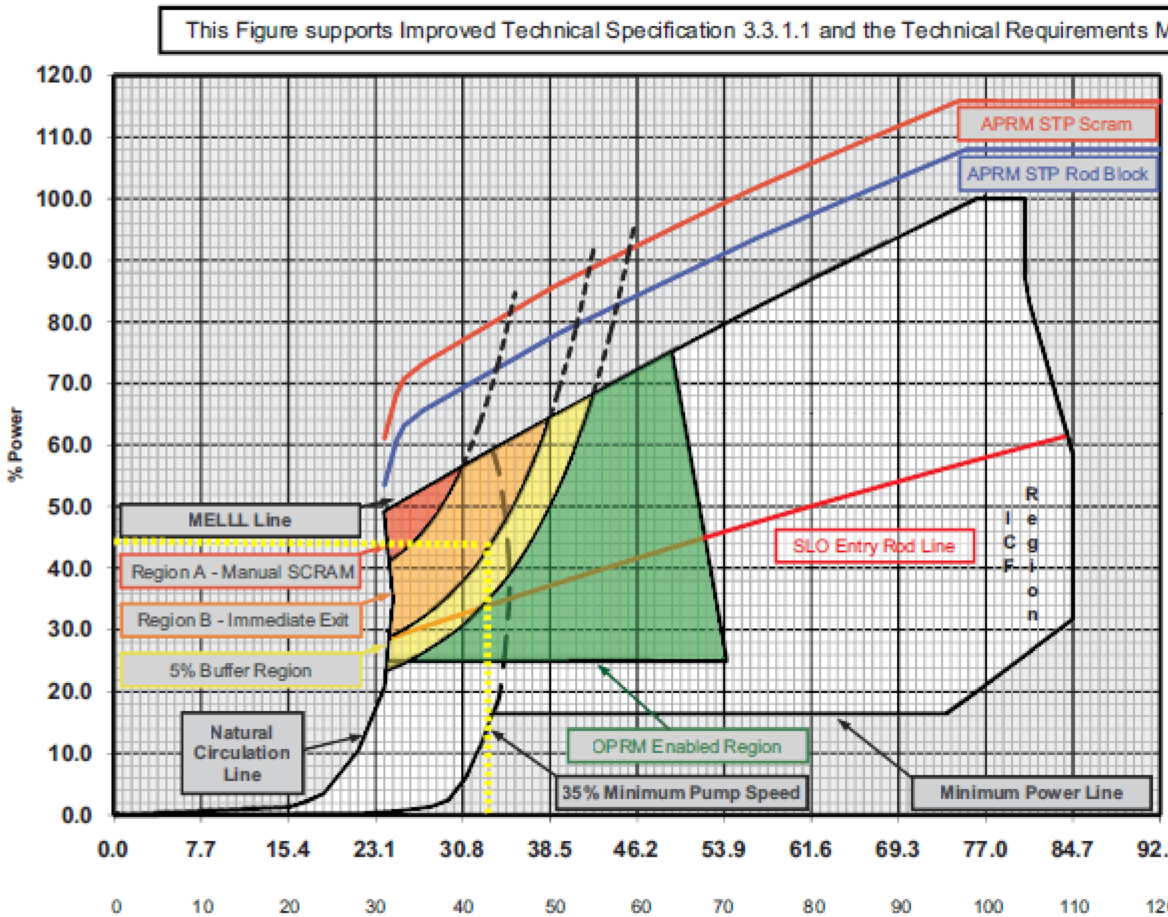
Choice C: Plausible because if OPRM Operable Single Loop Operation flow map is chosen, this would be correct, and scram would be appropriate if oscillations are observed.

Choice D: Plausible because if OPRM Operable Single Loop Operation flow map is chosen, this would be correct, and second part would be the correct action.

SRO Basis: N/A

Figure 2
Stability Option III Power/Flow Map

OPRM Inoperable, Two Loop Operation, 2923 MWt



Power %	Minimum (MELLL) Core Flow Mlbs/hr
100	76.19
99	75.04
98	73.89
97	72.75
96	71.61
95	70.49
94	69.36
93	68.25
92	67.13
91	66.03
90	64.93
89	63.83
88	62.74
87	61.66
86	60.58
85	59.50
84	58.43
83	57.37
82	56.31
81	55.25
80	54.20
79	53.16
78	52.12
77	51.08
76	50.05
75	49.02
74	48.00
73	46.98
72	45.96
71	44.95
70	43.94
69	42.94
68	41.94
67	40.95
66	39.96
65	38.97
64	37.99
63	37.01
62	36.04
61	35.06
60	34.10
59	33.13
58	32.17

4.2 Supplementary Actions (continued)

b. **IF** the current operating point is in Region B,
THEN exit the region using one of the following methods:

NOTE

- Total core flow should **NOT** exceed 45 x 10^6 lbs/hr (58%) with only one reactor recirculation pump in operation. .... ☐
- When raising core flow with two reactor recirculation pumps operating, jet pump loop flow mismatch should be maintained within the allowable limit. .... ☐

CAUTION

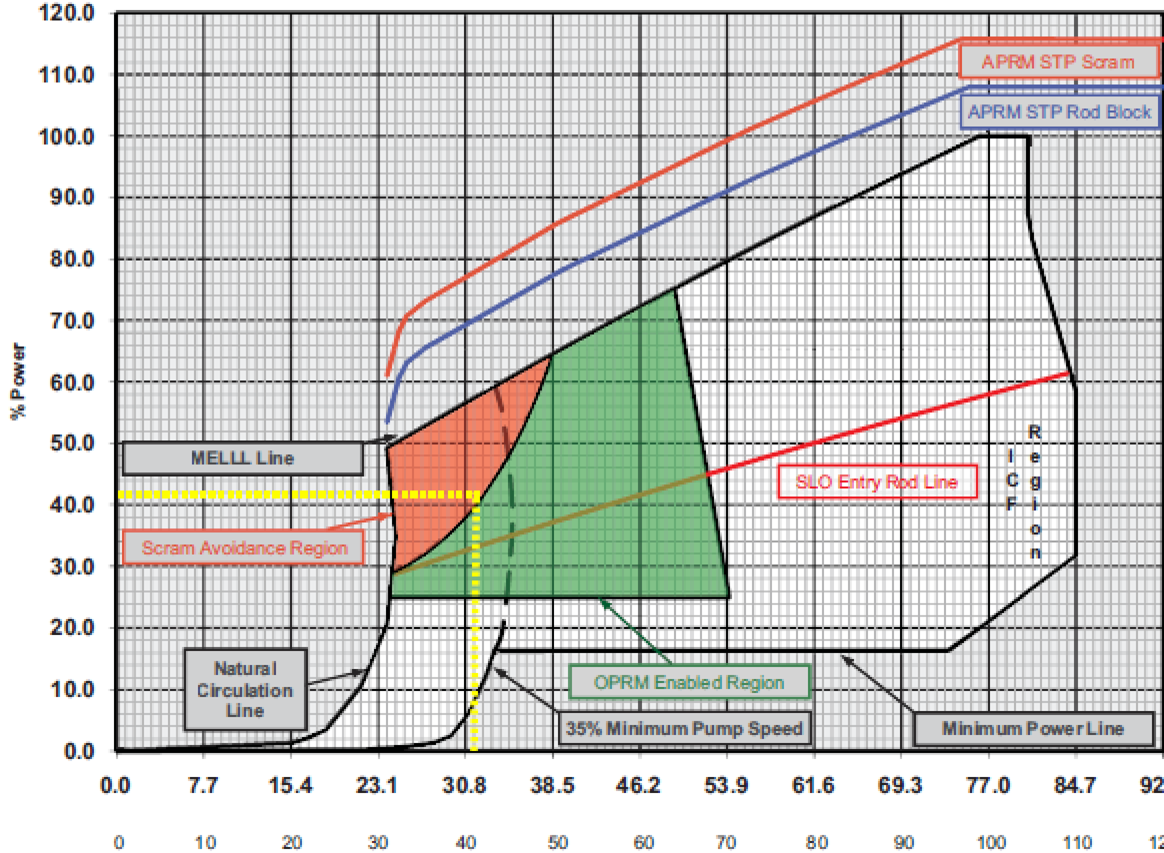
If operating in Region B, a reactor recirculation pump shall **NOT** be started to exit the region. .... ☐

- **Raise** core flow. .... ☐
- **Insert** control rods in accordance with 0ENP-24.5, Form 2, Immediate Reactor Power Reduction Instructions. .... ☐

Figure 1
Stability Option III Power/Flow Map

OPRM Operable, Two Loop Operation, 2923 MWt

This Figure supports Improved Technical Specification 3.3.1.1 and the Technical Requirements Manual Specification 3.3



Power %	Minimum (MELLL) Core Flow Mlbs/hr	M
100	76.19	
99	75.04	
98	73.89	
97	72.75	
96	71.61	
95	70.49	
94	69.36	
93	68.25	
92	67.13	
91	66.03	
90	64.93	
89	63.83	
88	62.74	
87	61.66	
86	60.58	
85	59.50	
84	58.43	
83	57.37	
82	56.31	
81	55.25	
80	54.20	
79	53.16	
78	52.12	
77	51.08	
76	50.05	
75	49.02	
74	48.00	
73	46.98	
72	45.96	
71	44.95	
70	43.94	
69	42.94	
68	41.94	
67	40.95	
66	39.96	
65	38.97	
64	37.99	
63	37.01	
62	36.04	
61	35.06	
60	34.10	
59	33.13	
58	32.17	

38. 295003 1

Unit Two is operating at 30% power when the following sequence of events occurs:

Time	Generator Frequency
1208	59.8 Hz
1212	59.2 Hz
1216	58.8 Hz
1218	58.3 Hz

Which one of the following completes the statements below?

UA-06 (1-2) *Gen Under Freq Relay* first alarms at ___(1)___.

Given the conditions above, IAW 0AOP-22.0, Grid Instability, the required operator action(s) is(are) to ___(2)___.

A. (1) 1208
   (2) trip the main turbine ONLY

B. (1) 1208
   (2) manually scram the reactor and then trip the main turbine

C. (1) 1216
   (2) trip the main turbine ONLY

D. (1) 1216
   (2) manually scram the reactor and then trip the main turbine

Answer: B

K/A:
295003 Partial or Complete Loss of A.C. Power
G2.4.50 Ability to verify system alarm setpoints and operate controls identified in the alarm response manual. (CFR: 41.10 / 43.5 / 45.3)

RO/SRO Rating: 4.2/4.0

Pedigree: New

Objective: LOI-CLS-LP-302-G, Objective 3b
Given plant conditions and any of the following AOPs, determine the required Supplemental Actions: AOP-22.0, Grid Instability.

Reference: None

Cog Level: High

Explanation: APP UA-06, 1-2 , setpoint 59.8 Hz, directs operator to enter 0AOP-22.0. Cautions in AOP-22.0 identify what actions are required at various frequencies. At max allowable time at a given frequency, operator is directed to scram reactor if power is greater than or equal to 26%, then trip the turbine. This matches the K/A because there are no actions identified in the APP except to enter the AOP.

Distractor Analysis:

Choice A: Plausible because the first part is correct, and if power were less than 26%, then the turbine would be tripped.

Choice B: Correct Answer, see explanation.

Choice C: Plausible because the operator must know the alarm setpoint, and if power were less than 26%, then the turbine would be tripped.

Choice D: Plausible because the operator must know the alarm setpoint, and the second part is correct.

SRO Basis: N/A

Unit 2
APP UA-06 1-2
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GEN UNDER FREQ RELAY

AUTO ACTIONS

1. Generator MWs increase to the limits of the pressure set.

CAUSE

1. Insufficient generation for system load.
2. Circuit malfunction.

OBSERVATIONS

1. Frequency decreasing (GEN-FM-736).
2. Increase in generator MW (GEN-MW-727).
3. System voltage decreasing (GEN-VM-732).

ACTIONS

NOTE:	A sudden increase in system frequency is possible if load shedding or other actions should result in turning a generation shortage into a generation excess.

1. Enter 0AOP-22.0, Grid Instability.
2. Increase turbine output to the maximum consistent with plant conditions per 0GP-04, Increasing Turbine Load to Rated Power.
3. If the system frequency is less than 58.1 hertz, trip the turbine immediately.
4. If a circuit malfunction is suspected, ensure a WO is prepared.

DEVICE/SETPOINTS

Generator Frequency Relay 81	59.8 Hz
230 KV PCB-29A Position Relay 29A/X	Closed
230 KV PCB-29B Position Relay 29B/X	Closed

4.2 Supplementary Actions

NOTE

A sudden rise in system frequency may be observed due to additional generation or load shedding. Automatic load shedding (10% of system load) occurs at each of the following frequencies: 59.3, 59.0, and 58.5 Hz. ☐

CAUTION

The maximum allowable time at a given frequency is as follows: ☐

- Below 58.1 Hz, operation is prohibited
- Between 58.1-58.5 Hz, operation for 1 minute is allowed
- Between 58.6-59.3 Hz, operation for 5 minutes is allowed
- Between 59.4-60.6 Hz, operation is allowed indefinitely
- Between 60.7-61.4 Hz, operation for 5 minutes is allowed
- Between 61.5-61.9 Hz, operation for 1 minute is allowed
- Above 61.9 Hz, operation is prohibited

CAUTION

- Off-frequency operation can stimulate resonance vibration in low pressure blades. ☐
- A total loss of off-site power (LOOP) should be anticipated if the turbine is tripped. ☐
- With grid voltage or frequency unstable or grid vulnerability identified, diesel generators should **NOT** be paralleled with any E bus connected to the grid since severe load swings may occur and possibly overload the diesel generators. ☐

1. **IF** the maximum allowable time at a given frequency is exceeded, **THEN perform** the following:

   a. **IF** reactor power is greater than or equal to 26%, **THEN insert** a manual scram. ☐

   b. **Trip** the main turbine. ☐

   c. **IF** the unit was scrammed, **THEN enter** 1EOP-01-RSP(2EOP-01-RSP), Reactor Scram Procedure. ☐

39. 295004 1

UA-23 (3-8) *250 VDC Battery B Ground* alarm is received and sealed in on Unit Two. An AO reports the following readings:

2B-1	135 VDC
2B-2	135 VDC
N Bus	1.1 ma
PN Bus	0.4 ma
P Bus	2.7 ma

Which one of the following completes the statements below?

The ground is located on the ___(1)___ Bus.

Isolation of this ground is performed in order to prevent inadvertent __(2)___.

(Reference provided)

A. (1) N
(2) pick up of a de-energized relay

B. (1) N
(2) hold in of an energized relay

C. (1) P
(2) pick up of a de-energized relay

D. (1) P
(2) hold in of an energized relay

Answer: B

K/A:
295004 Partial or Complete Loss of D.C. Power
AK3 Knowledge of the reasons for the following responses as they apply to PARTIAL OR COMPLETE LOSS OF D.C. POWER : (CFR: 41.5 / 45.6)
02 Ground isolation/fault determination

RO/SRO Rating: 2.9/3.3

Pedigree: Bank

Objective: LOI-CLS-LP-051, Objective 9
Given Ground Detector readings and AI-115, determine which bus has the ground and the required actions per AI-115.

Reference: OP-51, Attachment 2

Cog Level: High

Explanation: OP-51, Attachment 2, result in calculation of 21 Kohms. With P>N, ground is in N bus per OP-51, Table 1. 0AI-115, Section 3.0 indicates relays that a ground below 25 Kohms could hold-in a normally energized relay.

Distractor Analysis:

Choice A: Plausible because N bus is correct. Based on calculation, this could be correct for calculation below 15 Kohms.

Choice B: Correct Answer, see explanation.

Choice C: Plausible because location can either be P or N bus. Based on calculation, this could be correct for calculation below 15 Kohms.

Choice D: Plausible because Plausible because location can either be P or N bus. Second part is correct.

SRO Basis: N/A

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3.0 BASIS FOR ACTION LEVELS

Electrical Evaluation BNP-E-6.116 was issued to provide a basis for the setpoint of the battery ground detectors. The value established by this evaluation is 25 kilohms (kΩ). The ground detectors operate with a ± 15% band, which corresponds to 21.3 - 28.8 KΩ.

Electrical devices at BNP were researched and the most sensitive device of concern was determined to be GE HFA relays. These relays have a nominal dropout current of 3.75 mA. This value was utilized to derive an appropriate setpoint.

At ground levels below 25 kΩ, situations (involving two very strategically located grounds) could develop that might hold-in a normally energized relay.

At ground levels below 15 kΩ, sufficient currents develop which could result in holding-in a normally energized relay, dropping a normally energized relay or picking up a de-energized relay. In general, with ground levels below 15 kΩ, the relays may not perform in a predictable manner and therefore this is the level that is considered most urgent to correct.

C
Continuous
Use

Data Sheet for Battery Ground Detection

Date:________________

Time:________________

NOTE:	The "- 50" factor in the equation below accounts for the presence of a 50 KΩ resistor in series with the milliamp meter.
NOTE:	An example calculation is provided on the following page.

Battery 2A

Current **P** bus: _______ mA

Current **PN** bus: ______ mA

Current **N** bus: _______ mA

Voltage 2A-1: _______ VDC

Voltage 2A-2: _______ VDC

2A Resistance = $\dfrac{\text{VDC} + \text{VDC}}{\mathbf{P}\ (\text{mA}) + \mathbf{N}\ (\text{mA})}$ **- 50** = ___________ KΩ

Battery 2B

Current **P** bus: 2.7 mA

Current **PN** bus: ______ mA

Current **N** bus: 1.1 mA

Voltage 2B-1: 135 VDC

Voltage 2B-2: 135 VDC

2B Resistance = $\dfrac{\text{VDC} + \text{VDC}}{\mathbf{P}\ (\text{mA}) + \mathbf{N}\ (\text{mA})}$ **- 50** = 21. KΩ

TABLE 1

General Guidelines for Determining Which Bus Is Grounded

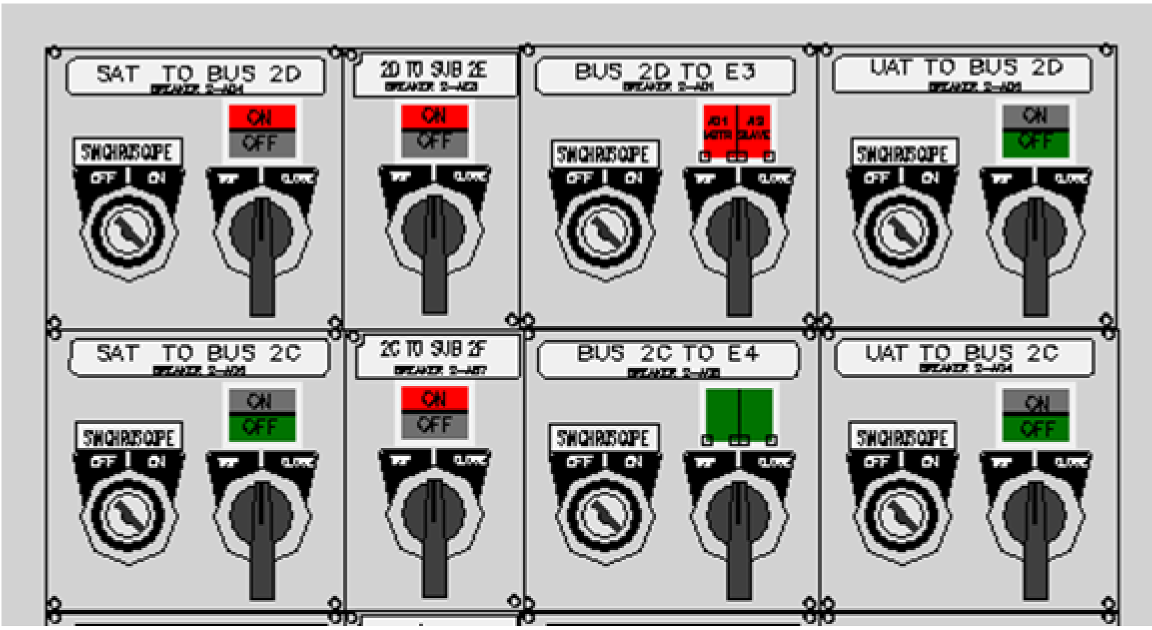
I
Information
Use

There will always be some resistance to ground which means there will always be a mA reading to ground on each bus. Also, one bus will typically have a larger mA reading to ground than the other, but unless the overall resistance of the system falls below the setpoint, this will generally be acceptable. If a ground below 25 KΩ is found to exist, these guidelines should be referred to for assistance.

mA Reading	Grounded Bus
$P > N$	N
$P < N$	P
* $P \approx N$	PN

40. 295005 1

Unit Two is operating at rated power when a fault trips the Main Generator Primary Lockout relay. The following breaker lineup is observed:



Which one of the following completes the statements below?

Bus E3 is energized from ___(1)___

Bus E4 is energized from ___(2)___.

A. (1) DG3
   (2) DG4

B. (1) DG3
   (2) Off-Site Power

C. (1) Off-Site Power
   (2) DG4

D. (1) Off-Site Power
   (2) Off-Site Power

Answer: C

K/A:
295005 Main Turbine Generator Trip
AA1 Ability to operate and/or monitor the following as they apply to MAIN TURBINE GENERATOR TRIP: (CFR: 41.7 / 45.6)
07 A.C. electrical distribution

RO/SRO Rating: 3.3/3.3

Pedigree: New

Objective: LOI-CLS-LP-027, Objective 5
State the effect that actuation of a main generator lockout relay will have on the Main Generator and station loads.
LOI-CLS-LP-039, Objective 12
Given plant conditions, determine if permissives are satisfied for the EDG output breaker to close (either automatically or manually).

Reference: None

Cog Level: High

Explanation: Based on the conditions, the RO will have to determine the status in order to report to the CRS which meets the monitoring AC electrical on a generator trip. Generator primary lockout is a loss of off-site power signal to DG auto start logic. All four DGs will receive an auto start signal. Bus 2C fails to transfer from UAT to SAT on the trip. This results in loss of BOP Bus 2C which feeds E4. DG4 will tie to bus E4.

Distractor Analysis:

Choice A: Plausible because this would be the configuration if both BOP busses failed to transfer.

Choice B: Plausible because this would be the configuration if the BOP bus that failed to transfer to the SAT was Bus 2D rather than 2C.

Choice C: Correct Answer, see explanation.

Choice D: Plausible because this would be the configuration if both BOP busses transferred.

SRO Basis: N/A

3.2.4 Automatic Start

The DG auto start circuitry actuates on a loss of power at designated points in the plant electrical system and also actuates on a loss-of-coolant accident. The following is a list of the parameters or conditions which will initiate an auto start of the EDGs. Each of the automatic starting logic schemes is discussed below.

2. Electrical System Faults (Figure 39-12)

A Loss Of Off-Site Power (LOOP) DG auto start signal will be generated for **all four EDGs** if any one of the following conditions exists on **either unit**:

- Generator Primary Lockout for either unit (Division I logic only) which is caused by:
  - generator overall differential
  - generator reverse power
  - distance relay
  - generator output breaker failure
  - UAT differential phase overcurrent
  - Generator loss of field
- SAT Lockout (Division I logic only) for either unit.
- Generator Differential Lockout (Division II logic only) either unit.
- Transformer Bus Differential Lockout (Division I logic only) for either unit.
- SAT secondary side undervoltage.

41. 295006 1

Following a loss of the Uninterruptible Power Supply on Unit One, a reactor scram occurs.

Which one of the following completes the statements below?

IAW OI-37.5, ATWS Basis Document, Reactor power ___(1)___ be determined to be below 2%.

The Reactor ___(2)___ S/D Without Boron under all conditions.

A. (1) can
   (2) is

B. (1) can
   (2) is NOT

C. (1) can NOT
   (2) is

D. (1) can NOT
   (2) is NOT

Answer: B

K/A:
295006 SCRAM
AA2 Ability to determine and/or interpret the following as they apply to SCRAM:
(CFR: 41.10 / 43.5 / 45.13)
02 Control rod position

RO/SRO Rating: 4.3/4.4

Pedigree: New

Objective: LOI-CLS-LP-300-E, Objective 11e
Given plant conditions and the ATWS Control Procedure, determine the following:
e. If the reactor is shutdown.

Reference: None

Cog Level: High

Explanation: The APRM downscale setpoint is determined to be below 2% power per OI-37.5, ATWS Basis document. With a loss of UPS, there is no way to readily determine control rod positions and therefore no way to declare the reactor shutdown under all conditions without boron. RPIS inputs to the process computer are invalid.

Distractor Analysis:

Choice A: Plausible because part 1 is correct. With a loss of UPS, RPIS information is lost. There is no way to readily determine control rod position.

Choice B: Correct Answer, see explanation.

Choice C: Plausible because there is only one condition listed that will identify power level. With a loss of UPS, RPIS information is lost. There is no way to determine control rod position.

Choice D: Plausible because there is only one condition listed that will identify power level. Part 2 is correct.

SRO Basis: N/A

If reactor power is below the APRM downscale trip setpoint, tripping the recirculation pumps results in little, if any, reduction in reactor power since power is already near the decay heat level. In this case, forced recirculation flow is permitted to continue for the purpose of maximizing boron mixing should boron injection later be required.

If reactor power is at or below the APRM downscale trip setpoint (2%), it is highly unlikely that the core bulk boiling boundary would be below that which provides suitable stability margin for operation at high powers and

1.0 PURPOSE

1. This procedure lists symptoms and automatic actions and provides operator actions for loss of uninterruptible power.

2.0 SYMPTOMS

1. Actual complete or partial loss of UPS which may be indicated by one or more of the following indications:

   a. Possible indications:

   - Loss of instrument power to Control Room Panels XU-2, XU-3, P603, and various nuclear instrument recorders
   - Loss of RPIS, full core display and rod position display
   - Loss of RWM Operator Display and RMCS resulting in an inability to move control rods

Reactor Manual Control - Control rods cannot be moved by normal means (scram function is unaffected). Power is lost to the rod position display panel. Full core display is lost. Since the rod position information system is lost the NIs must be closely monitored to ensure the reactor is shutdown and remains shutdown during any subsequent cooldown.

Rod Worth Minimizer - Power is lost to the RWM buffer and operator display. This results in the inability to move rods and cannot be bypassed.

Turbine Supervisory Instrumentation - Will not provide vibration protective trips for the main turbine or indication of expansion and

Table Q-1
Shutdown Without Boron

- All rods in
- Only one rod NOT fully inserted
- NO more than 10 rods withdrawn to position 02 AND NO rod beyond position 02
- As determined by Reactor Engineering

42. 295007 1

Following a MSIV closure on Unit One, RVCP is being executed. It is determined that SRVs are cycling.

Which one of the following is directed by RVCP to terminate SRV cycling?

Open SRV/ADS valves until reactor pressure drops to ___(1)___.

The SRV opening sequence ___(2)___ required while executing this step.

A. (1) 950 psig
   (2) is

B. (1) 950 psig
   (2) is NOT

C. (1) 1050 psig
   (2) is

D. (1) 1050 psig
   (2) is NOT

Answer: B

K/A:
295007 High Reactor Pressure
G2.4.11 Knowledge of abnormal condition procedures. (CFR: 41.10 / 43.5 / 45.13)

RO/SRO Rating: 4.0/4.2

Pedigree: Bank

Objective: LOI-CLS-LP-300-D, Objective 7
Given plant conditions, the Reactor Vessel Control Procedure, and which steps have been completed, determine the required operator actions.

Reference: None

Cog Level: Fund.

Explanation: IAW discussion with NRC, this K/A can be written to Abnormal/Emergency procedures, or Annunciator Procedure actions. See discussion in Notes Section for explanation of correct answer.

Distractor Analysis:

Choice A: Plausible because part 1 is correct, part 2 is plausible because the opening sequence is typically used.

Choice B: Correct Answer, see explanation.

Choice C: Plausible because the succeeding step of RVCP lowers pressure to below 1050 after defeating Group 1 islolation, part 2 is plausible because the opening sequence is typically used.

Choice D: Plausible because the succeeding step of RVCP lowers pressure to below 1050 after defeating Group 1 isolation, part 2 is correct.

SRO Basis: N/A

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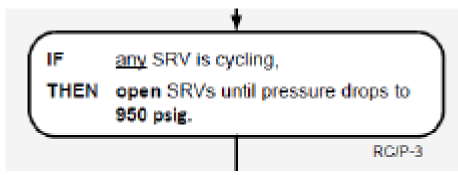
5.16 **Step RC/P-3**



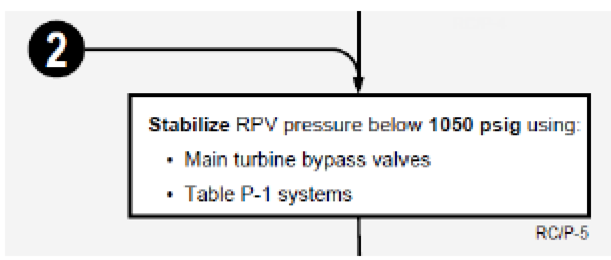
SRV cycling is defined as multiple, closely sequenced valve actuations with valve opening being initiated in response to RPV pressure increasing to/above the lift setpoint, and valve closure being governed by RPV pressure decreasing to/below the reset point. Potential severe consequences associated with SRV cycling require prompt manual action to reduce RPV pressure below the SRV lift setpoint. Actions to prevent SRV cycling will minimize:

5.16 Step RC/P-3 (continued)

As SRVs are opened, RPV pressure will drop and approach some equilibrium value dependent upon the thermal power being generated. Since the SRVs are of relatively large capacity, it is unlikely that final RPV pressure will exactly correspond to the pressure at which all turbine bypass valves are fully open; it will either be higher or lower. Opening SRVs until RPV pressure drops to 950 psig will most likely require a pressure reduction below the target. The requisite number should still be opened, even if it results in temporary closure of some bypass valves.

A RPV pressure control band or SRV opening sequence is not specified here, since the purpose of the instruction is simply to reduce RPV pressure quickly and effect direct, positive control of the SRVs. Direction to close the SRVs after they are manually opened is not included in this step. The control and stabilization of RPV pressure after SRV cycling is terminated is addressed in subsequent steps.

5.18 <u>Step RC/P-5</u>



This step stabilizes RPV pressure below 1050 psig to avoid SRV actuation and to permit the scram logic to be reset (if no other scram signal exists). No minimum value is specified since the RPV pressure at which the EOPs are entered cannot be predefined and the instruction must provide appropriate guidance for all events. The actual pressure band should be selected close to the initial value upon entry but below 1050 psig to permit use of available injection systems. An initial adjustment to establish an appropriate target pressure is permitted, provided the target can be reached expeditiously and the Technical Specification cooldown rate LCO is not exceeded.

43. 295008 1

Which one of the following completes the statement below IAW 2APP-A07 (2-2), *Reactor Water Level High/Low*?

A ___(1)___ Reactor pressure transient will cause high Reactor water level, which can result in increased ___(2)___.

A. (1) low
   (2) jet pump vibration

B. (1) low
   (2) erosion wear of turbine blades

C. (1) high
   (2) jet pump vibration

D. (1) high
   (2) erosion wear of turbine blades

Answer: B

K/A:
295008 High Reactor Water Level
AK1 Knowledge of the operational implications of the following concepts as they apply to HIGH REACTOR WATER LEVEL: (CFR: 41.8 to 41.10)
02 Component erosion/damage

RO/SRO Rating: 2.8/2.8

Pedigree: Bank

Objective: LOI-CLS-LP-01, Objective 8
With regard to moisture carryover:
a. Define the term
b. Describe how it is affected by reactor water level
c. Describe the adverse effects.

Reference: None

Cog Level: Fund.

Explanation: Moisture carryover is defined as that moisture entrained in the steam exiting the Reactor Pressure Vessel. The amount of carryover is affected by the reactor water level. If the water level is too high, the water draining out of the separators tends to back up resulting in increased moisture out the top of the separators. Too much moisture will overload the steam dryers with a resultant decrease in steam quality exiting the reactor vessel. This decreased steam quality will cause increased erosion of turbine blades.

Distractor Analysis:

Choice A: Plausible because part 1 is correct. Jet pump vibration is a low core flow concern, but may be considered for high or low level.

Choice B: Correct Answer, see explanation.

Choice C: Plausible because high pressure can cause level reduction and resultant jet pump cavitation. Jet pump vibration is a low core flow concern, but may be considered for high or low level. Second part would be true for low water level.

Choice D: Plausible because high pressure can cause level reduction and resultant jet pump cavitation. Jet pump vibration is a low core flow concern, but may be considered for high or low level, and part 2 is correct.

SRO Basis: N/A

a. **Moisture Carryover**

Moisture carryover is defined as that moisture entrained in the steam exiting the Reactor Pressure Vessel. The amount of carryover is affected by the reactor water level. If the water level is too high, the water draining out of the separators tends to back up resulting in increased moisture out the top of the separators. Too much moisture will overload the steam dryers with a resultant decrease in steam quality exiting the reactor vessel. The amount of carryover is minimized in order to: 1) increase turbine efficiency, 2) decrease turbine wear, and 3) minimize the amount of radioactivity carried over to the balance of plant (BOP).

b. Steam Carryunder

Steam carryunder is defined as that steam entrained with the liquid draining to the downcomer from the steam separators and dryers. Carryunder is always present to some extent, but can become excessive due to a low reactor water level condition when steam is pulled down into the bulk water region below the dryer skirt and mixed with feedwater. The problem with an excessive steam carryunder condition is that this entrained steam results in a lower density fluid reaching the reactor recirculation pumps and jet pumps and decreasing the available net positive suction head (NPSH). The decrease in NPSH increases the chance of recirculation pump and jet pump cavitation. Excessive steam carryunder also decreases the margin to Core Thermal Limits (MCPR).

2.2 High level could be due to one or more of the following:

2.2.1 Starting of a Condensate, Condensate Booster, Heater Drain or Reactor Feedwater Pump.

2.2.2 Pressure transient (reduction in pressure).

2.2.3 Feed flow has exceeded steam flow during manual level control.

2.2.4 Control signal malfunction.

CAUTION

Reducing core flow less than 30.8 x 10^6 lbs/hr (40% rated core flow) may cause idle loop temperature to lower and is to be minimized due to possible jet pump vibration and the potential for consequent stress fatigue of the riser brace welds to the vessel. The operating pump flow reduction is to be as close to pump start as possible. If delays in restart of the second pump are encountered, the operating loop flow is raised to greater than 30.8 x 10^6 lb/hr to ensure temperature in the idle loop is maintained. .... ☐

44. 295010 1

Following a line break in the drywell, Unit One conditions are:

Drywell pressure	6 psig
Drywell temperature	250°F
Torus pressure	7 psig
Torus level	-27 inches

Which one of the following completes the statements below?

The Suppression Chamber to Drywell Vacuum relief valves would be expected to be ___(1)___.

The purpose of the Suppression Chamber to Drywell Vacuum relief valves is to prevent ___(2)___.

A. (1) open
   (2) chugging at the downcomer openings of the drywell vents

B. (1) open
   (2) suppression pool water from being drawn into the drywell through the downcomers and vent pipes after a LOCA

C. (1) closed
   (2) chugging at the downcomer openings of the drywell vents

D. (1) closed
   (2) suppression pool water from being drawn into the drywell through the downcomers and vent pipes after a LOCA

Answer: B

K/A:
295010 High Drywell Pressure
AK2 Knowledge of the interrelations between HIGH DRYWELL PRESSURE and the following: (CFR: 41.7 / 45.8)
02 Drywell/Suppression Chamber differential pressure: Mark-I&II

RO/SRO Rating: 3.3/3.5

Pedigree: New

Objective: LOI-CLS-LP-004-A, Objective 8
Describe the operation of the Suppression Chamber to Drywell Vacuum Breakers.

Reference: None

Cog Level: Fund

Explanation: The Suppression Chamber to Drywell vacuum breakers are designed to open within 1 second after Torus pressure is 0.5 psid greater than Drywell pressure. This prevents Torus water from being drawn into the Drywell through the downcomers and vent pipes after a LOCA.

Distractor Analysis:

Choice A: Plausible because part 1 is correct. Part 2 is the basis for spraying the Torus before Torus pressure reaches 11.5 psig in PCCP.

Choice B: Correct Answer, see explanation.

Choice C: Plausible because the trainee needs to know how the vacuum breakers work and the direction of the operation. Part 2 is the basis for spraying the Torus before Torus pressure reaches 11.5 psig in PCCP.

Choice D: Plausible because the trainee needs to know how the vacuum breakers work and the direction of the operation. Part 2 is correct.

SRO Basis: N/A

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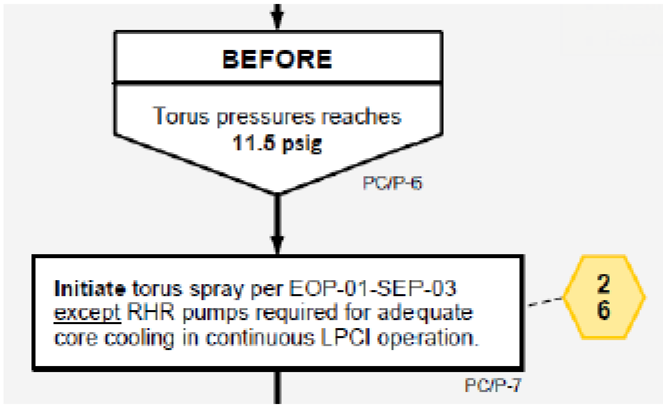
The Suppression Chamber to Drywell vacuum relief valves, (Figure 4-4B), prevent suppression pool water from being drawn into the drywell through the downcomers and vent pipes after a LOCA. This could occur when the Containment Spray System is actuated and the drywell pressure drops below that of the suppression chamber.

Ten vacuum relief valves are located on the ring vent header, with a direct flow channel into the drywell. These relief valves are intended to bleed non-condensable gases from the suppression chamber into the drywell to equalize pressure between the torus and drywell. The valves are designed to completely open within 1 second after 0.5 psid is applied across the valve seat, thus limiting the vacuum in the drywell.

The Suppression Chamber to Drywell vacuum relief valve test switches are located on Panel XU-2. The switches are three position (OPEN "X"-NEUTRAL-OPEN "Y") toggle type with spring return to NEUTRAL. The other two switch positions select one of two valves to open. When placed in the OPEN position, a pneumatic operator opens the associated vacuum relief through a lever. The switch spring returns to neutral when released and the vacuum relief closes.

5.20 Steps PC/P-6 and PC/P-7



The Torus Spray Initiation Pressure is defined to be the lowest torus pressure which can occur when 95% of the noncondensibles in the drywell have been transferred to the airspace of the torus. This pressure is utilized to preclude chugging: the cyclic condensation of steam at the downcomer openings of the drywell vents.

When a steam bubble collapses at the exit of the downcomers, the rush of water filling the void (some of it drawn up into the downcomer pipe) induces a severe stress at the junction of the downcomer and the vent header. Repeated application of this stress can cause these joints to experience fatigue failure (i.e., crack) thereby creating a pathway which bypasses the pressure suppression function of the containment. Subsequent steam discharges through the downcomers would directly pressurize the torus airspace rather than being discharged to and condensed in the torus.

45. 295014 1

The following APRM GAFs were recorded on the Core Performance Log after a loss of feedwater heating on Unit Two:

APRM 1	1.03
APRM 2	1.01
APRM 3	1.00
APRM 4	1.02

Which one of the following completes the statements below?

APRM GAFs are ___(1)___.

The most limiting thermal limit for loss of feedwater heating is ___(2)___.

A. (1) satisfactory
   (2) APRAT

B. (1) satisfactory
   (2) FLCPR

C. (1) unsatisfactory
   (2) APRAT

D. (1) unsatisfactory
   (2) FLCPR

Answer: D

K/A:

295014 Inadvertent Reactivity Addition

AA2 Ability to determine and/or interpret the following as they apply to INADVERTENT REACTIVITY ADDITION: (CFR: 41.10 / 43.5 / 45.13)

04 Violation of fuel thermal limits

RO/SRO Rating: 4.1/4.4

Pedigree: New

Objective: LOI-CLS-LP-106-A, Objective 7
Identify the limiting condition and explain the basis of: a. LHGR, b. APLHGR, c. CPR

Reference: None

Cog Level: Fund

Explanation: The APRMs GAFs greater than 1.0 are unsatisfactory. CPR, represented by FLCPR is limiting for plant transients. APLHGR, represented by APRAT is limiting for LOCA.

Distractor Analysis:

Choice A: Plausible because examinee must be able to review a Core Performance Log and know that GAFs greater than 1.0 are unsatisfactory. For the second part, the trainee must be able to differentiate the difference between thermal limits and what they protect.

Choice B: Plausible because examinee must be able to review a Core Performance Log and know that GAFs greater than 1.0 are unsatisfactory. The second part is correct.

Choice C: Plausible because first part is correct, and examinee must be able to differentiate between basis for FLCPR and APRAT.

Choice D: Correct Answer, see explanation.

SRO Basis: N/A

LCO	The APLHGR limits specified in the COLR are the result of the DBA analyses. For two recirculation loops operating, the limit is dependent on exposure. With only one recirculation loop in operation, in conformance with LCO 3.4.1, "Recirculation Loops Operating," the limit is determined by multiplying the exposure dependent APLHGR limit by a multiplier determined by a specific single recirculation loop analysis. Additional APLHGR operating limits adjustments may be provided in the COLR to support analyzed equipment out-of-service operation.

(continued)

APPLICABLE SAFETY ANALYSES	The analytical methods and assumptions used in evaluating the AOOs to establish the operating limit MCPR are presented in References 2, 3, 4, 5, 7, 8, 9, and 10. To ensure that 99.9% of the fuel rods avoid boiling transition during any transient that occurs with moderate frequency, limiting transients are analyzed to determine the largest reduction in critical power ratio (CPR). The types of transients evaluated are loss of flow, increase in pressure and power, positive reactivity insertion, and coolant temperature decrease. The limiting transient yields the largest change in CPR (ΔCPR). When the largest ΔCPR is added to the MCPR SL, the required operating limit MCPR (OLMCPR) is obtained.

The MCPR operating limits derived from the transient analysis are dependent on the operating core flow and power state ($MCPR_f$ and $MCPR_p$ respectively) to ensure adherence to fuel design limits during the worst transient that occurs with moderate frequency as identified in the UFSAR Chapter 15 (Reference 5).

Flow dependent MCPR limits are determined using steady state thermal hydraulic methods (Reference 7) to analyze slow flow runout transients. The $MCPR_f$ limits are dependent on the maximum core flow runout capability of the Recirculation System.

(continued)

46. 295016 1

Plant conditions require the control room to be abandoned in accordance with 0AOP-32, Plant Shutdown from Outside Control Room.

Which one of the following completes the statements below?

Reactor pressure will be controlled using ___(1)___.

Reactor water level will be controlled using ___(2)___.

A. (1) SRVs
   (2) HPCI

B. (1) SRVs
   (2) RCIC

C. (1) Turbine Bypass Valves
   (2) HPCI

D. (1) Turbine Bypass Valves
   (2) RCIC

Answer: B

K/A:
295016 Control Room Abandonment
AK2 Knowledge of the interrelations between CONTROL ROOM ABANDONMENT and the following: (CFR: 41.7 / 45.8)
01 Remote shutdown panel: Plant-Specific

RO/SRO Rating: 4.4/4.5

Pedigree: N/A

Objective: LOI-CLS-LP-062, Objective 3
List the systems that can be controlled from the Remote Shutdown Panel or local control stations.

Reference: None

Cog Level: Fund

Explanation: RCIC can be controlled and monitored from the RSDP. HPCI can be shutdown locally, but cannot be started or operated from the RSDP. CRD and RWCU cannot be operated from the RSDP. CRD can be operated locally.

Distractor Analysis:

Choice A: Plausible because SRV can be used to control pressure. HPCI can be shutdown locally, but cannot be started or controlled from the RSDP.

Choice B: Correct Answer, see explanation.

Choice C: Plausible because BPVs are normally used to control pressure but cannot be operated from the RSDP. HPCI can be shutdown locally, but cannot be started or controlled from the RSDP.

Choice D: Plausible because BPVs are normally used to control pressure but cannot be operated from the RSDP. Part 2 is correct.

SRO Basis: N/A

The equipment that may be operated at the Remote Shutdown Panel and locally to support a Shutdown from outside the Control Room include:

- EPA breakers for the RPS MG sets and Alternate Power Source permit shutting down the Reactor and closing the MSIVs if this action is not completed prior to evacuating the Control Room (located in Cable Spread Area).
- SRVs - Three SRVs (B,E,G) operated from the Remote Shutdown Panel to control Reactor Pressure while in Hot Shutdown and to cool down the Reactor.
- RCIC - can be started and secured locally in both the level control and pressure control modes; and controlled and monitored from the Remote Shutdown Panel. This is the primary means of controlling Reactor water level in Hot Shutdown and during the cooldown.
- CRD pumps - operated locally to provide cooling for the rod drives. A second pump may be started to assist in maintaining Reactor water level.
- Diesel Generators - Started locally and aligned to the E buses if power is loss to an E bus.
- RHR loop B - initially used for Suppression Pool cooling and then for Shutdown cooling when Reactor pressure is reduced to 50-100 psig. Operated locally and monitored at the Remote Shutdown Panel.
- RHR Service Water - operated locally to support RHR System operation.
- Nuclear Service Water - operated locally to support RHR System operation.
- Condensate System - Condensate Booster Pumps are tripped locally and the system is aligned to prevent injection to the Reactor vessel prior to Reactor pressure reaching 500 psig during the cooldown.
- HPCI - secured locally when no longer needed to maintain Reactor water level. If HPCI does not automatically initiate, it is not used to support the Shutdown.

47. 295017 1

An Off-Site Dose Projection calculation is being performed for an unmonitored ground release from the Reactor Building in order to complete a hard copy Emergency Notification Form (ENF).

The following Met Tower Data is provided by the Process Computer:

Ambient Temp:	80 Deg F.
Upper Wind Direction:	18.00 Deg
Lower Wind Direction:	15.00 Deg
Upper Wind Speed:	8.00 MPH
Lower Wind Speed:	4.00 MPH
Stability Class:	D

Which one of the following completes the statements below?

The Wind Direction given is measured ___(1)___.

For the given conditions, ___(2)___ wind speed and direction should be used.

A. (1) to
(2) lower

B. (1) to
(2) upper

C. (1) from
(2) lower

D. (1) from
(2) upper

Answer: C

K/A:
295017 High Off-Site Release Rate
AK1 Knowledge of the operational implications of the following concepts as they apply to HIGH OFF-SITE RELEASE RATE: (CFR: 41.8 to 41.10)
03 Meteorological effects on off-site release

RO/SRO Rating: 2.7/3.4

Pedigree: New

Objective: LOI-CLS-LP-301-A, Objective 6
Determine data required for offsite dose projection in accordance with AD-EP-ALL-0202, Emergency Response Offsite Dose Assessment, and PEP-03.6.1, Release Estimates Based Upon Stack/Vent Readings.

Reference: None

Cog Level: Fund.

Explanation: Met tower wind direction is always given 'from'. For a hard copy ENF, lower wind direction and speed are always used as indicated in 0PEP-02.6.21.

Distractor Analysis:

Choice A: Plausible because examinee must know the convention of Met Tower Data. It is either going to be 'to' or 'from'. Second part is correct.

Choice B: Plausible because examinee must know the convention of Met Tower Data. It is either going to be 'to' or 'from'. Given both upper and lower wind speed and direction, examinee must know which to use. Regardless of the release point, for a hard copy ENF, lower wind direction and speed are always used.

Choice C: Correct Answer, see explanation.

Choice D: Plausible because part 1 is correct. Given both upper and lower wind speed and direction, examinee must know which to use. Regardless of the release point, for a hard copy ENF, lower wind direction and speed are always used.

SRO Basis: N/A

20. **Rapid Assessment:** Rapid Assessment may be used to produce a dose projection with minimal user input, and is intended for use by on-shift personnel during events that progress quickly. It allows the development of a conservative but reasonable dose projection without excessively distracting staff from performing actions to mitigate the event. Many assumptions and predetermined standards are used in Rapid Assessment to limit the amount of data plant personnel must enter prior to completing the dose assessment.

5.5 Rapid Dose Assessment (continued)

4. If this is a spent fuel event, then perform the following:

   a. Select 'Damaged Spent Fuel Assembly'.

   b. Ensure the 'Last Irradiated' checkbox is checked.

   c. If the date the fuel assembly was last in the reactor is known, then enter the date in the 'Last Irradiated' textbox.

   d. If the date the fuel assembly was last in the reactor cannot be determined, then use default value.

5. Enter the meteorological data as follows:

   a. Select the applicable meteorological tower sensors by checking the corresponding checkbox in the 'Use' column of the Meteorological Data table.

   b. If the meteorological data is available from the plant computer system, then perform the following:

      (1) Enter the 'Wind Speed' in the appropriate units.

      (2) Enter the 'Wind Direction' (degrees from).

      (3) Enter the 'ΔT'.

   c. Tower sensor height should be selected based on the height of the release pathway whenever possible.

Guidance for Completion of Emergency Notification Form

LINE NO. **INSTRUCTIONS**

NOTE:	Information for Line 9 may be obtained from the STA (if in the Control Room) or the Radiological Control Manager (if in the EOF).
NOTE:	Information may not be available for Initial Notifications.

CAUTION:
Met Data entered on Line 9 must match Met Data used for PAR determination. Met Data on Line 9 may need to be changed to match data used for PAR determination.

______9 **"METEOROLOGICAL DATA"**

If using WebEOC and importing Met Data select "Import Plant/MET Data." Imported Met Data is current data. Use the "Clear Plant/Met Data" to clear.

Enter lower wind direction and wind speed if completing hard copy ENF.

Access information from ERFIS, National Weather Service or a meteorological service provider (see EPL-001, Emergency Phone List, Brunswick, Attachment 7, for telephone numbers) to complete information as follows.

- Enter **"Wind Direction"** in degrees. Note: Wind direction must be **"from".**
- Enter **"Wind Speed"** in mph.
- Enter **"Precipitation"** in inches.
- Mark appropriate block for **"Stability Class"**.

48. 295018 1

Unit Two is operating at rated power when the following alarms and indications are observed:

UA-03 (2-4) *TBCCW Pump Disch Header Press Low* alarm seals in.
TBCCW Discharge Pressure, TCC-PI-566-1 on XU-2, indicates 38 psig.
TBCCW Pump 2A is running.
TBCCW Pump 2B has tripped (no light indications).
TBCCW Pump 2C is aligned and running on Unit One.

Which one of the following completes the statements below?

The first action required IAW 0AOP-17.0, Turbine Building Closed Cooling Water System Failure, is to ___(1)___.

The 2-TCC-TV-609, TBCCW Heat Exchange Outlet Temperature Control Valve, ___(2)___ to provide maximum cooling to TBCCW.

A. (1) reduce Reactor Recirc flow to the 0ENP-24.5 limit
(2) opens

B. (1) reduce Reactor Recirc flow to the 0ENP-24.5 limit
(2) closes

C. (1) manually scram the reactor and enter 2EOP-01-RSP
(2) opens

D. (1) manually scram the reactor and enter 2EOP-01-RSP
(2) closes

Answer: B

K/A:
295018 Partial or Complete Loss of Component Cooling Water
AK3 Knowledge of the reasons for the following responses as they apply to PARTIAL OR COMPLETE LOSS OF COMPONENT COOLING WATER: (CFR: 41.5 / 45.6)
02 Reactor power reduction

RO/SRO Rating: 3.3/3.4

Pedigree: New

Objective: LOI-CLS-LP-302-H, Objective 4b
Given plant conditions, determine the required supplementary actions in accordance with the following AOPs:
b. 0AOP-17.0, TBCCW System Failure ***(LOCT)***

Reference: None

Cog Level: High

Explanation: IAW AOP 17.0, conditions are met to reduce reactor power to the 0ENP-24.5 limit. This will give time for system pressure to recover and provide adequate cooling to components. The TCV on TBCCW closes to force more water through the Hx. All the individual components cooled by TBCCW that have TCVs, open to provide more cooling.

Distractor Analysis:

Choice A: Plausible because part 1 is correct and individual component TCVs open to provide maximum cooling.

Choice B: Correct Answer, see explanation.

Choice C: Plausible because on a total loss of TBCCW, a reactor Scram is inserted. Part 2 is plausible because individual component TCVs open to provide maximum cooling.

Choice D: Plausible because on a total loss of TBCCW, a reactor Scram is inserted. Part 2 is correct.

SRO Basis: N/A

4.0 OPERATOR ACTIONS

NOTE

The following should be considered for establishment as critical parameters during performance of this procedure: .... ☐

- TBCCW pressure
- TBCCW heat exchanger outlet temperature
- SW-V3 and V4 position

4.1 Immediate Actions

None

4.2 Supplementary Actions

1. **Place** any available TBCCW pump(s) in service to the affected unit. .... ☐

2. **IF** power is **NOT** available to the TBCCW pumps
   OR the pumps will **NOT** start,
   THEN perform the following:

   a. **IF** the affected unit 4160V Bus C
      OR Bus D is de-energized,
      THEN enter 0AOP-36.1, Loss of Any 4160V Buses or 480V
      E-Buses **AND perform** concurrently with this procedure.. .... ☐

3. **IF** only one TBCCW pump is in service
AND TBCCW pressure is less than 42 psig,
THEN perform the following:

a. **Reduce** reactor power with recirc flow in accordance with 0ENP-24.5, Form 2, Immediate Reactor Power Reduction Instructions. ☐

b. **IF** TBCCW pressure is greater than 42 psig within 4 minutes,
THEN perform Section 4.2 Step 6, on page 7. ☐

c. **IF** TBCCW pressure is **NOT** greater than 42 psig within 4 minutes,
THEN perform Section 4.2 Step 7, on page 9. ☐

NOTE
A total loss of TBCCW is defined as system pressure less than 42 psig with all available pumps operating and expectations are that normal cooling can **NOT** be quickly re-established. ☐

7. **IF** there has been a total loss of TBCCW,
THEN:

a. **Insert** a manual scram. ☐

b. **Enter** 1EOP-01-RSP(2EOP-01-RSP), Reactor Scram Procedure **AND perform** concurrently with this procedure. ☐

The outlet temperature of the heat exchangers is controlled by a temperature control valve (TCC-TV-606) which is located in a line which bypasses the heat exchangers. The temperature control valve will increase or decrease the bypass flow to maintain the supply header temperature in the desired range. Maximum TCC outlet temperature will be maintained ≤ 100°F.

The temperature control valves used on the TBCCW system and cooled loads are air operated butterfly valves. The TBCCW system Hx outlet temperature control valve, TCC-TV-606, fails closed on a loss of air. This places full system flow through the TBCCW heat exchangers, and should result in low cooling water temperature. All other TCVs are in series with the cooling water flow through the cooled component. These valves fail open on a loss of air, providing maximum cooling.

49. 295019 1

With Unit Two at rated power, the following alarms and indications are noted:

UA-01 (3-2) *Air Compr D Trip*	Alarm sealed in
UA-01 (4-4) *Inst Air Press Low*	Alarm sealed in
Air Compressor 2B	Running
Instrument Air header pressure	101 psig

Which one of the following is required IAW 0AOP-20.0, Pneumatic (Air/Nitrogen) System Failures?

The operator is required to verify that:

A. IA-PV-722-1 and IA-PV-722-2, Interruptible Air Isolation Valves, have automatically closed.

B. RNA-SV-5262 and RNA-SV-5261, PNS Drywell Isolation Valves, have automatically closed.

C. SA-PV-706-1 and SA-PV-706-2, Service Air Isolation Valves, have automatically closed.

D. SA-PV-5067, Serv Air Dryer 2A Bypass Pressure Control Valve, has automatically opened.

Answer: C

K/A:
295019 Partial or Complete Loss of Instrument Air
AA1 Ability to operate and/or monitor the following as they apply to PARTIAL OR COMPLETE LOSS OF INSTRUMENT AIR: (CFR: 41.7 / 45.6)
04 Service air isolations valves: Plant-Specific

RO/SRO Rating: 3.3/3.2

Pedigree: Bank

Objective: LOI-CLS-LP-302-K, Objective 2
Given plant conditions, determine any automatic actions expected to occur in accordance with AOP-20.0, Pneumatic (Air/Nitrogen) System Failures.

Reference: N/A

Cog Level: Higher

Explanation: See Notes.

Distractor Analysis:

Choice A: Plausible because this is a manual action but does not occur automatically

Choice B: Plausible because although it might be desired, it is not an automatic action.

Choice C: Correct Answer, see explanation.

Choice D: Plausible because this occurs at 98 psig.

SRO Basis: N/A

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3.0 AUTOMATIC ACTIONS

1. **IF** service air header pressure lowers to approximately 110 psig, **THEN** Service Air Compressor 1B and Service Air Compressor 2B START and LOAD. ☐

2. **IF** service air header pressure lowers to 105 psig, **THEN** SA-PV-1&2 (Service Air Isol Vlvs), CLOSE. ☐

3. **IF** service air header pressure lowers to 98 psig, **THEN** 1(2)-SA-PV-5067 (Serv Air Dryer1(2)A Bypass Pressure Control Valve), begins to OPEN. ☐

4.2 Supplementary Actions (continued)

NOTE

Attachment 2, Components Required To Perform A Safety-Related Function After Loss of Normal Instrument Air and Attachment 3, Components Required To Fail In A Safe Position Upon Loss Of Instrument Air are provided in the event that instrument air pressure lowers to the point at which components start to drift toward their failed positions. .... ☐

CAUTION

Isolation of the interruptible instrument air header has the potential to cause a reactor scram from loss of reactor vessel level control due to Condensate and Feedwater System minimum flow valves failing open. .... ☐

g. **IF** all the following occur:

- Instrument air pressure lowers to less than 100 psig.......... ☐
- The leak is determined to be from the interruptible instrument air header.......... ☐
- Continued operation would be detrimental to the plant.......... ☐

THEN close IA-PV-722-1&2 (Intrpt Air Isol Vlvs). .... ☐

50. 295020 1

Unit Two was operating at rated power. An inadvertent Core Spray initiation signal resulted in the following plant conditions:

RPV level	170 inches
RPV pressure	950 psig
Drywell pressure	2.0 psig
Drywell temperature	152º F
Torus pressure	1.6 psig

Which one of the following identifies the action that is required to control Containment parameters?

A. Spray the Torus IAW SEP-03, Torus Spray Procedure.

B. Spray the Drywell IAW SEP-02, Drywell Spray Procedure.

C. Vent the Drywell IAW 2OP-10, Standby Gas Treatment System Operating Procedure.

D. Defeat Drywell Cooler LOCA Lockout per SEP-10, Circuit Alteration Procedure, and restart the Drywell Coolers.

Answer: D

K/A:
295020 Inadvertent Containment Isolation
AK3 Knowledge of the reasons for the following responses as they apply to INADVERTENT CONTAINMENT ISOLATION: (CFR: 41.5 / 45.6)
02 Drywell/containment pressure response

RO/SRO Rating: 3.3/3.5

Pedigree: 2007 NRC Exam

Objective: LOI-CLS-LP-300-K, Objective 19a
Explain the reason for defeating the following system isolations and initiations while in the EOPs: a. Drywell Cooler LOCA lockout

Reference: None

Cog Level: High

Explanation: Although the reason for Drywell parameters is not discussed, the examinee must know the reason in order to know the proper response. Therefore this question addresses the K/A. A 2-part question could ask for the reason, but the answer implies the reason. Drywell Coolers have tripped because of the LOCA signal. SEP-10 is permitted to defeat the Drywell Cooler LOCA lockout and restart the Drywell Coolers.

Distractor Analysis:

Choice A: Plausible because Torus Spray will lower Containment pressure, but cannot be used below 2.5 psig Torus pressure.

Choice B: Plausible because Drywell Spray will lower Drywell pressure, but cannot be used below 2.5 psig Drywell pressure.

Choice C: Plausible because venting the Drywell will lower Drywell pressure, but cannot be performed with Drywell pressure above 1.7 psig.

Choice D: Correct Answer, see explanation.

SRO Basis: N/A

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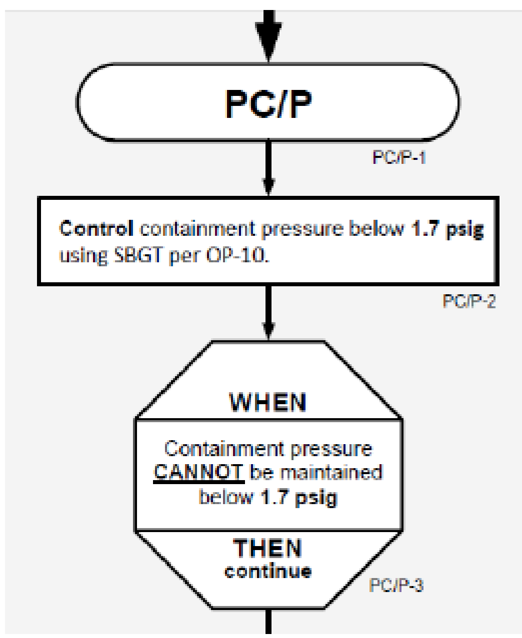
5.17 Steps PC/P-1 through PC/P-3

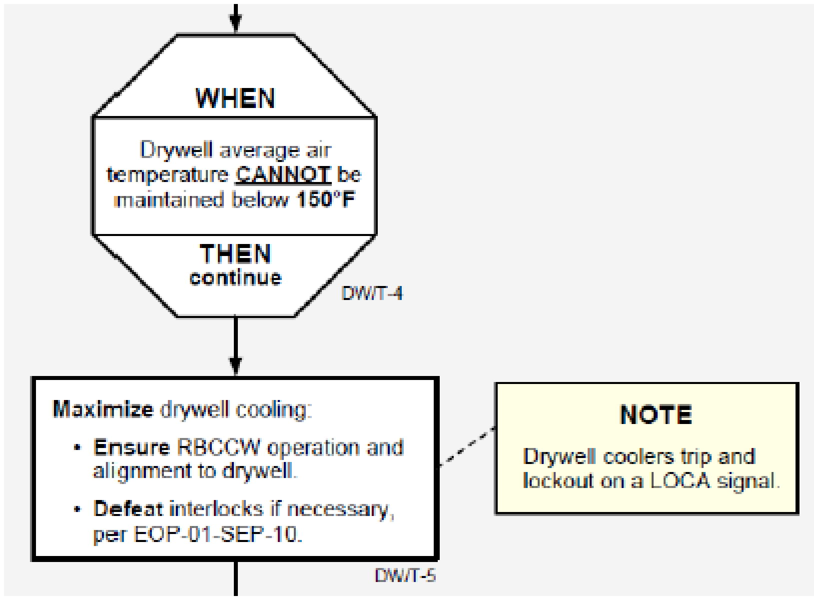


5.18 Step PC/P-4

IF	THEN
Containment pressure reduction is required to either: • Restore and maintain adequate core cooling • Reduce total offsite radiation dose	**Vent** containment per EOP-01-SEP-01 • **Exceed** offsite radioactivity release rates if necessary • IF pneumatic supply degrading, THEN commence EOP-01-FSG-05
Containment venting **CANNOT** be performed **OR** is **NOT** effective	**Perform** EDMG-003 to vent containment
E-bus load stripping prevents containment spray	**Align** electrical power per EOP-01-SBO-14 • 785 KW required
Torus pressure drops to **2.5 psig**	**Terminate** torus sprays
Drywell pressure drops to **2.5 psig**	**Terminate** drywell sprays

PC/P-4

5.11 Steps DW/T-4 and DW/T-5



51. 295021 1

Unit Two has just entered MODE 4. RHR Loop A is operating in Shutdown Cooling. DG4 is under clearance.

Subsequently, a Loss Of Off-Site Power occurs and a Shutdown Cooling flowpath cannot be reestablished.

Which one of the following completes the statements below?

An available method for feed and bleed operation IAW 0AOP-15.0, Loss of Shutdown Cooling is ___(1)___.

The reason this method is preferred is ___(2)___.

A. (1) Feed with CRD Pump 2A IAW 2OP-08, CRD Hydraulic System Operating Procedure. Bleed by RWCU Reject IAW 2OP-14, RWCU System Operating Procedure.
   (2) to provide flow through the bottom head region of the core.

B. (1) Feed with CRD Pump 2A IAW 2OP-08, CRD Hydraulic System Operating Procedure. Bleed by Maintaining RPV Level Using the Main Steam Line Drains IAW 2OP-25, Main Steam System Operating Procedure.
   (2) because power is available to establish these conditions.

C. (1) Feed with Core Spray Loop 2B IAW 2OP-18, Core Spray System Operating Procedure. Bleed by RWCU Reject IAW 2OP-14, RWCU System Operating Procedure.
   (2) to provide flow through the bottom head region of the core.

D. (1) Feed with Core Spray Loop 2B IAW 2OP-18, Core Spray System Operating Procedure. Bleed by Maintaining RPV Level Using the Main Steam Line Drains IAW 2OP-25, Main Steam System Operating Procedure.
   (2) because power is available to establish these conditions.

Answer: B

K/A:
295021 Loss of Shutdown Cooling
AK3 Knowledge of the reasons for the following responses as they apply to LOSS OF SHUTDOWN COOLING: (CFR: 41.5 / 45.6)
02 Feeding and bleeding reactor vessel

RO/SRO Rating: 3.3/3.4

Pedigree: New

Objective: LOI-CLS-LP-302-L, Objective 6
Describe how Alternate Shutdown Cooling is used to provide reactor cooling when other methods have failed in accordance with AOP-15.0.

Reference: None

Cog Level: High

Explanation: CRD 2A would be available from E3. Bleed by RWCU flowpath is unavailable due to LOOP. RWCU must be in service. Core Spray 2B is not available because D4 is under clearance. Bleed by Maintaining RPV Level Using the Main Steam Line Drains is available because the valves are either DC powered or Diesel-backed MCCs.

Distractor Analysis:

Choice A: Plausible because CRD is available, but second part is not correct because RWCU is not available.

Choice B: Correct Answer, see explanation.

Choice C: Plausible because Core Spray is powered from E Busses, but E4 is unavailable. Second part is not correct because RWCU is not available.

Choice D: Plausible because Core Spray is powered from E Busses, but E4 is unavailable. Second part is correct.

SRO Basis: N/A

4.2 Supplementary Actions (continued)

12. **IF** necessary to minimize reactor coolant temperature rise, **THEN perform** one of the following feed and bleed combinations: .... ☐

<table>
<tr><th>FEED</th><th>BLEED</th></tr>
<tr><td>Cond/FW in accordance with:
1OP-32
2OP-32</td><td>RWCU Reject in accordance with:
1OP-14
2OP-14</td></tr>
<tr><td>CRD in accordance with:
1OP-08
2OP-08</td><td>Reactor Water Level Control using Main Steam Lines in accordance with:
1OP-32
2OP-32</td></tr>
<tr><td>Core Spray in accordance with:
1OP18
2OP-18</td><td rowspan="2">Maintaining RPV Level Using the Main Steam Line Drains with:
1OP-25
2OP-25</td></tr>
<tr><td>LPCI in accordance with:
1OP-17
2OP-17</td></tr>
</table>

8.18 Reactor Water Level Control Using Main Steam Lines

8.18.1 Initial Conditions

1. All applicable prerequisites listed in Section 4.0 are met. ☐
2. Reactor in Mode 4 or 5. ☐
3. One of the following methods is utilized to provide forced circulation through the reactor:
   a. A reactor recirculation pump is running, ☐

      OR

   b. RHR System is in operation providing a shutdown cooling flowpath in accordance with 1OP-17 with or without RHR SW in operation, **AND** the following valves positioned for operation without RHR SW: ☐
      – *E11-F048A(B)* open ☐
      – *E11-F003A(B)* closed ☐
      – *E11-F047A(B)* closed ☐

4.3 Interrelationships With Other Systems

4.3.1 Reactor Protection System

A loss of both RPS Systems (Figure 25-7, 25-7A, 25-7B) will result in MSIV closure due to the initiation of a Group I Isolation signal. A loss of either RPS System will result in half of the logic being satisfied for an isolation signal yet no valve repositioning will occur.

Specifically a loss of RPS A will result in a loss of power to the Inboard MSIV's AC solenoid and the Outboard MSIV's DC solenoid. The DC solenoid power is lost indirectly as a result of the PCIS A logic also losing power when RPS A is de-energized. Likewise a loss of RPS B will result in a loss of power to the inboard MSIV's DC solenoid and the outboard MSIV's AC solenoid. The DC solenoid power is lost indirectly as a result of the PCIS B logic also losing power when RPS B is de-energized. No MSIV repositioning occurs on a loss of one RPS because at least either the AC or DC solenoid remains energized and both solenoids must de-energize to produce valve closure.

52. 295023 1

Unit Two is performing refueling operations when the refueling SRO reports that a spent fuel bundle has been dropped in the cattle chute. The following annunciators are in alarm on Panel 2-UA-3:

(2-3): *Rx Bldg Roof Vent Rad High*
(2-7): *Area Rad Rx Bldg High*
(3-7): *Area Rad Refuel Floor High*
(4-5): *Process Rx Bldg Vent Rad High*

Which one of the following is an Immediate Operator Action IAW 0AOP-5.0, Radioactive Spills, High Radiation, and Airborne Activity?

A. Verify Group 6 isolation.

B. Notify RP to perform area radiation surveys.

C. Ensure Control Room Emergency Ventilation System (CREVS) in operation.

D. Isolate Reactor Building Ventilation and place Standby Gas Treatment (SBGT) trains in operation.

Answer: C

K/A:
295023 Refueling Accidents
AA2 Ability to determine and/or interpret the following as they apply to REFUELING ACCIDENTS: (CFR: 41.10 / 43.5 / 45.13)
01 Area radiation levels

RO/SRO Rating: 3.6/4.0

Pedigree: 2014 NRC Exam

Objective: LOI-CLS-LP-302-J, Objective 2
Given plant conditions with spent fuel damage and a high airborne activity problem in progress, determine if the appropriate automatic actions have occurred in accordance with AOP-05.0, Radioactive Spills High Radiation and Airborne Activity

Reference: None

Cog Level: High

Explanation: None of the present alarms provide any automatic actions. An immediate action of AOP-5.0 is to ensure CREV is in service.

Distractor Analysis:

Choice A: Plausible because this action would be expected if Process Rx Bldg Hi-Hi was in alarm.

Choice B: Plausible because this is a Supplementary action of AOP-5.0.

Choice C: Correct Answer, see explanation.

Choice D: Plausible because this action would be expected if Process Rx Bldg Hi-Hi was in alarm.

SRO Basis: N/A

Unit 2
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Page 1 of 1

PROCESS RX BLDG VENT RAD HIGH

AUTO ACTIONS

NONE

CAUSE

1. High airborne activity in Reactor Building ventilation exhaust plenum
2. Circuit malfunction

OBSERVATIONS

1. *REACTOR BUILDING VENT RAD RECORDER, D12-RR-R605,* Channel A or B, indicates high radiation level
2. *PROCESS REACTOR BUILDING VENTILATION RADIATION MONITOR, 2-D12-RM-K609A/B*, indicates greater than 3 mR/hr on Panel P606

ACTIONS

1. **ENTER** 0EOP-03-SCCP, Secondary Containment Control Procedure.
2. **REFER** to 0AOP-05.0, Radioactive Spills, High Radiation, and Airborne Activity.
3. **NOTIFY** E&RC Health Physics due to potential to affect radiation levels.
4. **IF** alarming condition clears, **PERFORM** SV 100 Recorder Alarm Acknowledge Function, per 0OI-63.

DEVICE/SETPOINTS

D12-RR-R605, red or blue pen	3 mR/hr

POSSIBLE PLANT EFFECTS

1. Possible release to environs
2. If airborne activity rises to 4 mR/hr, Reactor Building HVAC isolation, Group 6 isolation, drywell purge isolation, and initiation of Standby Gas Treatment System occurs

REFERENCES

AREA RAD REFUEL FLOOR HIGH

AUTO ACTIONS

NONE

CAUSE

1. High radiation level in cask washdown area
2. Refueling cavity water seal failure
3. Dry Fuel Storage (DFS) cask loading
4. Circuit malfunction

OBSERVATIONS

ARM indicator and trip unit *HIGH* light illuminated on Panel P600

ACTIONS

1. **REFER** to 0EOP-03-SCCP, Secondary Containment Control Procedure, Table 3 **AND ENTER** EOP-03-SCCP as appropriate.
2. **REFER** to 0AOP-05.0, Radioactive Spills, High Radiation, and Airborne Activity.
3. **SUSPEND** refueling operation if due to fuel pool low level from refueling cavity water seal leakage.
4. **SUSPEND** refuel floor dry fuel storage cask processing if in progress.

NOTE:	An off scale indication on two or more refuel floor ARMs is indicative of a criticality event.

5. **IF** alarm is due to a dry fuel storage cask accidental criticality, **THEN EVALUATE** reportability in accordance with 0OI-01.07, Notifications.

DEVICE/SETPOINTS

ARM Channel 29 K2	40 mR/hr

RX BLDG ROOF VENT RAD HIGH

AUTO ACTIONS

NONE

CAUSE

1. High noble gas concentration in Reactor Building vent exhaust
2. Circuit malfunction

OBSERVATIONS

1. *REACTOR BLDG VENT NOBLE GAS MONITOR, CAC-AQH-1264-3*, on Panel XU-55, in alarm
2. *RX BLDG ROOF VENT MON RECORDER*, *CAC-AR-1264,* trending up

ACTIONS

1. **ENTER** 0EOP-04-RRCP, Radioactivity Release Control, **AND EXECUTE** concurrently with this procedure.
2. **IF** steam leaks in the Reactor Building are causing local area radiation levels or ambient temperatures to rise, **THEN ENTER** 0EOP-03-SCCP, Secondary Containment Control Procedure, as appropriate.
3. **REFER** to 0AOP-05.0, Radioactive Spills, High Radiation, and Airborne Activity.
4. **NOTIFY** E&RC Health Physics due to potential to affect radiation levels.

4.0 OPERATOR ACTIONS

NOTE

The following should be considered for establishment as critical parameters during performance of this procedure: ☐

- Area radiation levels
- Personnel habitability in the affected area

4.1 Immediate Actions

1. **IF** a fuel assembly was dropped or damaged,
THEN ensure the Control Room Emergency Ventilation System (CREVS) is in operation. {7.1.1} ☐

4.2 Supplementary Actions

NOTE

Consideration should be given to spill/release location and egress routes when announcing evacuation. ☐

1. **Evacuate** unnecessary personnel from the affected area. ☐

2. **Review** Emergency Action Levels in accordance with 0PEP-02.1, Initial Emergency Actions, with regard to an area or building evacuation. ☐

53. 295024 1

Following a loss of feedwater on Unit One, HPCI initiated on low reactor water level then tripped on high reactor water level.

Current plant conditions are:

Reactor water level	150 inches, steady
A-01 (3-1) *HPCI Turb Trip*	alarm sealed in
A-01 (4-1) *HPCI Turb Trip Sol Ener*	alarm sealed in
A-05 (5-5) *Pri Ctmt Hi/Lo Press*	alarm sealed in
A-05 (5-6) *Pri Ctmt Press Hi Trip*	alarm sealed in
HPCI Initiation Signal/Reset white light	LIT
HPCI High Water Level Signal Reset white light	LIT

Which one of the following is the <u>minimum</u> required operator action(s) (if any) to allow HPCI injection to the reactor?

A. No operator action is required.

B. Manually open 1-E41-F006, HPCI Injection Vlv.

C. Depress the High Water Level Signal Reset push button.

D. Depress the Isolation Sig/Reset push buttons and then manually open 1-E41-F006, HPCI Injection Vlv.

Answer: C

K/A:
295024 High Drywell Pressure
G2.4.50 Ability to verify system alarm setpoints and operate controls identified in the alarm response manual. (CFR: 41.10 / 43.5 / 45.3)

RO/SRO Rating: 4.2/4.0

Pedigree: NRC Exam 10-1

Objective: LOI-CLS-LP-19, Objective 3m, 16c
Given plant conditions, predict how the HPCI System will respond to the following events:
m. High RPV water level
Given plant conditions, determine if the following actions should occur:
c. HPCI System automatic initiation

Reference: None

Cog Level: Higher

Explanation: A Reactor High Water Level trip is initiated and the signal seals-in when a high level is sensed by two instruments. Once the high level condition clears, the trip is reset by a subsequent Reactor Low Level 2 signal or upon depressing the Panel P601 High Level Trip Reset pushbutton. A High Drywell Pressure Initiation signal will not reset the High Water Level trip. This question satisfied the K/A because HPCI receives an initiation signal from high drywell pressure.

Distractor Analysis:

Choice A: Plausible because if there was not a high level trip, high drywell pressure would automatically initiate HPCI with no action.

Choice B: Plausible because the high water level trip does automatically reset on LL2. Injection valve requires active initiation condition + Stop Valve (V8) & Steam Supply Valve (F001) not full closed to automatically open. Relay timing has caused HPCI initiation with injection valve not opening (LER 2-90-015).

Choice C: Correct Answer, see explanation

Choice D: Plausible because high water level does not automatically reset due to Hi DW Press. Injection valve requires active initiation condition + Stop Valve (V8) & Steam Supply Valve (F001) not full closed to automatically open. Relay timing has caused HPCI initiation with injection valve not opening (LER 2-90-015)

SRO Basis: N/A

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ACTIONS (Continued)

2. If reactor vessel water level drops below 206 inches and it is desired to operate HPCI in level or pressure control, perform the following:
   a. Reset the high water level trip by depressing High Water Level Signal Reset push button, E41-S25.
   b. Operate HPCI in accordance with the following applicable attachment of 1OP-19:
      - HPCI Instructional Aid (HPCI Injection in EOPs)
      - HPCI Instructional Aid (HPCI Pressure Control in EOPs)

A Reactor High Water Level trip is initiated and the signal seals-in when a high level is sensed by two instruments, both powered from 125 VDC Bus. Once the high level condition clears, the trip is reset by a subsequent Reactor Low Level 2 signal or upon depressing the Panel P601 High Level Trip Reset pushbutton. A High Drywell Pressure Initiation signal will not reset the High Water Level trip.

During a HPCI Turbine start, pump suction pressure could possibly drop below the trip initiation setpoint. For this reason, a 13 second time delay has been added to prevent spurious trips upon system initiation.

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54. 295025 1

The following annunciators/indications are observed on Unit One:

A-05 (3-6) *Reactor Vess Hi Press Trip*
A-03 (1-1), *Safety or Depress Vlv Leaking*
A-03 (1-10) *Safety / Relief Valve Open*
SRVs A, C, D, E, F, G, H, and K have cycled open
Both Recirc Pumps have tripped

Which one of the following identifies the highest reactor pressure reached?

A. 1060 psig

B. 1130 psig

C. 1140 psig

D. 1150 psig

Answer: C

K/A:
295025 High Reactor Pressure
EK2 Knowledge of the interrelations between HIGH REACTOR PRESSURE and the following: (CFR: 41.7 / 45.8)
05 Safety/relief valves: Plant-Specific

RO/SRO Rating: 4.1/4.2

Pedigree: Modified from 2014 NRC Exam SRO question. Added opening of 2 more SRVs and Recirc Pump trip. Deleted second part which dealt with basis for EOP actions. See Notes Section for original question.

Objective: LOI-CLS-LP-020, Objective 9
List the SRV pressure relief setpoints.

Reference: None

Cog Level: Higher

Explanation: Since 8 SRVs have opened, pressure must have reached at least 1140 psig. See SRV setpoints in Notes Section. Recirc Pumps trip at 1137.8 psig, and the reactor scrams at 1060 psig.

Distractor Analysis:

Choice A: Plausible because RPS trips at 1060 psig

Choice B: Plausible because 4 SRV have opening setpoints of 1130 psig

Choice C: Correct Answer, see explanation.

Choice D: Plausible because 3 SRVs have opening setpoints of 1150 psig.

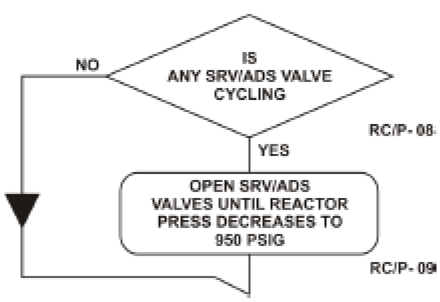
SRO Basis: N/A

From various SDs:

Reactor Protection System scram 1060 psig
SRV lift setpoints 4 @ 1130 (A, C, F, G), 4 @ 1140 (D, E, K, H), 3 @ 1150 (B, L, J)
Alternate Rod Insertion scram 1137.8 psig
Recirculation Pump Trip 1137.8 psig

2014 NRC SRO Question:
During an ATWS on Unit One the following annunciators/indications are observed:

A-05 (3-6) *Reactor Vess Hi Press Trip*
A-03 (1-10) *Safety / Relief Valve Open*
SRV A, C, F, and G are cycling open



Which one of the following completes the statements below?

The highest that reactor pressure reached was at least ___(1)___ psig.

The bases for Step RC/P-09 of LPC is to ___(2)___.

A. (1) 1060
(2) conserve SRV accumulator pressure
B. (1) 1060
(2) minimize heat discharged to the suppression pool
C. (1) 1130
(2) conserve SRV accumulator pressure
D. (1) 1130
(2) minimize heat discharged to the suppression pool

55. 295026 1
Unit One is operating at rated power. A Safety Relief Valve has failed open. Torus temperature is 96°F and rising.

Which one of the following completes the statements below IAW PCCP?

A reactor scram is required ___(1)___ torus temperature reaches 110°F.

This assures ___(2)___.

A. (1) before
(2) torus temperature will remain in the safe region of the Heat Capacity Temperature Limit graph

B. (1) before
(2) reactor shutdown is attempted by control rod insertion before the requirement to initiate SLC is reached

C. (1) when
(2) torus temperature will remain in the safe region of the Heat Capacity Temperature Limit graph

D. (1) when
(2) reactor shutdown is attempted by control rod insertion before the requirement to initiate SLC is reached

Answer: B

K/A:
295026 Suppression Pool High Water Temperature
EK1 Knowledge of the operational implications of the following concepts as they apply to SUPPRESSION POOL HIGH WATER TEMPERATURE: (CFR: 41.8 to 41.10)
02 Steam condensation

RO/SRO Rating: 3.5/3.8

Pedigree: Bank

Objective: LOI-CLS-LP-300-L, Objective 8a
Given the Primary Containment Control Procedure, and plant conditions, determine if the following action are required: Manual Reactor Scram.

Reference: N/A

Cog Level: High

Explanation: See Notes.

Distractor Analysis:

Choice A: Plausible because first part is correct, but basis is not. Basis is for BIIT.

Choice B: Correct Answer, see explanation

Choice C: Plausible because 110°F is the cutoff, but procedure reads 'before'. Basis is also incorrect. this is basis for BIIT.

Choice D: Plausible because 110°F is the cutoff, but procedure reads 'before'. Second part is correct.

SRO Basis: N/A

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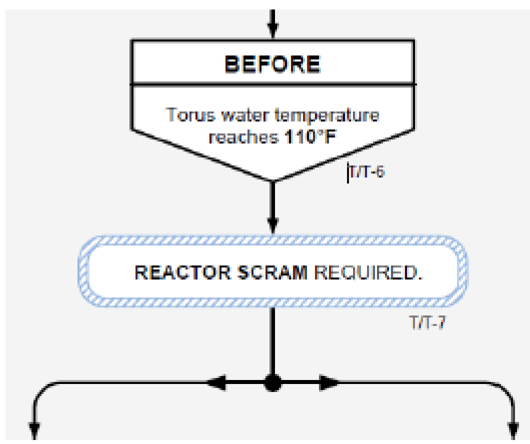
5.4 Steps T/T-6 and T/T-7



The reactor scram flag indicates that a reactor scram is required "before" torus water temperature reaches 110°F to ensure the reactor is scrammed and shutdown, by control rod insertion, before the requirement for boron injection is reached. The 110°F value is the single value selected for the Boron Injection Initiation Temperature. This value also corresponds to the Technical Specification value which requires a manual reactor scram.

A reactor scram is effected indirectly, through entry of RSP, rather than through an explicit direction in PCCP, to ensure that RPV level, RPV pressure and reactor power are properly coordinated following the scram.

The Boron Injection Initiation Temperature (BIIT) is a function of reactor power and is the torus temperature before which boron injection must be initiated if a reactor depressurization, due to exceeding the Heat Capacity Temperature Limit (HCTL), is to be precluded. This temperature is 110°F.

The Boron Injection Initiation Temperature is defined to be the greater of:

- Torus temperature at which initiation of a reactor scram is required by Technical Specifications
- The highest torus temperature at which initiation of boron injection using SLC will result in injection of HSBW of boron before torus temperature exceeds HCTL.

The second bullet is a function of reactor power; a higher reactor power level causes higher integrated heat energy to be rejected to the torus, thus requiring a lower torus temperature for initiation of boron injection, if HCTL is not to be exceeded before reactor shut down is achieved.

At Brunswick, a single value is used for BIIT (110°F) for procedure simplification.

4.2 Supplementary Actions

1. **IF** an SRV is stuck OPEN,
   THEN reduce reactor power in accordance with 0ENP-24.5, Form 2, Immediate Reactor Power Reduction Instructions, in anticipation of a reactor scram. .... ☐

2. **Monitor** primary containment parameters. .... ☐

3. **Enter** 0AOP-14.0, Abnormal Primary Containment Parameters,
   AND perform concurrently with this procedure. .... ☐

4. **IF** suppression pool water temperature exceeds 95°F,
   THEN enter 0EOP-02-PCCP, Primary Containment Control Procedure
   AND perform concurrently with this procedure. .... ☐

5. Before suppression pool temperature reaches 110°F,
   THEN:

   a. **Insert** a manual scram. .... ☐

   b. **Enter** 1EOP-01-RSP(2EOP-01-RSP), Reactor Scram Procedure. .... ☐

56. 295028 1

Which one of the following completes the statement below?

The RTGB level indications that are least affected by elevated drywell temperatures during accident conditions are ___(1)___ , and the reason these instruments are least affected is because the ___(2)___.

A. N036/37, Fuel Zone
instruments are calibrated for accident conditions

B. N036/37, Fuel Zone
reference leg vertical drop in the drywell is only 24 inches

C. N026A/B, Wide Range
instruments are calibrated for accident conditions

D. N026A/B, Wide Range
reference leg vertical drop in the drywell is only 24 inches

Answer: D

K/A:
295028 High Drywell Temperature
EK1 Knowledge of the operational implications of the following concepts as they apply to HIGH DRYWELL TEMPERATURE: (CFR: 41.8 to 41.10)
01 Reactor water level measurement

RO/SRO Rating: 3.5/3.7

Pedigree: Bank

Objective: LOI-CLS-LP-01.2, Objective 5
Explain the effect that the following will have on reactor vessel level and /or pressure indications:
c. High containment (primary or secondary) temperatures

Reference: N/A

Cog Level: Fund

Explanation: During emergency conditions the Wide Range water level instruments may be used to determine reactor water level. However, Emergency Operating Procedures Caution 1 must be referenced to determine operability. The major portion of the reference leg is located in the reactor building and therefore, reactor building temperature is used in determining operability of the instruments. The REACTOR SATURATION LIMIT GRAPH is applicable to the Wide Range level instruments because of the small amount of reference leg located within the Drywell. The instruments may still be considered operable, even in the UNSAFE region, if level is greater than 20 inches.

During emergency conditions EOP Caution 1 must be used to determine level instrument operability. N036 and N037 level instruments may be used provided: The reference leg area drywell temperature is in the SAFE region of the REACTOR SATURATION LIMIT GRAPH

Distractor Analysis:

Choice A: Plausible because these instruments are often used during accident conditions, but the reference legs have long drywell runs which will make them unreliable at high drywell temperatures.

Choice B: Plausible because these instruments are often used during accident conditions, but the reference legs have long drywell runs which will make them unreliable at high drywell temperatures. Second part is correct for Wide Range instruments.

Choice C: Plausible because these instruments can be used under most conditions but the reason they can be reliable with high Drywell temperatures is due to their short vertical run in the Drywell.

Choice D: Correct Answer, see explanation.

SRO Basis: N/A

RPV Level Caution

Caution 1 (Continued)

Instrument	Conditions for Use
Wide Range Level Instruments B21-LI-R604A, B (N026A, B) C32-PR-R609 (N026B) Indicating Range 0-210 Inches Cold Reference Leg	• Temperature on the Reactor Building 50' below 140°F (B21-XY-5948A A2-4, B21-XY-5948B A2-4, ERFIS Computer Point B21TA102, **OR** B21TA103) **AND** • **IF** the reference leg area drywell temperature is in the UNSAFE region of Attachment 19, RPV Saturation Limit, **THEN** the indicated level is greater than 20 inches **OR** **IF** the reference leg area drywell temperature is in the SAFE region of Attachment 19, RPV Saturation Limit, **THEN** the indicated level is greater than 10 inches.

Level transmitters N026A and N026B, have reference leg arrangements that are different from the other level instruments. The instrument range is 0 to 210 inches with a reference leg of about 222 inches. However, only about a vertical drop of twenty-four (24) inches of the reference leg is exposed to the primary containment environment. The remainder of the reference leg is exposed to the secondary containment. This arrangement minimizes reactor water level indication error in LOCA conditions when primary containment temperatures are elevated. During high energy line break (HELB) conditions however, with secondary containment temperatures elevated above 140 degrees, these instruments will not be valid for RPV level indication.

The Fuel Zone water level transmitters are calibrated for:

- Reactor Pressure 0 psig
- Drywell Temperature 212 °F
- Reactor Building Temperature 140 °F
- No Jet Pump Flow

During emergency conditions EOP Caution 1 must be used to determine level instrument operability. N036 and N037 level instruments may be used provided:

- The reference leg area drywell temperature is in the SAFE region of the REACTOR SATURATION LIMIT GRAPH

AND

- The reference leg area drywell temperature is less than 440°, the indicated level is greater than minus 150 inches

OR

- If the Reference Leg Area Drywell Temperature is greater than 440 °F, the indicated level is greater than minus 130 inches

AND

- Both Reactor Recirculation Pumps are secured.

Caution 1 also contains three other graphs related to determining water level with the Fuel Zone instruments based on reference leg temperature during extremely degraded conditions. These graphs are based on three specific water levels used to assure adequate core cooling during transient conditions:

- Top of Active Fuel (LL3); used with RPV injection available above 120 psig
- Steam Cooling Water Level (LL4); with failure to scram (ATWS)
- Zero Injection RPV Water Level (LL5) with no RPV injection above 120 psig.

RPV Level Caution

Caution 1 (Continued)

Instrument	Conditions for Use
Fuel Zone Level Instruments B21-LI-R610 (N036) B21-LR-R615 (N037) Indicating Range -150 - +150 Inches Cold Reference Leg	• **IF** the reference leg area drywell temperature is less than 440°F, **THEN** the indicated level is greater than -150 inches **OR** **IF** the reference leg area drywell temperature is greater than or equal to 440°F, **THEN** the indicated level is greater than -130 inches. **AND** • Reactor Recirculation Pumps are shutdown. To determine RPV level at TAF, see **Unit 1 Only**: Attachment 23 **Unit 2 Only**: Attachment 24 To determine RPV level at the minimum steam cooling RPV level (LL-4), see **Unit 1 Only**: Attachment 25 **Unit 2 Only**: Attachment 26 To determine RPV level at the minimum zero injection level (LL-5), see **Unit 1 Only**: Attachment 27 **Unit 2 Only**: Attachment 28 To determine RPV level at 90 inches, see Attachment 29.

57. 295030 1

Following a DBA LOCA on Unit Two, plant conditions are as follows:

Reactor water level	55 inches and rising
Reactor pressure	150 psig
Torus temperature	220°F
Torus pressure	10.5 psig
Torus level	- 43 inches
2A Core Spray (CS) Pump flow	5,000 gpm
2A and 2C RHR Pump flow	5,000 gpm per pump (Torus Cooling mode)

Which one of the following identifies the ECCS pump(s), if any, that is (are) operating outside their associated NPSH limit?

(Reference provided)

A. Both RHR and CS are within NPSH limits.

B. Both RHR and CS are exceeding NPSH limits.

C. RHR is within NPSH limits. CS is exceeding NPSH limits.

D. RHR is exceeding NPSH limits. CS is within NPSH Limits.

Answer: C

K/A:
295030 Low Suppression Pool Water Level
EK1 Knowledge of the operational implications of the following concepts as they apply to LOW SUPPRESSION POOL WATER LEVEL: (CFR: 41.8 to 41.10)
02 Pump NPSH

RO/SRO Rating: 3.5/3.8

Pedigree: 2008 NRC Exam

Objective: CLS-LP-300B, Objective 17
Given plant condition and the NPSH and vortex limit graphs for the RHR and CS, determine if the NPSH and/or vortex limits have been exceeded for either of the two systems.

Reference: 0EOP-01-UG, Attachments 8 and 9. Core Spray & RHR NPSH Limits

Cog Level: Higher

Explanation: The student will need to plot each point on NPSH limit graph.
Torus pressure must be corrected down 0.5 psig to obtain the proper restriction line.
The correct torus pressure is 10.5 psig - 0.5 psig = 10 psig.
This correction must be performed for both the RHR and CS graphs.
See graphs in Notes Section.

Distractor Analysis:

Choice A: Plausible because if student fails to adjust torus pressure, they would both be within limits.

Choice B: Plausible because CS is outside limits, and if RHR is not plotted correctly, it could be outside limits.

Choice C: Correct answer, see explanation.

Choice D: Plausible because it requires correct plot to determine the correct answer including torus pressure adjustment.

SRO Basis: N/A

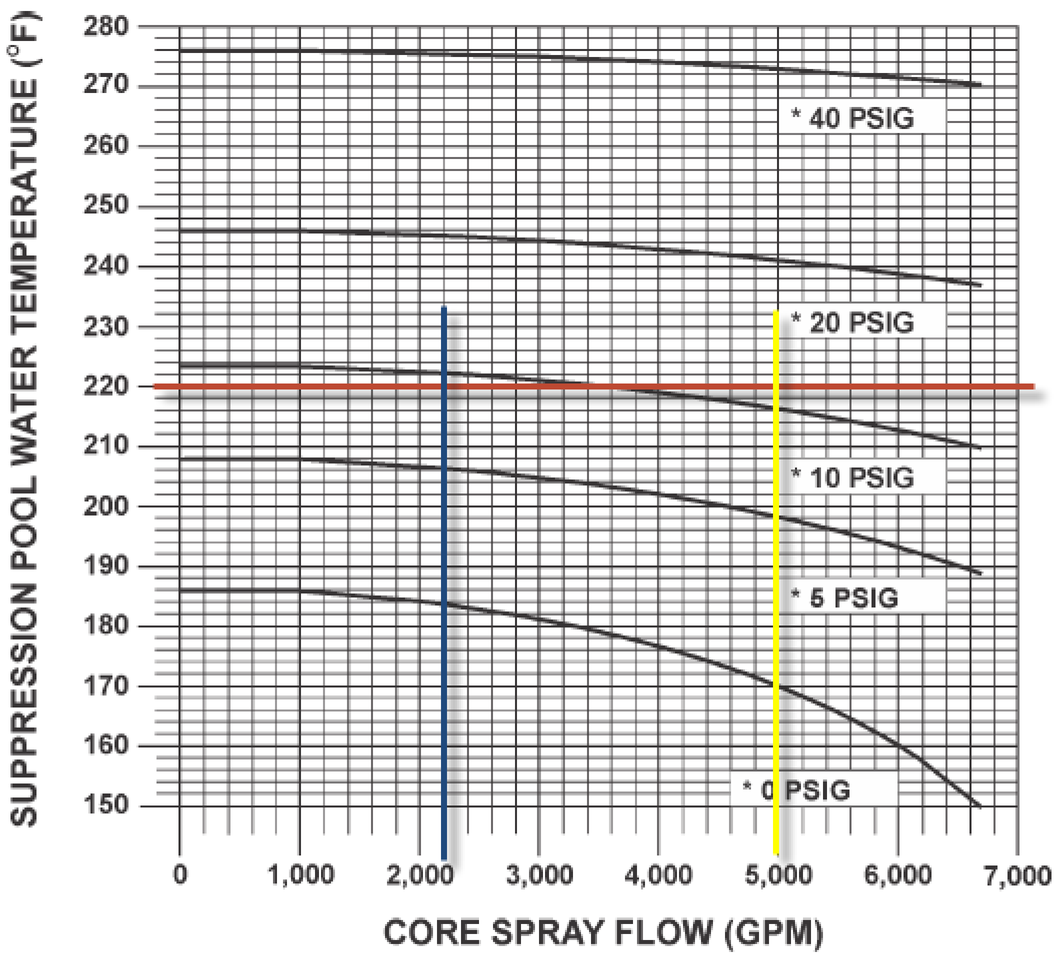
40. **NPSH Limit:** The highest torus temperature which provides adequate net positive suction head for an ECCS pump taking suction on the torus. This limit is utilized to preclude ECCS damage due to cavitation. (Attachment 8, Attachment 9 and Attachment 10)

For NPSH graphs, adequate net positive suction head is available when the flow rate and torus water temperature combination is below the adjusted torus pressure curve. The indicated torus pressure must be reduced by 0.5 psig for every foot of water level less than -2.6 feet to determine the correct torus pressure curve to be used in evaluating NPSH.

Loop A flow in yellow, Loop B flow in blue, Temp in red, safe below the 10 psig line. **Disregard CS Loop B flow due to question revision.**

FIGURE 5

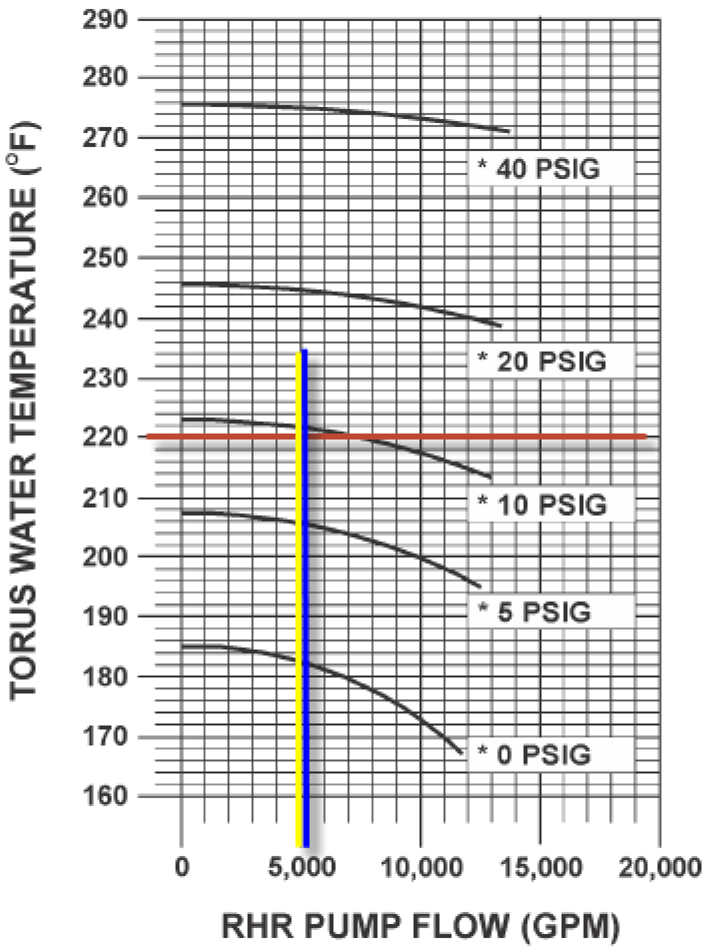
Core Spray NPSH Limit



SUBTRACT 0.5 PSIG FROM INDICATED SUPPRESSION CHAMBER PRESSURE FOR EACH FOOT OF WATER LEVEL BELOW A SUPPRESSION POOL WATER LEVEL OF -31 INCHES (-2.6 FEET).

*SUPPRESSION CHAMBER PRESSURE (CAC-PI-1257-2A OR CAC-PI-1257-2B)

RHR NPSH Limit



Subtract 0.5 psig from indicated torus pressure for each foot of water level below a torus water level of -31 inches (-2.6 feet).

58. 295031 1

A line break occurs in the Unit One Drywell with the following plant conditions:

RPV water level	180 inches steady on N026A/B, Wide Range Level
RPV water level	155 inches steady on N004A/B/C, Narrow Range Level
RPV water level	190 inches steady on N027A/B, Shutdown Range Level
RPV pressure	50 psig
Drywell ref leg temp	340°F
Drywell average temp	255°F
Reactor Building temp	128°F

Which one of the following provides reliable RPV water level indication?

(Reference provided)

A. N026A/B **ONLY**

B. N026A/B and N027A/B **ONLY**

C. N026A/B and N004A/B/C **ONLY**

D. N026A/B and N027A/B and N004A/B/C

Answer: C

K/A:
295031 Reactor Low Water Level
G2.2.37 Ability to determine operability and/or availability of safety related equipment.
(CFR: 41.7 / 43.5 / 45.12)

RO/SRO Rating: 3.6/4.6

Pedigree: 2001 NRC Exam

Objective: LOI-CLS-LP-300-B, Objective 16
Given Plant conditions, determine if the RPV water level instrument is providing valid trending information IAW Caution 1.

Reference: Caution 1 (EOP-01-UG, Attachment 6)

Cog Level: High

Explanation: Using Caution 1, level instruments are in the Safe Region of Reactor Saturation Limit curve. N026A/B are in the Safe area, N027A/B are in the Unsafe Region of Figure 16, and N004A/B/C are in the Safe Region of Figure 15.

Distractor Analysis:

Choice A: Plausible because N026A/B can be used, but are not the only instruments available.

Choice B: Plausible because N026s are available, but not the N027s. The examinee would have to plot the conditions correctly to determine this.

Choice C: Correct Answer, see explanation.

Choice D: Plausible because N026s and N027s are available, but the N027s are not.

SRO Basis: N/A

ATTACHMENT 6
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FIGURE 14
Reactor Saturation Limit

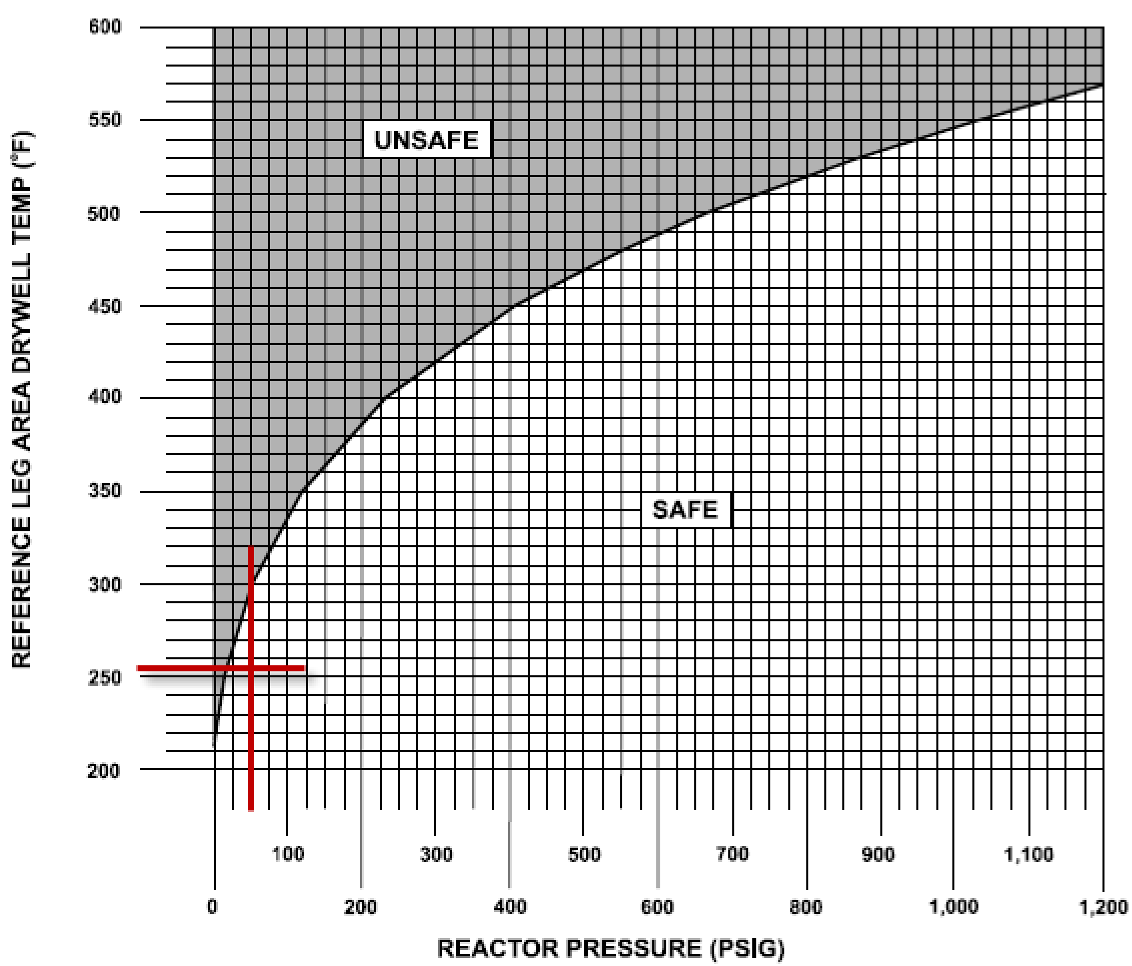

TABLE 1 (Cont'd)
Reactor Water Level Caution
(Caution 1)

Instrument	Conditions for Use
Wide Range Level Instruments B21-LI-R604A, B (N026A, B) C32-PR-R609 (N026B) Indicating Range 0-210 Inches Cold Reference Leg	1. Temperature on the Reactor Building 50' below 140°F (B21-XY-5948A A2-4, B21-XY-5948B A2-4, ERFIS Computer Point B21TA102, **OR** B21TA103) **AND** 2. **IF** the reference leg area drywell temperature is in the UNSAFE region of the Reactor Saturation Limit (Figure 14), **THEN** the indicated level is greater than 20 inches **OR** **IF** the reference leg area drywell temperature is in the SAFE region of the Reactor Saturation Limit (Figure 14), **THEN** the indicated level is greater than 10 inches.

ATTACHMENT 6
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FIGURE 16
Shutdown Range Level
Instrument (N027A, B) Caution

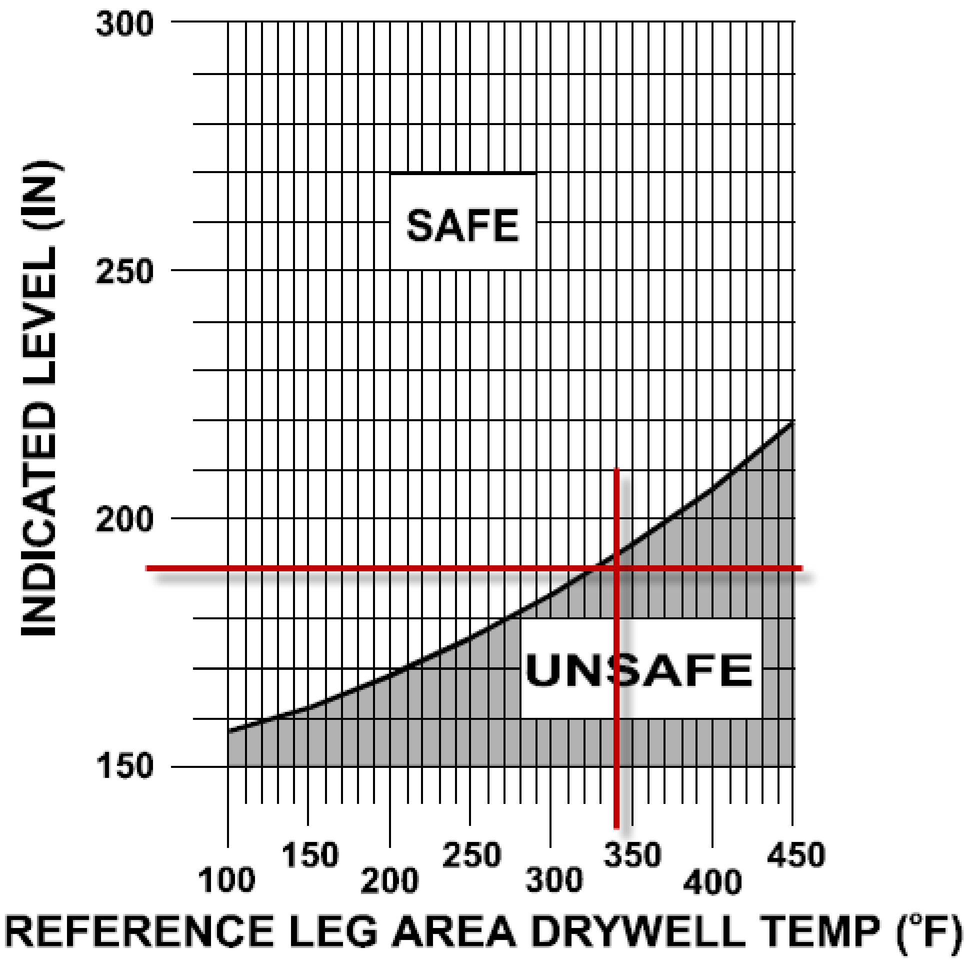
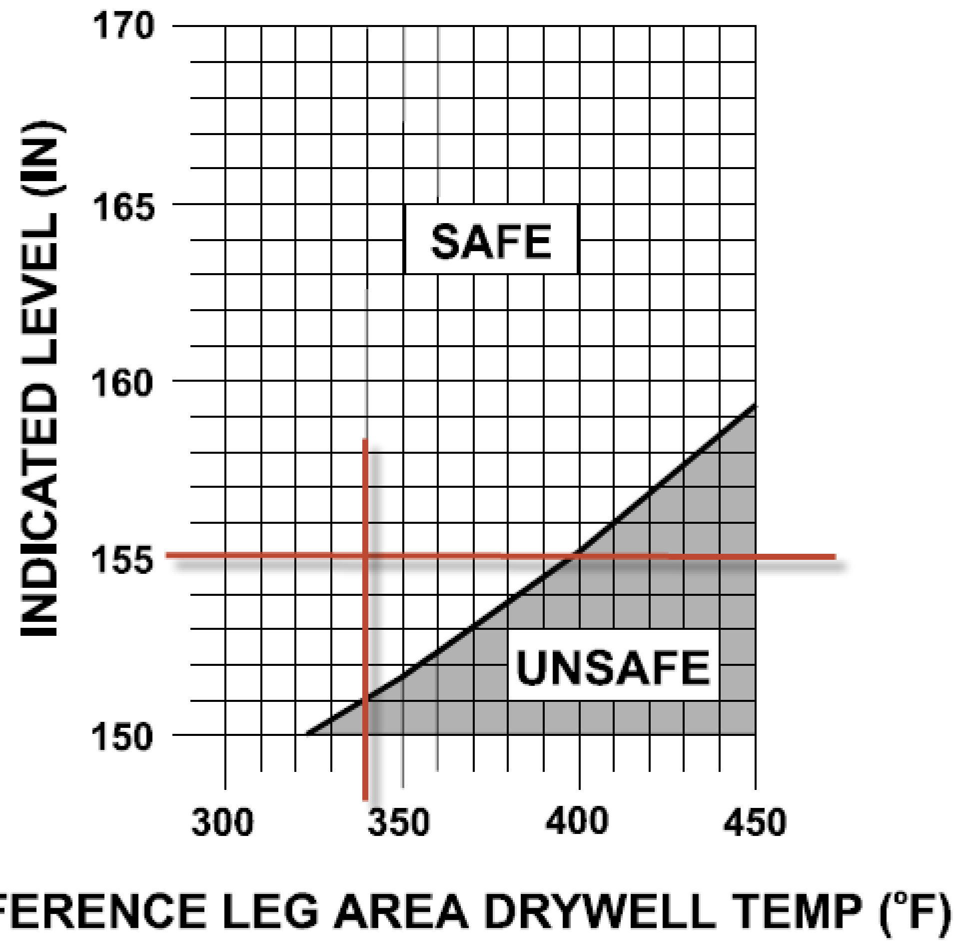


FIGURE 15

Unit 1 Narrow Range Level
Instrument (N004A, B, C) Caution



59. 295037 1

A reactor scram signal due to a loss of Division I 250 VDC Switchboard 2A, results in the following indications on Unit Two:

APRM readings:	16%
Control rods:	118 not full in
Blue scram lights:	137 illuminated

Which one of the following identifies the method capable to insert control rods IAW Scram Immediate Actions or LEP-02, Alternate Control Rod Insertion?

A. Scram Individual Control Rods

B. Reactor Manual Control System (RMCS)

C. Initiate Alternate Control Rod Insertion (ARI)

D. De-energize Scram Solenoids and Vent Scram Air Header

Answer: B

K/A:
295037 SCRAM Condition Present and Reactor Power Above APRM Downscale or Unknown
EK3 Knowledge of the reasons for the following responses as they apply to SCRAM CONDITION PRESENT AND REACTOR POWER ABOVE APRM DOWNSCALE OR UNKNOWN: (CFR: 41.5 / 45.6)
07 Various alternate methods of control rod insertion: Plant-Specific

RO/SRO Rating: 4.2/4.3

Pedigree: 2001 NRC Exam

Objective: LOI-CLS-LP-300-J, Objective 5
Given plant conditions, determine which sections of the Alternate Control Rod Insertion Procedure should be utilized for Control Rod Insertion in accordance with EOP-01-LEP-02.

Reference: None

Cog Level: High

Explanation: Venting the scram air header and de-energizing the Scram Pilot Valve solenoids would have the same effect as a full scram. Since all blue scram lights are lit, this would not accomplish control rod insertion. ARI is powered by DC electrical, so this would not work. RMCS has power and would be the appropriate means of inserting control rods.

Distractor Analysis:

Choice A: Plausible because this method would normally be used to insert control rods. Candidate must recognize that with all scram valves open, as indicated by blue lights, this method would not accomplish rod insertion.

Choice B: Correct Answer, see explanation

Choice C: Plausible because ARI is an alternate means of inserting control rods. In this case with a loss of DC power, ARI would not be available.

Choice D: Plausible because this method would normally work to insert control rods, but with scram valves open as indicated by blue lights, in this case it would not accomplish rod insertion..

SRO Basis: N/A

On a valid ARI initiation, all (8) of the Solenoid Valves will energize to vent off the air from the Scram air header.

Power is supplied to the ARI Solenoid Valves by 125 VDC Panel 11(12)A.

Normally 90% Control Rod travel occurs by approximately 3.5 seconds typically: 5.0 seconds maximum (with Scram Pilot Valves and backup). With the ARI valves only ~ 90% Control Rod travel, 15 seconds.

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2.3 Scram Pilot Solenoid Valves (SV-117 and SV-118) (Figures 03-4, 03-5, and 03-6)

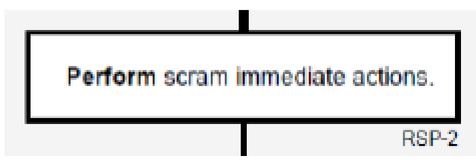
The Scram Pilot Solenoid Valves Direct air pressure to the Scram Valves, holding the Scram valve closed during Reactor Operation. With an RPS full Scram signal, the Solenoid Valves deenergize and reposition to rapidly bleed air pressure from the Scram Valves causing Control Rod insertion.

There is a single Scram Pilot Solenoid Valve for each pair of Scram Valves for a total of 137 pilot valves. Each Scram Pilot Solenioid Valve is provided with two redundant solenoids. Both solenoids must deenergize to vent air from the Scram Valves on a particular Hydraulic Control Unit. (Figure 03-6)

The pilot is a three-way 120 VAC solenoid operated valve. When both solenoids are deenergized the main valve will go to the vent position, bleeding air off the Scram Valves, allowing them to open.

The solenoids are divided into two sets. The set of SV-117 valves is energized by RPS Bus A and the set of SV-118 valves is energized by RPS Bus B. Each set is divided into four groups of valves (Group I, II, III, IV) to minimize current requirements through contacts and relays, hence increasing component life. Each group, contains about 1/4 of the HCUs.

5.2 <u>Step RSP-2</u>



Step RSP-2 includes the potential for multiple sensor and sensor relay failures in the automatic RPS logic where an automatic reactor scram should have initiated but did not. If needed a manual scram is inserted to accomplish an automatic action which should have taken place. A manual reactor scram is also required when directed from other EOPs and no condition exists which would have automatically initiated a reactor scram (e.g., entry from PCCP because of high torus temperature).

Step RSP-2 also addresses other Reactor Operator scram immediate actions and includes:

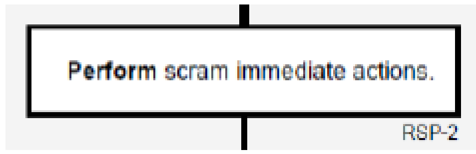
- ARI initiation is an additional means of inserting control rods if needed.

2.1.3 Operator Actions (continued)

8. **Insert** control rods by one or more methods:

- Section 2.3, Reset RPS and Initiate a Manual Scram on Page 15. .... ☐ RO
- Section 2.4, Reactor Manual Control System (RMCS) on Page 18. .... ☐ RO
- Section 2.5, Increasing Cooling Water Header Pressure on Page 20. .... ☐ RO
- Section 2.6, Scram Individual Control Rods on Page 22. .... ☐ RO
- Section 2.7, De-energize Scram Solenoids and Vent Scram Air Header on Page 26. .... ☐ RO
- Section 2.8, Venting Over Piston Area on Page 32 .... ☐ RO

5.2 Step RSP-2



Step RSP-2 includes the potential for multiple sensor and sensor relay failures in the automatic RPS logic where an automatic reactor scram should have initiated but did not. If needed a manual scram is inserted to accomplish an automatic action which should have taken place. A manual reactor scram is also required when directed from other EOPs and no condition exists which would have automatically initiated a reactor scram (e.g., entry from PCCP because of high torus temperature).

Step RSP-2 also addresses other Reactor Operator scram immediate actions and includes:

- ARI initiation is an additional means of inserting control rods if needed.

60. 295038 1

Primary Containment Venting is in progress on Unit Two. The Main Stack flow instrument loop (2-VA-FT-3359) is not operational. The following conditions exist:

Main Stack Rad Recorder (2-D12-RR-4599):	3.8 E-2 μCi/cc
Total Unit 1 flow to Main Stack:	4450 cfm
Total Unit 2 flow to Main Stack	21400 cfm
Common systems discharging to Main Stack	AOG Bldg Exhaust RW Bldg Fan A

Which one of the following is the Source Term release rate estimation from the Main Stack IAW 0PEP-03.6.1?

(Reference attached)

A. 3.8 E-2 μCi/sec

B. 3.8 E+5 μCi/sec

C. 1.2 E+6 μCi/sec

D. 1.5 E+6 μCi/sec

Answer: C

K/A:

295038 High Off-Site Release Rate

EA2 Ability to determine and/or interpret the following as they apply to HIGH OFF-SITE RELEASE RATE: (CFR: 41.10 / 43.5 / 45.13)

02 Total number of curies released

RO/SRO Rating: 2.5/3.3

Pedigree: New

Objective: LOI-CLS-LP-301-A, Objective 6
Determine data required for offsite dose projection in accordance with AD-EP-ALL-0202, Emergency Response Offsite Dose Assessment, and PEP-03.6.1, Release Estimates Based Upon Stack/Vent Readings.

Reference: None

Cog Level: High

Explanation: Total flow is given for Units 1 and 2. AOG Building and Radwaste Ventilation term must be determined using Attachment 6 of 0PEP-3.6.1. Total flow is 67,050 cfm. Calculation on Attachment 1 of 0PEP-3.6.1 results in a release of 1.2 E+6 µCi/sec.

Distractor Analysis:

Choice A: Plausible because it takes the release rate and uses it as the total release without conversion.

Choice B: Plausible because it considers only the release from Unit 2, since this is Unit 2 accident.

Choice C: Correct Answer, see explanation

Choice D: Plausible because it doubles the AOG Bldg. Exhaust since there are 2 supply and exhaust fans in AOG. Note on Attachment 6 must be used to determine proper configuration.

SRO Basis: N/A

ATTACHMENT 1
Page 1 of 1

Source Term Calculation From Plant Stack Monitors

Release rate is read in µCi/sec directly from 2-D12-RR-4599-4 (effluent channel) when the 2-VA-FT-3359 flow instrument loop is operational. The following calculations are necessary when this loop is not operational.

TIME	MONITOR[1] READING (µCi/cc)	FLOW[2] (cfm)	CONVERSION FACTOR cc/sec / cfm	RELEASE RATE[3] (µCi/sec)
	3.8E-2	4450 21400 18000 23200 67050	472	1.2E6

1 The monitor automatically selects the most accurate operational channel, either low, mid, or high range. Read the µCi/cc from the appropriate channel (low, mid, or high) of 2-D12-RR-4599.

2 If not available, use the sum of design flows for systems exhausting to the stack per Attachment 6.

3 Release rate (µCi/sec) = (µCi/cc) x (cfm) x (472)

Section Break (Next Page)

61. 300000 1
Which one of the following is the power supply to Air Compressor 1D?

A. 480 V Substation 1E

B. 480 V Substation 1F

C. 4160 V Bus 1C

D. 4160 V Bus 1D

Answer: C
K/A:
300000 Instrument Air System
K2 Knowledge of electrical power supplies to the following: (CFR: 41.7)
01 Instrument air compressor

RO/SRO Rating: 2.8/2.8

Pedigree: Bank

Objective: LOI-CLS-LP-046-A, Objective 2
State the power supplies for the following Pneumatic System components:
b. Air Compressor D

Reference: None

Cog Level: Fund.

Explanation: Air compressor 1D (unlike most BOP loads) is not equipped with either LOCA or Unit Trip Load Shed. Power is from BOP Bus 1C.

Distractor Analysis:

Choice A: Plausible because Compressor 1B is powered from 480 V Substation 1E

Choice B: Plausible because Substation 1F is powered from 4160 VAC Bus 1C

Choice C: Correct Answer, see explanation

Choice D: Plausible because 1D is the other 4160 VAC Non-ESF Bus.

SRO Basis: N/A

TABLE 46-2
Pneumatic Systems Power Supplies

Compressor 1B	480V Sub 1E
Compressor 1D	4160 V - 1C
Compressor 1D Control Cabinet Vent Fan	MCC-1TE
Compressor 2B	480V Sub 2E
Compressor 2D	4160 V - 2C
Compressor 2D Control Cabinet Vent Fan	MCC-2TD

BUS 1C

480 VAC Substation 1F
Circulating Water Pump 1A
Circulating Water Pump 1C
Condensate Booster Pump 1A
Condensate Booster Pump 1C
Condensate Pump 1B
Heater Drain Pump 1B
Unit 1 Refrigeration Machine 1A-RM-TB
Service Air Compressor 1D
Emergency Bus E2 Feeder

BUS 1D

480 VAC Substation 1E
Circulating Water Pump 1B
Circulating Water Pump 1D
Condensate Booster Pump 1B
Condensate Pump 1A
Condensate Pump 1C
Heater Drain Pump 1A
Heater Drain Pump 1C
Emergency Bus E1 Feeder

62. 400000 1

Which one of the following completes the statements below?

The highest CSW system pressure that will auto start the standby CSW pump is ___(1)___.

If pressure remains below this setpoint for at least ___(2)___ the SW-V3(V4), SW TO TBCCW HXS OTBD(INBD) ISOL, will reposition to their throttled positions.

A. (1) 40 psig
   (2) 30 seconds

B. (1) 40 psig
   (2) 70 seconds

C. (1) 65 psig
   (2) 30 seconds

D. (1) 65 psig
   (2) 70 seconds

Answer: B

K/A:
400000 Component Cooling Water System (CCWS)
K4 Knowledge of CCWS design feature(s) and or interlocks which provide for the following: (CFR: 41.7)
01 Automatic start of standby pump

RO/SRO Rating: 3.4/3.9

Pedigree: 2010-1 NRC Exam

Objective: LOI-CLS-LP-043, Objective 6d
Given plant conditions, predict whether any of the following pumps should start:
d. Conventional Service Water Pumps

Reference: None

Cog Level: Fund.

Explanation: The CSW pumps will auto start at 40 psig, the RCC pumps start at 65 psig. The SW-V3/4 throttle to a mid position if the low pressure exists for 70 seconds. The DG cooling valves swap to the opposite unit after low pressure for 30 seconds.

Distractor Analysis:

Choice A: Plausible because Part 1 is correct, the DG cooling valves swap to the opposite unit after low pressure for 30 seconds.

Choice B: Correct Answer, see explanation.

Choice C: Plausible because the RCC pumps auto start at 65 psig and the DG cooling valves swap to the opposite unit after low pressure for 30 seconds.

Choice D: Plausible because the RCC pumps auto start at 65 psig. Part 2 is correct.

SRO Basis: N/A

2.0 AUTOMATIC ACTIONS

2.1 Standby pump selected to the conventional service water header starts at 40 psig.

2.3 **IF** conventional service water header pressure remains below 40 psig for 70 seconds, **THEN**:

- *SW TO TBCCW HXS OTBD ISOL, SW-V3* closes to a throttled position
- *SW TO TBCCW HXS INBD ISOL, SW-V4* closes to a throttled position

0AOP-19.0	Rev. 18	Page 2 of 7

From AOP-16:

2.1 **IF** system pressure decreases to 65 psig, **THEN** the standby RBCCW pump will start.

From the SD-43:

3. Diesel Generator Cooling Water Supply Valves

Downstream of the Diesel Generator cooling water header valve, each diesel generator has two supply Valves 1(2)-SW-V679 for Diesel Generator 1, 1(2)-SW-V680 for Diesel Generator 2, 1(2)-SW-V681 for Diesel Generator 3, and 1(2)-SW-V682 for Diesel Generator 4. One supply valve is designated as the normal supply valve and will open when the diesel generator start is initiated and the diesel speed reaches 500 rpm. The other valve is the alternate supply valve. If sufficient pressure of 5.6 psig is not reached in ≃ 30 seconds, the alternate supply valve will open. Once the alternate supply valve is full open, the normal supply valve will close. This transfer sequence is initiated anytime service water pressure is lost when a diesel generator is operating. When the engine is shutdown and speed drops below 500 rpm, the open valve will automatically close.

Initial service water cooling to the diesel generators (i.e., 10 minutes)

63. 500000 1

Which one of the following completes the statements below concerning a trip of RPS MG Set A on Unit One?

CAC-AT-4410, Division II Hydrogen/Oxygen Monitor, ___(1)___ isolate.

CAC-AT-4409, Division I Hydrogen/Oxygen Monitor, ___(2)___ be unisolated using the CAM overrides.

A. (1) will
   (2) can

B. (1) will
   (2) can NOT

C. (1) will NOT
   (2) can

D. (1) will NOT
   (2) can NOT

Answer: A

K/A:
500000 High Containment Hydrogen Concentration
EA1 Ability to operate and monitor the following as they apply to HIGH CONTAINMENT HYDROGEN CONTROL: (CFR: 41.7 / 45.6)
02 Primary containment oxygen instrumentation

RO/SRO Rating: 3.3/3.2

Pedigree: Bank

Objective: LOI-CLS-LP-012, Objective 10
Describe how each group isolation signal that can be overridden is manually overridden.

Reference: None

Cog Level: Fund.

Explanation: Group 6 logic power comes from RPA A and RPS B. Loss of RPS A will close the Inboard Isolation valves. Power to the valves is not from RPS. Override switches allow overriding the isolation signal and opening the CAM valves to place the 4409 and 4410 in service.

Distractor Analysis:

Choice A: Correct Answer, see explanation.

Choice B: Plausible because both will be isolated due to Div. I logic and the isolation of the inboard isolations valve. If the valve were powered from Div. I RPS, then this would be a logical conclusion.

Choice C: Plausible because with loss of RPS A, 4409 is Div. I and the isolation logic is from Div. I. The 4409 could be unisolated, but not only 4409.

Choice D: Plausible because with loss of RPS A, 4409 is Div. I and the isolation logic is from Div. I. Since the isolation came from a Div. I signal, it might be concluded that Div. I is isolation valve power and cannot be unisolated.

SRO Basis: N/A

SD-24:

The isolation valves associated with the H_2/O_2 analyzers are arranged such that all the valves associated with AT-4409 are Div I and may be overridden with CS-2986 while all AT-4410 valves are Div II and may be overridden with CS-3452.

Control switches CS-2986 and CS-3452 are two position (NORMAL-ON) switches located on Control Room panel XU-51. Each switch has a Red and Green status light associated with it. The Green status light is illuminated when there are no isolation signals in effect. The Red status light is illuminated when the associated override switch is in the "ON" position bypassing the isolation signals to the associated valves.

Override and reset of the isolation signal for the H_2/O_2 analyzer valves requires positioning the associated override switch to "ON", reopening the valves by positioning the control switches to "Close" and back to "Open", selecting a sample point, and depressing the analyzer "Sample Start" pushbutton. The sample point selector switch and "Sample Start" pushbutton are both located on XU-51.

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In order to allow for post accident operation of the CAM system for sampling oxygen, hydrogen concentrations and containment radiation levels, an override function (Figure 12-32) is provided by control switches CAC-CS-2986 (Div. I) and CAC-CS-3452 (Div. II) located on the XU-51 panel. The Div I Hydrogen/Oxygen monitor CAC-AT-4409 can be placed in service using the Div I override switch, and the Div II monitor CAC-AT-4410 using the Div. II switch.

For analyzers AT-1260, AT-1261, and AT-1262, containment radiation monitors, each path has two valves in series, one a Div. I valve and the other a Div. II. One override switch is used to bypass the Division I isolation signal and the other override switch is used to bypass the Division II isolation signal. Once placed to ON, these switches provide a "hard" override, that is the valves remain open regardless of any subsequent isolation signal.

Each of the CAM override switches has a RED and GREEN indicating light. The GREEN light is normally on, and goes out when an isolation signal is present. The RED light will come on when the override switch is placed to ON. The override logic power comes from the same circuit as the isolation logic;
Div I 120 Vac Panel 31A (2A), and Div II 120 Vac Panel 31B (2D).

4.3.11 Reactor Protection System (RPS)

A single RPS MG A(B) set trip will result in the following:

- A half Group 1 A1/A2 (B1/B2) MSIV isolation occurs due to loss of power to isolation logic. The Inboard (Outboard) steam line drain and reactor sample valve will close.
- The respective inboard (outboard) isolation valves close for Groups 2, 3, 6 and 8. The Inboard (Outboard) CAC could be overridden open. The CAM and PASS Division I and II isolation valves will close, but could be overridden open.
- Both trains of SBGT will start, reactor building ventilation will isolate, and CREV will auto-start on loss of power to the PCIS logic.

64. 600000 1

Which one of the following completes the statements below?

SBGT utilizes ___(1)___ to detect a fire in the train.

If a fire is detected, water flow ___(2)___ automatically occur through the deluge valve.

A. (1) temperature switches
   (2) will

B. (1) temperature switches
   (2) will NOT

C. (1) ionization detectors
   (2) will

D. (1) ionization detectors
   (2) will NOT

Answer: B

K/A:
600000 Plant Fire On Site
AK2 Knowledge of the interrelations between PLANT FIRE ON SITE and the following:
01 Sensors / detectors and valves

RO/SRO Rating: 2.6/2.7

Pedigree: NRC Exam 10-1

Objective: LOI-CLS-LP-041, Objective 18
Given plant conditions, predict the response of the Fire Suppression and Fire Detection Systems.

Reference: None

Cog Level: Fund.

Explanation: There are two temperature switches to monitor the temperature of each Carbon Filter in each SBGT train.(TS 3/4) Switches VA-TS-5302-1 (VA-TS-5302-2), and VA-TS-5297-1 (VA-TS-5297-2) monitor Carbon Filter Bank No. 1 and actuate at 210°F, rising, to indicate a fire in the filter bank. Actuation of any switch will automatically open the Fire Suppression System's deluge valve for the associated carbon bank (note that the associated isolation valves must be opened manually for this system to inject)

Distractor Analysis:

Choice A: Plausible because temperature switches is correct. High temperature trips the train with inlet temp < 180°F but fire sprinkler flow will not occur until the local manual deluge valve is opened.

Choice B: Correct Answer, see explanation.

Choice C: Plausible because Ionization detectors detect the early products of combustion before they become visible smoke. High temperature trips the train with inlet temp < 180°F but fire sprinkler flow will not occur until the local manual deluge valve is opened.

Choice D: Plausible because Ionization detectors detect the early products of combustion before they become visible smoke and will not is correct due to local valve manipulations required to flow water.

SRO Basis: N/A

2.1.8 Deluge Valve System

Each SBGT System is equipped with a deluge system, including two deluge valves. The purpose of the deluge system is to extinguish a fire sensed in the carbon filter compartments. The deluge valves will open automatically, as sensed by rising temperature in the filters, or manually.

NOTE:	The deluge valves are manually isolated. In order for water to flow, the isolation valves for the deluge valve must be manually opened

2.2 Standby Gas Treatment System Flowpaths

2.2.1 Normal Flow Path

Figure 10-1 illustrates the arrangement of components and piping for the various flow paths of the SBGT System.

The normal system intake is from the 50' elevation of the Reactor Building through two motor operated intake isolation dampers (D, H) and into a common inlet duct. All areas of the Reactor Building communicate with this area. The common inlet duct splits and is routed to each Filter train through a motor operated, train inlet isolation damper (C, G).

Each Filter train component is duplicated in each train. Flow entering the Filter train first encounters the Moisture Separator then the electric Heater. Flow then passes through the Prefilter, HEPA Filter No. 1, Charcoal Filters Nos. 1 and 2, and HEPA Filter No. 2.

Flow exiting the filters passes through a duct to the Fan inlet. Flow from the Fan is routed through a check damper and motor operated discharge isolation damper (B, E). From the discharge isolation damper flow is routed to the Plant Stack.

A penetration of the common duct downstream of the fans permits sampling the gas stream with the Post Accident Sampling System (PASS) prior to its entering the Plant Stack. The sample line is isolated by a solenoid operated valve.

2.2.2 Primary Containment Purge (Vent) Flow Path

The inlet to the SBGT Filter trains may be aligned to either the Primary Containment Drywell or the Suppression Chamber air space for purging operation.

3. Carbon Filter Banks

There are two temperature switches to monitor the temperature of each Carbon Filter in each SBGT train.

(TS 3/4)
Switches VA-TS-5302-1 (VA-TS-5302-2), and VA-TS-5297-1 (VA-TS-5297-2) monitor Carbon Filter Bank No. 1 and actuate at 210°F, rising, to indicate a fire in the filter bank. Actuation of <u>any</u> switch will automatically open the Fire Suppression System's deluge valve for the associated carbon bank (note that the associated isolation valves must be opened for this system to inject) and trip the associated Fan and Heater unless compartment inlet temperature is > 180°F. Local and remote lights indicate switch actuation.

(TS 5/6)
Switches VA-TS-5303-1 (VA-TS-5303-2), and VA-TS-5298-1 (VA-TS-5298-2), monitor Carbon Filter Bank No. 2 and actuate at 210°F, rising, to indicate a fire in the #2 filter bank. Actuation of <u>any</u> switch will automatically open the Fire Suppression System's deluge valve for the associated carbon bank (note that the associated isolation valves must be opened for this system to inject) and trip the associated Fan and Heater unless compartment inlet temperature is > 180°F. Local and remote lights indicate switch actuation.

4. HEPA Filter No. 2 Compartment

Switches TSL-3456 (3455) provide annunciation of SBGT Filter train A/B Hi humidity.

3.2.6 Automatic

1. Upon receipt of an automatic initiation signal both trains of SBGT will start.

Unit 1 ONLY

The dampers associated with Unit 1 SBGT System will receive automatic open signals when an initiation signal is received <u>EXCEPT</u> for the train inlet and outlet dampers, (BFVs-1B,1C,1E,and 1G). Should these normally open dampers be manually closed locally via their CLOSE/OPEN pushbuttons, **they will NOT automatically reopen and the associated SBGT will not automatically start**.

65. 700000 1

Due to grid instability, UA-06 (1-2), *Gen Bus Under Freq Relay* is in alarm.

Which one of the following completes the statements below IAW 0AOP-22.0, Grid Instability?

The operator is directed to raise ___(1)___.

Off-frequency operation can stimulate resonance vibration in the ___(2)___.

A. (1) unit generation
   (2) generator stator

B. (1) unit generation
   (2) low pressure turbine blades

C. (1) generator voltage
   (2) generator stator

D. (1) generator voltage
   (2) low pressure turbine blades

Answer: B

K/A:
700000 Generator Voltage and Electric Grid Disturbances
AA1 Ability to operate and/or monitor the following as they apply to GENERATOR VOLTAGE AND ELECTRIC GRID DISTURBANCES: (CFR: 41.5 and 41.10 / 45.5, 45.7, and 45.8)
02 Turbine/generator controls

RO/SRO Rating: 3.8/3.7

Pedigree: New

Objective: LOI-CLS-LP-027, Objective 16b
Given plant conditions, predict the changes in Main Generator parameters associated with operation the following equipment:(LOCT)
a. Main Generator Manual Voltage Regulator
b. Main Generator Automatic Voltage Regulator

Reference: None

Cog Level: Fund.

Explanation: The Generator Bus Under frequency alarm comes in at 59.8 Hz. In order to raise frequency, 0AOP-22.0 directs the operator to raise unit generation. A caution in 0AOP-22.0 identifies the low pressure turbine blades as the concern for off-frequency operation.

Distractor Analysis:

Choice A: Plausible because part 1 is correct, with generator under frequency, generator damage is plausible.

Choice B: Correct Answer, see explanation.

Choice C: Plausible because generator voltage adjust rheostat is often used by the operator to make adjustment on the generator, and with generator under frequency, generator damage is plausible. In addition, APP UA-06 (1-2) identifies system voltage decreasing as an observation.

Choice D: Plausible because generator voltage adjust rheostat is often used by the operator to make adjustment on the generator, and part 2 is correct. In addition, APP UA-06 (1-2) identifies system voltage decreasing as an observation.

SRO Basis: N/A

0AOP-22.0:

9. IF system frequency is low,
THEN:

   a. **Raise** unit generation to the maximum consistent with unit conditions in accordance with 0GP-04, Increasing Turbine Load to Rated Power, or 0GP-12, Power Changes.......................... ☐

   b. **Establish** communication with the Load Dispatcher........................ ☐

   c. **Continue** maximum unit generation as directed by the Unit CRS **AND** **coordinate** with the Load Dispatcher. .... ☐

CAUTION

- Off-frequency operation can stimulate resonance vibration in low pressure blades. .... ☐

8.7 Generator Voltage Adjustments

C
Continuous
Use

8.7.1 Initial Conditions

1. The Generator and Exciter System is in operation in accordance with Section 5.1. ☐

2. The generator voltage requires adjustment to maintain 232 to 237.5 KV **OR** voltage is to be changed at the request of the System Load Dispatcher. ☐

8.7.2 Procedural Steps

NOTE:	Actions should be coordinated with the System Load Dispatcher to maintain the voltage and generator megavars within the established limits.

1. **IF** *REGULATOR MODE SELECTOR, 43CS*, is in *AUTO*, **THEN ADJUST** *GEN AUTO VOLT ADJ RHEO, 90CS*, to desired voltage. ☐

2. **IF** *REGULATOR MODE SELECTOR, 43CS*, is in *MAN*, **THEN ADJUST** *GEN MANUAL VOLT ADJ RHEO,* 70CS, to desired voltage. ☐

Unit 2
APP UA-06 1-2
Page 1 of 1

GEN UNDER FREQ RELAY

AUTO ACTIONS

1. Generator MWs increase to the limits of the pressure set.

CAUSE

1. Insufficient generation for system load.
2. Circuit malfunction.

OBSERVATIONS

1. Frequency decreasing (GEN-FM-736).
2. Increase in generator MW (GEN-MW-727).
3. System voltage decreasing (GEN-VM-732).

ACTIONS

NOTE:	A sudden increase in system frequency is possible if load shedding or other actions should result in turning a generation shortage into a generation excess.

1. Enter 0AOP-22.0, Grid Instability.
2. Increase turbine output to the maximum consistent with plant conditions per 0GP-04, Increasing Turbine Load to Rated Power or 0GP-12, Power Changes.
3. If the system frequency is less than 58.1 hertz, trip the turbine immediately.
4. If a circuit malfunction is suspected, ensure a WO is prepared.

66. CONDUCT OF OPERATION 1

Which one of the following identifies the location of the 2A Heater Drain Pump Unit Trip Load Shed Switch?

A. Control Room back panel area at the XU-23 panel.

B. Front of the breaker cubicles at the associated Emergency Bus.

C. Breezeway adjacent to the Heater Drain Deaerator level controller.

D. Front of the breaker cubicle at the associated Balance of Plant (BOP) Bus.

Answer: D

K/A:
G2.1.30 Ability to locate and operate components, including local controls. (CFR: 41.7 / 45.7)

RO/SRO Rating: 4.4/4.0

Pedigree: Bank

Objective: LOI-CLS-LP-300-J, Objective 6e
State the location of the following:
Heater Drain Pump Unit Trip Load Shed Enable/Disable switches

Reference: None

Cog Level: Fund.

Explanation:

Distractor Analysis:

Choice A: Plausible because the Master Load Shed switch is located in the back panels.

Choice B: Plausible because if the Heater Drain Pumps were powered from the Emergency switchgear, this is where they would be located.

Choice C: Plausible because Heater Drain Pumps are located near the breezeway and many EOP overrides are located in the breezeway.

Choice D: Correct Answer, see explanation

SRO Basis: N/A

Section 2 (Continued)

NOTE:	Heater drain deaerator level must be greater than or equal to 48 inches to start a heater drain pump.

AO: 7. **PLACE** the Unit Trip Load Shed Selector Switch for the heater drain pump to be started, in *DISABLED*. ☐

RO: 8. **START** the selected heater drain pump. ☐

RO: 9. **PLACE** *STARTUP LEVEL CONTROL VALVE, FW-LIC-3269*, in *M* (Manual) **AND OPEN**, as necessary. ☐

Initials

10. **WHEN** heater drain pump injection is no longer required, **THEN PERFORM** the following:

RO: a. **STOP** selected heater drain pump. ____

AO: b. **PLACE** Unit Trip Load Shed Selector Switch for the selected heater drain pump in *ENABLED*. ___/___ Ind.Ver.

Turbine Building - 4160V Bus 2C - El. 20 ft.			
AC3	Heater Drain Pump 2B	RACKED IN	
AC3	Heater Drain Pump 2B 650 Watt Motor Heater (RC)	ON	
AC3	Heater Drain Pump 2B Elapse Time Meter (RB)	ON	
AC3	LOCA Load Shed Selector Switch	DISABLED [Note 1]	/
AC3	Unit Trip Load Shed Selector Swith	ENABLED [Note 1]	/

Note 1 Independent Verification Required

67. CONDUCT OF OPERATION 2

Core reload is in progress during a refueling outage. The initial loading of fuel bundles around each SRM centered 4-bundle cell was completed with all four SRMs fully inserted and reading 50 cps.

It is now approximately half way through the core loading sequence and SRMs read 80 cps.

Which one of the following identifies the count rate when fuel movement must first be suspended following loading of additional multiple bundles IAW FH-11, Refueling?

A. 100 cps.

B. 200 cps.

C. 250 cps.

D. 400 cps.

Answer: C

K/A:
G2.1.42 Knowledge of new and spent fuel movement procedures. (CFR: 41.10 / 43.7 / 45.13)

RO/SRO Rating: 2.5/3.4

Pedigree: Bank

Objective: LOI-CLS-LP-305, Objectives 17, 18

17. Describe the SRM Monitoring requirements in FH-11, Refueling.
18. Given the conditions during a refueling outage state the operator actions required for rising SRM count rates and/or inadvertent criticality.

Reference: None

Cog Level: Fund

Explanation: See Notes Section. Suspension of fuel movement and notification of the Reactor Engineer is required if an SRM rise by a factor of 5 relative to the SRM baseline.

Distractor Analysis:

Choice A: Plausible because doubling with a single bundle would be reason to suspend fuel movement.

Choice B: Plausible because this is an increase by a factor of 4 from baseline.

Choice C: Correct Answer, see explanation.

Choice D: Plausible because this is an increase by a factor of 5 from the given 80 cps at the halfway point.

SRO Basis: N/A

FH-11:

24. Suspension of fuel movement and notification of the Reactor Engineer is required if either of the following occur:

    - An SRM reading rise by a factor of two upon insertion of any single bundle. During a spiral reload, this restriction applies only after the initial loading of fuel bundles around each SRM is complete. During a Core Shuffle, this restriction does **NOT** apply to the SRM that is having an adjacent fuel bundle inserted or removed. .... ☐

    - An SRM rise by a factor of five relative to the SRM baseline count rate recorded on Attachment 6, Documentation for SRM Baseline .... ☐

25. SRM count rate may drop to less than 3 cps during either of the following conditions:

    - With less than or equal to four fuel assemblies adjacent to the SRM and **NO** other fuel assemblies in the associated core quadrant .... ☐

    - During a core spiral offload .... ☐

68. EMERGENCY PROCEDURE 1

Which one of the following completes the statement below?

The 'Minimum Number of SRVs Required for Emergency Depressurization' is ___(1)___ , which corresponds to a Decay Heat Removal Pressure sufficiently low that the ECCS with the ___(2)___ head will be capable of making up the SRV steam flow at the corresponding Decay Heat Removal Pressure.

A. (1) five
(2) lowest

B. (1) five
(2) highest

C. (1) seven
(2) lowest

D. (1) seven
(2) highest

Answer: A

K/A:
G2.4.17 Knowledge of EOP terms and definitions. (CFR: 41.10 / 45.13)

RO/SRO Rating: 3.9/4.3

Pedigree: New

Objective: LOI-CLS-LP-300-P, Objective 3
Given plant conditions requiring Emergency Depressurization and the Emergency Operating Flowchart, determine the correct operator actions.

Reference: None

Cog Level: Fund.

Explanation: The Minimum Number of SRVs Required for Emergency Depressurization (5) is defined to be the least number of SRVs which correspond to a Decay Heat Removal Pressure sufficiently low that the ECCS with the lowest head will be capable of making up the SRV steam flow at the corresponding Decay Heat Removal Pressure.

Distractor Analysis:

Choice A: Correct Answer, see explanation.

Choice B: Plausible because part 1 is correct, part 2 would provide ECCS injection with the higher pressure systems.

Choice C: Plausible because the operator attempts to open 7 SRVs when entering EDP, part 2 is correct.

Choice D: Plausible because the operator attempts to open 7 SRVs when entering EDP, part 2 would provide ECCS injection with the higher pressure systems.

SRO Basis: N/A

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5.8.1 Step EDP-8 First Contingency

If one or more ADS valves cannot be opened, other SRVs are opened until the number of open SRVs equals seven (number of SRVs dedicated to ADS). If a non-ADS SRV is stuck open, seven ADS valves should be opened, resulting in a total of eight open SRVs. This provides the requisite depressurization rate without exceeding any design criteria.

The requirement for opening additional SRVs is based on the number of ADS valves that can be opened rather than the number that are open. The phrase "any ADS valve cannot be opened" is used to accommodate events in which SRVs must be reclosed to preserve adequate core cooling using steam driven injection systems and events in which the RPV is already depressurized when EDP is entered. If ADS valves have been closed to preserve adequate core cooling or are closed because RPV pressure is below the minimum SRV re-opening pressure, opening other SRVs is not appropriate.

As in the preferred strategy the use of non-ADS SRVs is allowed only if torus level is above -8 feet.

5.8.2 Step EDP-8 Second Contingency

If less than five SRVs can be opened and RPV pressure is more than 100 psig above torus pressure, Table P-2 systems are used to depressurize the RPV and keep it depressurized. The Minimum Number of SRVs Required for Emergency Depressurization (5) is defined to be the least number of SRVs which correspond to a Decay Heat Removal Pressure sufficiently low that the ECCS with the lowest head will be capable of making up the SRV steam flow at the corresponding Decay Heat Removal Pressure.

69. EMERGENCY PROCEDURE 2
Which one of the following completes the statements below?

The emergency response facility that has the primary function to perform in-plant repairs is the ___(1)___.

The primary location for this facility is the ___(2)___.

A. (1) Technical Support Center
(2) O&M Building

B. (1) Technical Support Center
(2) Simulator area

C. (1) Operational Support Center
(2) O&M Building

D. (1) Operational Support Center
(2) Simulator area

Answer: C

K/A:
G2.4.42 Knowledge of emergency response facilities. (CFR: 41.10 / 45.11)

RO/SRO Rating: 2.6/3.8

Pedigree: 2008 NRC Makeup Exam

Objective: LOI-CLS-LP-300-A, Objective12
Describe the following in accordance with PEP-02.6.12, Activation and Operation of the Operational Support Center (OSC):
a. Primary and alternate locations
b. Function

Reference: None

Cog Level: Fund.

Explanation: The OSC is located in the O&M Building as indicated in PEP-026.12, Attachment 1. The primary function of the OSC is to facilitate in-plant repair and assessment activities.

Distractor Analysis:

Choice A: Plausible because the TSC provides direction to the OSC and the location is correct.

Choice B: Plausible because the TSC provides direction to the OSC, and the simulator is the backup location for the OSC.

Choice C: Correct Answer, see explanation.

Choice D: Plausible because the OSC is correct and the simulator is the backup location.

SRO Basis: N/A

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3.2 The primary function of the OSC is to facilitate in-plant repair and assessment activities.

3.5 The OSC receives direction from the Technical Support Center (TSC) concerning repair activities and priorities.

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3.7 In the event that the OSC can no longer meet habitability requirements, the OSC can be relocated to the Simulator area of the EOF/TSC Operations Training Building. See Attachment 4, Alternate OSC Relocation Checklist, and Attachment 5, Alternate OSC

70. EQUIPMENT CONTROL 1

Which one of the following identifies the bases for the Minimum Critical Power Ratio (MCPR) Safety Limit IAW Technical Specifications Bases 2.1.1, Reactor Core Safety Limits?

The MCPR Safety Limit ensures that:

A. the calculated changes in core geometry shall be such that the core remains amenable to cooling.

B. plastic strain of the cladding does not exceed 1% during all modes of operation.

C. the calculated total oxidation shall nowhere exceed 0.17 times the total cladding thickness before oxidation.

D. during normal operation and during Anticipated Operational Occurrences, at least 99.9% of the fuel rods in the core do not experience transition boiling.

Answer: D

K/A:

G2.2.25 Knowledge of the bases in Technical Specifications for limiting conditions for operations and safety limits. (CFR: 41.5 / 41.7 / 43.2)

RO/SRO Rating: 3.2/4.2

Pedigree: 2010-1 NRC Exam

Objective: LOI-CLS-LP-200-B, Objective 3
State each TS Safety Limit and discuss the basis for each of the Safety Limits.

Reference: None

Cog Level: Fund.

Explanation: Requires knowledge of TS Safety Limit Bases and the ability to distinguish between Safety Limits and Operating Limits. The MCPR fuel cladding integrity SL ensures that during normal operation and during AOOs, at least 99.9% of the fuel rods in the core <u>do not</u> experience transition boiling.

Distractor Analysis:

Choice A: Plausible because since this is ECCS acceptance criteria, which can be sometimes confused with Safety Limit bases.

Choice B: Plausible because since this is the basis for the LHGR limit, which is one of the Safety Limits.

Choice C: Plausible because since this is ECCS acceptance criteria, which can be confused with Safety Limit bases.

Choice D: Correct Answer, see explanation.

SRO Basis: N/A

B 2.0 SAFETY LIMITS (SLS)

B 2.1.1 Reactor Core SLs

BASES

BACKGROUND	SLs ensure that specified acceptable fuel design limits are not exceeded during steady state operation, normal operational transients, and anticipated operational occurrences (AOOs). The fuel cladding integrity SL is set such that no fuel damage is calculated to occur if the limit is not violated. Because fuel damage is not directly observable, a stepback approach is used to establish an SL, such that the MCPR is not less than the limit specified in Specification 2.1.1.2. MCPR greater than the specified limit represents a conservative margin relative to the conditions required to maintain fuel cladding integrity. The fuel cladding is one of the physical barriers that separate the radioactive materials from the environs. The integrity of this cladding barrier is related to its relative freedom from perforations or cracking. Although some corrosion or use related cracking may occur during the life of the cladding, fission product migration from this source is incrementally cumulative and continuously measurable. Fuel cladding perforations, however, can result from thermal stresses, which occur from reactor operation significantly above design conditions. While fission product migration from cladding perforation is just as measurable as that from use related cracking, the thermally caused cladding perforations signal a threshold beyond which still greater thermal stresses may cause gross, rather than incremental, cladding deterioration. Therefore, the fuel cladding SL is defined with a margin to the conditions that would produce onset of transition boiling (i.e., MCPR = 1.00). These conditions represent a significant departure from the condition intended by design for planned operation. The MCPR fuel cladding integrity SL ensures that during normal operation and during AOOs, at least 99.9% of the fuel rods in the core do not experience transition boiling.

(continued)

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.3 LINEAR HEAT GENERATION RATE (LHGR)

BASES

BACKGROUND

The LHGR is a measure of the heat generation rate of a fuel rod in a fuel assembly at any axial location. Limits on the LHGR are specified to ensure that fuel design limits are not exceeded anywhere in the core during normal operation, including anticipated operational occurrences (AOOs). Exceeding the LHGR limit could potentially result in fuel damage and subsequent release of radioactive materials. Fuel design limits are specified to ensure that fuel system damage, fuel rod failure or inability to cool the fuel does not occur during the anticipated operating conditions identified in References 1 and 2.

For GNF fuel, LCO 3.2.1 "AVEARGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)" ensures that the fuel design limits are not exceeded during normal operation and anticipated operational occurrences.

APPLICABLE SAFETY ANALYSES

The analytical methods and assumptions used in evaluating the fuel system design are presented in References 1 and 2. The fuel assembly is designed to ensure (in conjunction with the core nuclear and thermal hydraulic design, plant equipment, instrumentation, and protection system) that the fuel damage will not result in the release of radioactive materials in excess of the guidelines of 10 CFR, Parts 20, 50, and 50.67. The mechanisms that could cause fuel damage during normal operations and operational transients and that are considered in fuel evaluations are:

a. Rupture of the fuel rod cladding caused by strain from the relative expansion of the UO_2 pellet; and

b. Severe overheating of the fuel rod cladding caused by inadequate cooling

A value of 1% plastic strain of the fuel cladding has been defined as the limit below which fuel damage caused by overstraining of the fuel cladding is not expected to occur (Ref. 3).

Fuel design evaluations have been performed and demonstrate that the 1% fuel cladding plastic strain design limit is not exceeded during continuous operation with LHGRs up to the operating limit specified in the COLR. The analysis also includes allowances for short term transient operation above the operating limit to account for AOOs.

(continued)

71. EQUIPMENT CONTROL 2
The quarterly performance of 0PT-10.1.1, RCIC System Operability Test, on Unit One has failed and the RCIC System is declared inoperable by the CRS.

Which one of the following identifies the required action statement and completion time IAW Technical Specifications, LCO 3.5.3, RCIC System?

Verify HPCI is operable by:

A. administrative means, immediately.

B. administrative means, within one hour.

C. performance of 0PT-9.2, HPCI System Operability Test, immediately.

D. performance of 0PT-9.2, HPCI System Operability Test, within one hour.

Answer: A

K/A:
G2.2.40 Ability to apply Technical Specifications for a system. (CFR: 41.10 / 43.2 / 43.5 / 45.3)

RO/SRO Rating: 3.4/4.7

Pedigree: Bank

Objective: LOI-CLS-LP-200-B, Objective 19
State the Technical Specification required actions with a completion time of less than or equal to one hour.

Reference: None

Cog Level: High

Explanation: With RCIC inoperable, HPCI must be verified operable by administrative means immediately.

Distractor Analysis:

Choice A: Correct Answer, see explanation

Choice B: Plausible because ROs must know TS actions within 1 hour.

Choice C: Plausible because ROs must know what actions are required to determine operability. If PT is current, then it does not have to be performed.

Choice D: Plausible because ROs must know TS actions within 1 hour and requirement to determine operability.

SRO Basis: N/A

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS) AND REACTOR CORE ISOLATION COOLING (RCIC) SYSTEM

3.5.3 RCIC System

LCO 3.5.3 The RCIC System shall be OPERABLE.

APPLICABILITY: MODE 1,

MODES 2 and 3 with reactor steam dome pressure > 150 psig.

ACTIONS

---NOTE---

LCO 3.0.4.b is not applicable to RCIC.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. RCIC System inoperable.	A.1 Verify by administrative means High Pressure Coolant Injection System is OPERABLE. <u>AND</u> A.2 Restore RCIC System to OPERABLE status.	Immediately 14 days

72. EQUIPMENT CONTROL 3

Which one of the following identifies the significance of the yellow dot affixed to an annunciator window?

The yellow dot means that the alarm:

A. is disabled.

B. is due to a clearance.

C. is an expected/nuisance alarm.

D. has one or more inputs to a multiple input annunciator disabled.

Answer: D

K/A:
G2.2.43 Knowledge of the process used to track inoperable alarms. (CFR: 41.10 / 43.5 / 45.13)

RO/SRO Rating: 3.0/3.3

Pedigree: New

Objective: LOI-CLS-LP-201-D, Objective 1m
Explain/describe the following IAW AD-OP-ALL-1000, Conduct of Operations, 0OI-01.01, BNP Conduct of Operations Supplement and OPS-NGGC-1314, Communications:
m. Annunciator response and status control requirements

Reference: None

Cog Level: Fund

Explanation: A yellow dot on an annunciator window IAW 0OI-01.01 indicates an annunciator has one or more inputs to a multiple input annunciator disabled.

Distractor Analysis:

Choice A: Plausible because this would be indicated by a blue dot.

Choice B: Plausible because this would be indicated by a red dot.

Choice C: Plausible because this would be indicated by a green dot.

Choice D: Correct Answer, see explanation

SRO Basis: N/A

5.14.2 Lit/Disabled Annunciators (continued)

NOTE

An annunciator card may be removed temporarily for troubleshooting purposes. Annunciators identified by a Technical Requirements Manual Specification or an Offsite Dose Calculation Manual Specification may be removed from operation for the performance of trouble shooting for up to 30 minutes, without entering the associated specification, provided the conditions identified in the applicable specification and bases for each annunciator are met. This provision does not apply for annunciators identified with a Technical Specification.

g. Attachment 24, Annunciator Removal From Service Form, shall be completed to permanently document disabled annunciators (in addition to the computer database update).

(1) The original shall be maintained in the Disabled Annunciators section of the associated binder while the annunciator is disabled.

(2) Once the annunciator is restored to service, the completed form is forwarded to Document Services by the Operations Nuclear Technical Assistant.

h. An annunciator may be disabled for a short duration (not to exceed shift turnover) without completing Attachment 24 if approved by the CRS. If approved, a label will be affixed to the RTGB indicating which annunciator is disabled. This exception to completing Attachment 24 is intended to be used to address nuisance alarms which only exist for a short duration due to changing plant conditions.

i. Lit/Degraded Annunciators that will remain past the end of shift will be flagged/coded as follows and tracked:

- Green - Expected/nuisance
- Blue - Disabled or removed from service
- Yellow - One or more inputs to a multiple input annunciator are disabled.
- Red - Due to a clearance

Detailed instructions for venting and purging containment are provided in supporting procedures. In summary:

- It is possible to vent the primary containment from the torus or the drywell.
- The torus vent path exhausts the airspace of the torus through the primary containment vent penetration located in the torus. The elevation of the torus vent is +6 feet. If torus water level is above the elevation of the torus vent penetration, the torus vent path is not used since the vent lines are not designed to accommodate the flow of water and the isolation valves and downstream components of the vent lines may be damaged.
- If the torus cannot be vented, the drywell is vented.
- Torus or drywell purge is appropriate only if the torus or drywell is being vented. Purging without an open vent path will result in repressurizing the drywell without lowering the partial pressure or mass of hydrogen or oxygen.

73. RADIATION CONTROL 1

Following a large line break in the drywell, H2/O2 monitors have been placed in service.
Plant conditions:

Drywell hydrogen	2.5% (ERFIS)
Drywell oxygen	3.5% (ERFIS)
Torus hydrogen	1.4% (ERFIS)
Torus oxygen	3.5% (ERFIS)
Torus level	-36 inches

Which one of the following completes the statements below?

The action directed by PCCP is to vent and purge Primary Containment ___(1)___.

Venting from the ___(2)___ is preferred.

A. (1) ONLY within ODCM release rate limits
(2) torus

B. (1) ONLY within ODCM release rate limits
(2) drywell

C. (1) irrespective of the off-site release rate
(2) torus

D. (1) irrespective of the off-site release rate
(2) drywell

Answer: A

K/A:
G2.3.11 Ability to control radiation releases. (CFR: 41.11 / 43.4 / 45.10)

RO/SRO Rating: 3.8/4.3

Pedigree: Bank

Objective: LOI-CLS-LP-300-L, Objective 11
Given Primary Containment Control Procedure, which steps have been completed and plant parameters, determine the required operator actions.

Reference: None

Cog Level: High

Explanation: With Torus level below +6 feet, the Torus vent path is preferred. Venting is only permitted under the stated conditions if ODCM limits will not be exceeded. This is also stated in an override.

Distractor Analysis:

Choice A: Correct Answer, see explanation

Choice B: Plausible because part 1 is correct. The drywell is 1 of 2 possibilities.

Choice C: Plausible because there are conditions when venting is required irrespective of off-site release rates. Part 2 is correct.

Choice D: Plausible because there are conditions when venting is required irrespective of off-site release rates. The drywell is 1 of 2 possibilities.

SRO Basis: N/A

PC/G

PC/G-1

IF	THEN
Containment H2/O2 monitoring system NOT available	**Monitor** H2/O2 values per E&RC-1505
Either: • ODCM offsite radioactivity release rate is reached • Containment H2 can be maintained below 1%	**Secure** containment vent and purge unless required by PC/P flowpath

PC/G-2

Place H2/O2 monitoring in service (OP-24).

PC/G-3

IF ERFIS NOT available,
THEN **compensate** H2/O2 values per EOP-01-UG, Attachment 35.

PC/G-4

NOTE

Applies to direct readings or sample results.

WHEN

Drywell OR torus H2 reaches 1%

THEN continue

PC/G-5

Notify E&RC to sample containment for venting.

PC/G-6

Trip Hydrogen Water Chemistry.

PC/G-7

WHEN

Determined ODCM will NOT be exceeded

THEN continue

PC/G-8

Vent containment per EOP-01-SEP-01.

PC/G-9

IF containment can be vented,

Detailed instructions for venting and purging containment are provided in supporting procedures. In summary:

- It is possible to vent the primary containment from the torus or the drywell.

- The torus vent path exhausts the airspace of the torus through the primary containment vent penetration located in the torus. The elevation of the torus vent is +6 feet. If torus water level is above the elevation of the torus vent penetration, the torus vent path is not used since the vent lines are not designed to accommodate the flow of water and the isolation valves and downstream components of the vent lines may be damaged.

- If the torus cannot be vented, the drywell is vented.

- Torus or drywell purge is appropriate only if the torus or drywell is being vented. Purging without an open vent path will result in repressurizing the drywell without lowering the partial pressure or mass of hydrogen or oxygen.

74. RADIATION CONTROL 2

Which one of the following identifies the purpose of the installed Drywell High Range Area Radiation Monitors?

These instruments are used to provide:

A. an entry condition into 0AOP-05.4, Radiological Release.

B. compliance with LCO 3.4.5, RCS Leakage Detection Instrumentation.

C. an entry condition into RRCP, Radioactivity Release Control Procedure.

D. estimates of the extent of severe core damage during accident conditions.

Answer: D

K/A:

G2.3.15 Knowledge of radiation monitoring systems, such as fixed radiation monitors and alarms, portable survey instruments, personnel monitoring equipment, etc. (CFR: 41.12 / 43.4 / 45.9)

RO/SRO Rating: 2.9/3.1

Pedigree: 2008 NRC Makeup Exam

Objective: LOI-CLS-LP-011.1, Objective 1
State the purpose of the Area Radiation Monitoring System

Reference: None

Cog Level: Fund

Explanation: Drywell high range area monitors provide indications of gross fuel failure and are used to determine emergency plan emergency action level associated with abnormal core conditions. The readings on these monitors can also used by the emergency response organization to estimate the extent of core damage. The first indication of minor fuel failure is normally provided by SJAE radiation monitors. AOP-05.0 lists area radiation monitors as an entry but not drywell high range area monitoring. EOP-04 lists effluent monitors as entry, drywell high range monitors do not monitor effluent.

Distractor Analysis:

Choice A: Plausible because since several radiation monitors are listed as entry conditions to AOP-05.4.

Choice B: Plausible because since these radiation monitors are located in the drywell and their readings could increase if leakage into the drywell increased.

Choice C: Plausible because since several radiation monitors are listed as entry conditions to EOP-04-RRCP.

Choice D: Correct Answer, see explanation.

SRO Basis: N/A

2.5 Drywell High Range Radiation Monitors FIGURE 11.1- 4 and FIGURE 11.1- 5

The purpose of the Drywell High Range Radiation Monitoring System is to detect significant radioactive releases inside the drywell, to assess these releases, to provide long-term post accident surveillance, and to provide emergency plan actuation.

75. RADIATION CONTROL 3

Access is required to a Unit One plant area for a routine inspection. Radiation levels in the area are 2.4 Rem/hr.

Which one of the following completes the statements below?

IAW AD-RP-ALL-2005, Posting of Radiological Hazards, this area is required to be posted as a ___(1)___.

IAW 0E&RC-0040, Administrative Controls for High Radiation Areas, Locked High Radiation Areas, and Very High Radiation Areas, the minimum approval required to enter this area is ___(2)___.

A. (1) Very High Radiation Area
   (2) RP Supervisor

B. (1) Very High Radiation Area
   (2) Operations Manager

C. (1) Locked High Radiation Area
   (2) RP Supervisor

D. (1) Locked High Radiation Area
   (2) Operations Manager

Answer: C

K/A:
G2.3.07 Ability to comply with radiation work permit requirements during normal or abnormal conditions. (CFR: 41.12 / 45.10)

RO/SRO Rating: 3.5/3.6

Pedigree: 2012 NRC Exam Modified

Objective: LOI-CLS-LP-201-F, Objective 10
Explain the requirement regarding control of High Radiation Areas per E&RC-0040.

Reference: None

Cog Level: Low

Explanation: Locked High Radiation Area (LHRA) criteria is > 1000 mrem/hr at 30 cm but < 500 Rads/hr at one meter. Very High Radiation Area (VHRA) criteria is 1000 mrem/hr at 30 cm and > 500 Rads/hr at one meter. See Definitions in Notes Section. This question provides criteria for a VHRA which requires RP Supervisor, RP Manager and Plant Manager approval for a routine entry IAW 0E&RC-0040.

Distractor Analysis:

Choice A: Plausible because raising the radiation levels would lead to a VHRA. Part 2 is correct.

Choice B: Plausible because raising the radiation levels would lead to a VHRA. Shift Manager can approve VHRA in emergency. Operations Manager is in the Operations chain of command.

Choice C: Correct answer, see explanation.

Choice D: Plausible because first part is correct. Shift Manager can approve VHRA in emergency. Operations Manager is in the Operations chain of command.

SRO Basis: N/A

U1 Tech Specs:

5.7.2 High Radiation Areas with Dose Rates Greater than 1.0 rem/hour (at 30 centimeters from the radiation source or from any surface penetrated by the radiation), but less than 500 rads/hour (at 1 meter from the radiation source or from any surface penetrated by the radiation)

a. Each accessible entryway to such an area shall be conspicuously posted as a high radiation area and shall be provided with a locked door, gate, or guard that prevents unauthorized entry, and in addition:

1. All such door and gate keys shall be maintained under the administrative control of the shift superintendent or the radiation control supervisor or designated representative; and

2. Doors and gates shall remain locked or guarded except during periods of personnel or equipment entry or exit.

(continued)

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17. **Locked High Radiation Area (LHRA):** An area, accessible to individuals, in which radiation levels could result in an individual receiving a dose equivalent in excess of 1.0 rem (1000 mrem) (10 mSv) in one hour at 30 centimeters from the radiation source or from any surface that the radiation penetrates or an area accessible to individuals with dose rates in excess of 1.0 rem per hour at 30 cm from the radiation source or 30 cm from any surface that the radiation penetrates but less than 500 rads in one hour at one meter from the radiation source or from any surface penetrated by the radiation.

25. **Very High Radiation Area (VHRA):** An area, accessible to individuals, in which radiation levels from radiation sources external to the body could result in an individual receiving an absorbed dose in excess of 500 rads (five grays) in one hour at one meter from a radiation source or one meter from any source that the radiation penetrates.

5.15 Access to Locked High Radiation Areas

NOTE

- RP supervision must approve each Locked High Radiation Area entry. Use valid historical data to develop accurate exposure estimates.
- The RP Supervisor or Manager may elect to classify the activity as High Risk per Section 7.2 Procedure 9, to invoke additional radiological controls to prevent significant unplanned external or internal dose, including the use of stay times and the assignment of a time keeper.
- Exceptions may apply due to special circumstances which require urgent plant response.

1. **Perform** the following for all Locked High Radiation Area entries:

   a. **Obtain** RP supervision's approval for the LHRA entry and to issue the LHRA key.

      (1) Access to a LHRA requires the following approvals by RP supervision:

5.15 Access to Locked High Radiation Areas (continued)

(a) Less than 10 Rem per hour - RP Supervisor or RP General Supervisor (verbal approval)

(b) Greater than or equal to 10 Rem per hour - RP Manager/or designee (written approval/ signature required on Attachment 7, Part 1)

5.17 Access to Very High Radiation Areas

1. **WHEN** a key to a VHRA is needed,
THEN complete applicable portions of Attachment 7, Part 1.

2. **Obtain** appropriate site management approvals to enter into a VHRA.

a. Access to a VHRA requires the following approvals:

(1) RP Manager/or designee (written approval/signature required on Attachment 7, Part 1).

NOTE

Except for the RP Manager's/or designee's written approval/signature required on Attachment 7, Part 1, additional management approvals may be granted by telephone and documented on Attachment 7.

(2) Plant Manager or Site VP/or designees for entry into areas greater than 500 Rads/hr.

(3) In an emergency situation, the Operations Shift Manager may approve the VHRA entry in lieu of the Plant Manager or Site VP.

2012 NRC Exam:

Access is required to a Unit One plant area for inspection.
Radiation levels in the area are 1100 Mrem/hr at 30 cm and 510 Rads/hr at one meter from the radiation source.

Which one of the following choices completes the statements below IAW 0E&RC-0040, Administrative Controls for High Radiation Areas, Locked High Radiation Areas, and Very High Radiation Areas?

This area is required to be posted as a ____(1)____.

The MINIMUM approvals required to enter this area are the E&RC manager (or designee), Rad Protection Supervisor, and ____(2)____.

A. (1) Very High Radiation Area
(2) Plant General Manager
B. (1) Very High Radiation Area
(2) Shift Manager
C. (1) Locked High Radiation Area
(2) Plant General Manager
D. (1) Locked High Radiation Area
(2) Shift Manager

76. S203000 1

During a LOCA with a LOOP, the following plant conditions exist:

2A RHR pump	Injecting and has just exceeded its NPSH Limit
2A CS pump	Injecting and approaching its NPSH Limit
All other ECCS Pumps	Unavailable
Reactor Water Level	2/3 core height and steady

Which one of the following completes the statements below?

Continued RHR Pump operation outside its NPSH limit ___(1)___ authorized IAW 0OI-37.4, Reactor Vessel Control Procedure Basis Document.

The CRS will direct performance of ___(2)___ to maintain adequate core cooling IAW RVCP?

A. (1) is
   (2) LEP-01, Alternate Coolant Injection, Section 5, Fire Protection/ Demineralized Water Tank Injection

B. (1) is
   (2) 2OP-18, Core Spray System Operating Procedure, Section 8.2, Shifting Suction Source from Suppression Pool to CST

C. (1) is NOT
   (2) LEP-01, Alternate Coolant Injection, Section 5, Fire Protection/ Demineralized Water Tank Injection

D. (1) is NOT
   (2) 2OP-18, Core Spray System Operating Procedure, Section 8.2, Shifting Suction Source from Suppression Pool to CST

Answer: A

K/A:
203000 RHR/LPCI: Injection Mode
G2.4.6 Knowledge of EOP mitigation strategies. (CFR: 41.10 / 43.5 / 45.13)

RO/SRO Rating: 3.7/4.7

Pedigree: 2010-2 NRC exam

Objective: CLS-LP-300-D, Objective 9 Given plant conditions, the Reactor Vessel Control Procedure, and which steps have been completed, determine the required operator actions.

Reference: None

Cog Level: High

Explanation: From OI-37.4, Immediate and catastrophic failure is not expected if a pump is operated beyond the NPSH or vortex limit. The undesirable consequences of uncovering the reactor core could thus outweigh the risk of equipment damage, so operation outside of the NPSH limit is required to maintain adequate core cooling. LEP-01 uses the RHR B loop for injection of Fire Water to the vessel. The CS pump cannot be transferred to the CST as this would require the pumps to be shutdown to perform this evolution.

Distractor Analysis:

Choice A: Correct Answer, see explanation.

Choice B: Plausible because immediate pump damage is not expected to occur and the pump is required to maintain adequate core cooling. If level was higher this would be correct. The CS pump cannot be secured to transfer the suction to the CST, or adequate core cooling would not be assured.

Choice C: Plausible because immediate pump damage is not expected to occur and the pump is required to maintain adequate core cooling. If level was higher this would be correct. The second part is correct.

Choice D: Plausible because with level at 2/3 core height the CS pump cannot be secured to transfer the suction to the CST, or adequate core cooling would not be assured.

SRO Basis: Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations. [10 CFR 55.43(b)(5)]

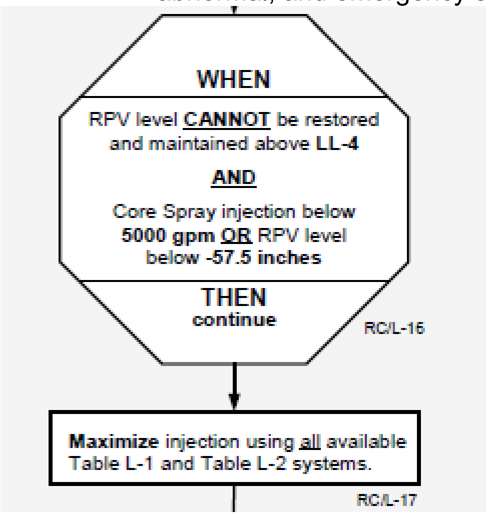

Table L-1
Preferred Injection Systems

System	Operating Details
Condensate/ Feedwater (OP-32)	**Defeat** if necessary: • High RPV level turbine trip per EOP−01-SEP−10
CRD	**Use** EOP-01-SEP-09
RCIC (OP-16)	**Use** CST suction if **NO** unit SBO **Transfer** to torus suction if unit SBO • **Return** to CST suction when torus temperature **190°F** **Defeat** if necessary per EOP−01-SEP−10: • Low RPV pressure isolation • High RPV level closure of the turbine steam supply valve • High exhaust pressure turbine trip • High area temperature isolations
HPCI (OP-19)	**Use** CST suction if **NO** unit SBO **Transfer** to torus suction if unit SBO • **Return** to CST suction when torus temperature **170°F** **Defeat** if necessary per EOP−01-SEP−10: • High RPV level turbine trip • High area temperature isolations • High torus level suction transfer
Core Spray (OP-18)	**Limit** flow to **5000 gpm**
LPCI (OP-17)	**Inject** through the heat exchangers as soon as possible

Table L-2
Alternate Injection Subsystems

System		Operating Details
SLC boron tank		**Start** SLC Pumps A and B **WHEN** SLC tank level drops to **0%**, **THEN stop** SLC Pumps A and B
SLC demin water/fire water		**Use** EOP-01-LEP-01
Heater Drain Pumps		
Demin water		
RHR B Loop	Fire System	
	Service Water	
	Demin water	
RCIC at RSDP		**Use** EOP-01-LEP-04 **Use** CST suction if **NO** unit SBO **Transfer** to torus suction if unit SBO • **Return** to CST suction when torus temperature **190°F** **Defeat** if necessary per EOP-01-LEP-04: • High RPV level closure of the turbine steam supply valve
LPCI SBO operation		**Align** electrical power per EOP-01-SBO-14 • 750 KW required
RCIC local manual operation		**Use** EOP-01-LEP-01 **Use** CST suction if available **Use** only if: • **NO** AC power available • **NO** Div II DC power available
EDMP		**Use** EDMG–004 actions for RPV injection
FLEX pump		**Use** FSG-002

IF	THEN
PCPL A CANNOT be maintained in safe region, but only if adequate core cooling is assured	**Terminate** RPV injection from sources external to the primary containment
RPV level drops to **LL-4**	**Perform** Alternative Source Term actions per EOP-01-SEP-11

RC/L-3

Step RC/L-7 through RC/L-9 (continued)

Systems may now be operated irrespective of NPSH and vortex limits, since restoration of adequate core cooling takes precedence over adherence to normal operating limits. The consequences of uncovering the reactor core outweigh the risk of equipment damage which could result if NPSH or vortex limits are exceeded. Immediate and catastrophic pump failure is not expected to occur.

8.2.1 Initial Conditions

1. Core Spray Loop A(B) in Standby in accordance with Section 5.1. ☐

2. Reactor in Mode 4 or 5. ☐

3. CST level greater than 16 feet on *CST LEVEL, CO-LI-1160A*, or *CST LEVEL, CO-LI-1160B*, or greater than 14 feet 5 inches using a local pressure gauge in accordance with 0OP-31.2, Condensate and Demineralized Water Storage and Transfer System. ☐

8.2.2 Procedural Steps

Loop A(B):_____

77. S209001 1

Unit Two is operating at rated power. During performance of 0PT-07.2.4A, Core Spray Loop A Operability, the RB AO reports that the Core Spray A room cooler breaker has tripped on thermal overload.

Which one of the following completes the statements below?

Due to the tripped room cooler breaker, Core Spray Loop A is ___(1)___.

Based on the conditions above, an immediate one time attempt to reset the Core Spray A room cooler breaker ___(2)___ allowed IAW AD-OP-ALL-1000, Conduct of Operations.

A. (1) operable
   (2) is

B. (1) operable
   (2) is NOT

C. (1) inoperable
   (2) is

D. (1) inoperable
   (2) is NOT

Answer: D

K/A:

209001 Low Pressure Core Spray System

A2. Ability to (a) predict the impacts of the following on the LOW PRESSURE CORE SPRAY SYSTEM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: (CFR: 41.5 / 45.6)07

07 Loss of room cooling

RO/SRO Rating: 2.6/2.8

Pedigree: 2008 NRC exam.

Objective: CLS-LP-18, Objective 18
Given plant conditions and TS, including bases, TRM, ODCM, and COLR, determine the required actions to be taken in accordance the TS associated with the Core Spray System. (SRO/STA only)

Reference: None

Cog Level: High

Explanation: IAW 0OI-01.01, if a room cooler is inoperable then the associated equipment is inoperable IAW the applicable TS. The room cooler is not required in Mode 4 or 5. IAW AD-OP-ALL-1000, the one time reset would be valid only if needed in emergency conditions without the cause of the trip known.

Distractor Analysis:

Choice A: Plausible because the TS basis does not address the room cooler as part of the operability. If this was during an emergency condition then resetting the breaker once would be appropriate.

Choice B: Plausible because the TS basis does not address the room cooler as part of the operability. Resetting the breaker would not be appropriate under these conditions.

Choice C: Plausible because the loop is declared inoperable and if this was during an emergency condition then resetting the breaker once would be appropriate

Choice D: Correct Answer, see explanation

SRO Basis: Facility operating limitations in the TS and their bases. [10 CFR 55.43(b)(2)]

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS) AND REACTOR CORE ISOLATION COOLING (RCIC) SYSTEM

3.5.1 ECCS—Operating

LCO 3.5.1 Each ECCS injection/spray subsystem and the Automatic Depressurization System (ADS) function of six safety/relief valves shall be OPERABLE.

APPLICABILITY: MODE 1,

BNP CONDUCT OF OPERATIONS SUPPLEMENT	0OI-01.01
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4. ECCS Room Coolers {7.1.3}

NOTE
• The following step is not required to be performed if the ECCS Room Cooler is INOPERABLE due to the loss of a 4160V or 480V E-Bus. E-Bus INOPERABILITY impacts the OPERABILITY of ECCS subsystems. Technical Specifications and the SFDP will provide Required Actions to be taken for the loss of the E-Bus. • In Mode 4 and Mode 5, ECCS Room Coolers are not required to be OPERABLE to support OPERABILITY of the associated ECCS Systems.

a. When any ECCS Room Cooler is determined to be INOPERABLE, then the ECCS equipment associated with that room cooler is to be declared INOPERABLE per the applicable Technical Specifications.

5.19 Resetting Protective Devices

{7.1.4}

5.19.1 Standards

1. Protective devices should not be reset without a clear understanding of the reason for the protective device trip.

2. The overriding priority for the operating crew upon the trip of any protective device is to stabilize the plant and restore the systems to the safest possible condition.

5.19.2 Expectations

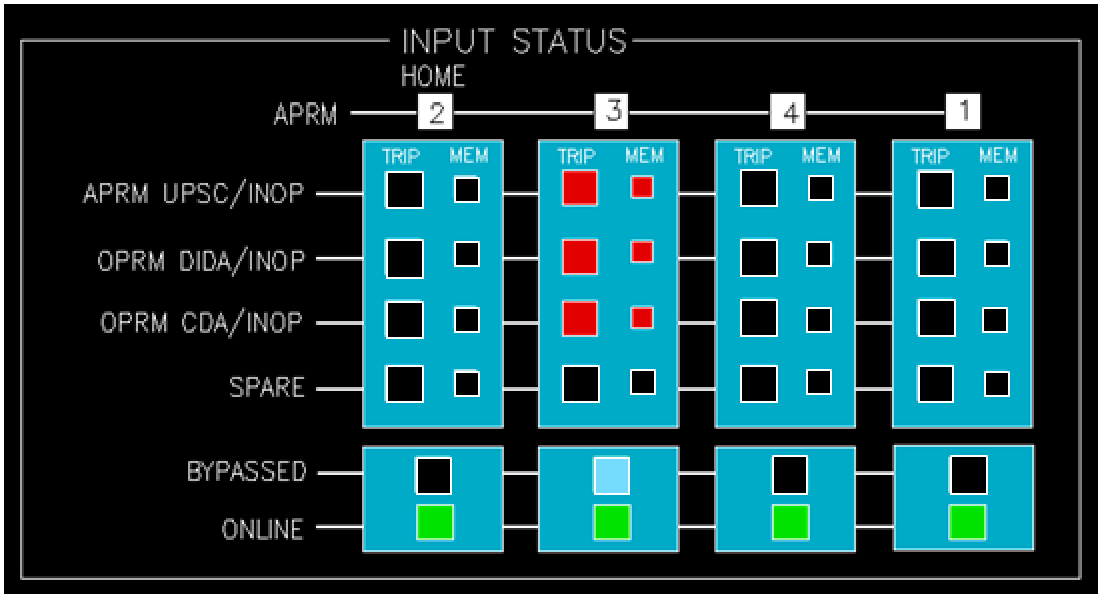
1. Protection devices which have actuated (breakers, fuses, bistables, MOV thermal overloads, lockouts, etc.) should only be restored with shift supervision approval, under the following conditions. The following conditions do not apply to 120 volt breakers that only supply lighting or receptacles.

   a. The cause of the actuation has been identified and corrected.

   b. Restoring the protective device is not recommended unless plant conditions dictate that the component repositioning must be completed before Maintenance and Engineering personnel are available. Remote operation of the component with no personnel in the immediate area after resetting the protective device is recommended if repositioning is required prior to completion of the evaluation by Maintenance and Engineering.

2. The SM may approve additional protective device resetting after consultation with Engineering.

78. S212000 1
A Unit One operator observes the following indications on Panel P608:



Subsequently, APRM 1 fails upscale. Which one of the following completes the statements below in response to this failure?

(Reference provided)

RPS Channel A ____(1)____ de-energize.

IAW Tech Specs, APRM 1 must be placed in trip in ____(2)____ hours.

A. (1) will
(2) 6

B. (1) will
(2) 12

C. (1) will NOT
(2) 6

D. (1) will NOT
(2) 12

Answer: D

K/A:
212000 Reactor Protection System

A2 Ability to (a) predict the impacts of the following on the REACTOR PROTECTION SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: (CFR: 41.5 / 45.6)

06 High reactor power

RO/SRO Rating: 4.1/4.2

Pedigree: New

Objective: LOI-CLS-LP-009.6, Objective 7
Describe the operational relationships between the PRNMS and the following:
a. Reactor Protection System
LOI-CLS-LP-09.6, Objective 27b
Given plant conditions associated with the PRNMS system determine the required action(s): to be taken in accordance with Technical Specifications, TRM, ODCM and COLR associated with the PRNMS System. (SRO/STA only) (LOCT)

Reference: TS 3.3.1.1

Cog Level: Hi

Explanation: While APRMs 1 and 3 are conventionally considered to be RPS A inputs with APRM 3 bypassed this would not trip RPS. TS require the channel to be placed in a trip condition in 12 hours. Condition B is excluded for this function per the note.

Distractor Analysis:

Choice A: Plausible because normally two APRM high tips would trip RPS, but APRM 3 is bypassed. While more than one function is inoperable the 6 hour requirement is excluded by the note.

Choice B: Plausible because normally two APRM high tips would trip RPS, but APRM 3 is bypassed.

Choice C: Plausible because more than one function is inoperable but the 6 hour requirement is excluded by the note.

Choice D: Correct Answer, see explanation

SRO Basis: Facility operating limitations in the TS and their bases. [10 CFR 55.43(b)(2)]

3.3 INSTRUMENTATION

3.3.1.1 Reactor Protection System (RPS) Instrumentation

LCO 3.3.1.1 The RPS instrumentation for each Function in Table 3.3.1.1-1 shall be OPERABLE.

APPLICABILITY: According to Table 3.3.1.1-1.

ACTIONS

------------------------------------NOTE------------------------------------
Separate Condition entry is allowed for each channel.

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One or more required channels inoperable.	A.1	Place channel in trip.	12 hours
	<u>OR</u>		
	A.2	----------NOTE--------- Not applicable for Functions 2.a, 2.b, 2.c, 2.d, or 2.f. ----------------------------- Place associated trip system in trip.	12 hours

B.	---------NOTE----------- Not applicable for Functions 2.a, 2.b, 2.c, 2.d, or 2.f. --------------------------- One or more Functions with one or more required channels inoperable in both trip systems.	B.1 <u>OR</u> B.2	Place channel in one trip system in trip. Place one trip system in trip.	6 hours 6 hours
C.	One or more Functions with RPS trip capability not maintained.	C.1	Restore RPS trip capability.	1 hour
D.	Required Action and associated Completion Time of Condition A, B, or C not met.	D.1	Enter the Condition referenced in Table 3.3.1.1-1 for the channel.	Immediately
E.	As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	E.1	Reduce THERMAL POWER to < 26% RTP.	4 hours
F.	As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	F.1	Be in MODE 2.	6 hours
G.	As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	G.1	Be in MODE 3.	12 hours
H.	As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	H.1	Initiate action to fully insert all insertable control rods in core cells containing one or more fuel assemblies.	Immediately
I.	As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	I.1	Initiate alternate method to detect and suppress thermal hydraulic instability oscillations.	12 hours
J.	Required Action and associated Completion Time of Condition I not met.	J.1	Reduce THERMAL POWER to < 20% RTP.	4 hours

79. S215001 1

Unit Two is performing a TIP trace and the TIP Mode Switch is switched from AUTOMATIC to MANUAL when the detector reaches the core top limit.

A small steam leak in Containment cause Drywell pressure to rise to 2.7 psig.

Which one of the following completes the statements below?

The TIP Detector ___(1)___ be expected to retract from the core.

If conditions exist for the TIP to withdraw to the in-shield position, and the system fails, the REQUIRED Tech Spec 3.6.1.3 ACTION is ___(2)___.

(Reference provided)

A. (1) would
   (2) A.1 and A.2

B. (1) would
   (2) C.1 and C.2

C. (1) would NOT
   (2) A.1 and A.2

D. (1) would NOT
   (2) C.1 and C.2

Answer: B

K/A:
215001 Traversing In-Core Probe
A2 Ability to (a) predict the impacts of the following on the TRAVERSING IN-CORE PROBE; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: (CFR: 41.5 / 45.6)
07 Failure to retract during accident conditions

RO/SRO Rating: 3.4/3.7

Pedigree: New

Objective: LOI-CLS-LP-09, Objective 5b.
Explain the effects of the following on the TIP System: High Drywell Pressure

Reference: None

Cog Level: High

Explanation: With an isolation signal, the TIP should retract and the ball valve should close. Tech Spec 6.6.1.3.C is applicable to the TIP System in MODES 1,2, and 3. With Drywell pressure at 2.7 psig, the plant would be in MODE 3 and Conditions C.1 and C.2 would be applicable.

Distractor Analysis:

Choice A: Plausible because part 1 is correct. Part 2 is plausible because there is a ball valve and a shear valve. The shear valve is not considered a PCIV, so Action A for 2 PCIVs is plausible, but not correct.

Choice B: Correct Answer, see explanation.

Choice C: Plausible because there are 3 positions for the TIP Mode switch. The OP cautions to not put the Mode switch in Off, because that inops the isolation capability. In Manual, the TIP will still withdraw and isolate. Part 2 is plausible because there is a ball valve and a shear valve. The shear valve is not considered a PCIV, so Action A for 2 PCIVs is plausible, but not correct.

Choice D: Plausible because there are 3 positions for the TIP Mode switch. The OP cautions to not put the Mode switch in Off, because that inops the isolation capability. In Manual, the TIP will still withdraw and isolate. Part 2 is correct.

SRO Basis: Facility operating limitations in the TS and their bases. [10 CFR 55.43(b)(2)]

4.2 Abnormal Operation

4.2.1 TIP Operation with a Group 2 Isolation Signal Present

Upon receipt of a Group 2 Isolation signal from the PCIS System:

- Low Reactor Water Level
- Hi Drywell Pressure

On this signal, any TIP not in the Shield Chamber is automatically transferred to the manual reverse mode of operation (Result of relay logic in Drive Control Unit). The detector will be retracted from the core at fast speed. When the detector is In-Shield as indicated by the limit switch, the TIP Ball Valve is closed.

4.2.2 TIP Fails To Isolate With an Isolation Signal Present

CAUTION
1. The MODE Switch should **NOT** be placed OFF while the TIP Probe is inserted past the TIP Ball Valve to ensure that the PCIS Isolation Logic is not defeated. 2. The MODE Switch should **NOT** be placed in OFF until the TIP Ball Valve CLOSED position indicating light is ON. 3. IF the TIP Probe becomes stuck beyond the shield, the Unit SCO must be notified that the Primary Containment Isolation Logic is defeated for the associated TIP Ball Valve. 4. The TIP Ball Valve will **NOT** CLOSE and the TIP Probe will **NOT** STOP if the shield proximity switch fails to actuate while retracting the TIP Probe from the Indexer to the in-shield position. Should the proximity switch fail, using the MANUAL mode, the TIP Probe must be placed at the in-shield position and the Unit SCO informed immediately to determine the TIP Ball Valve operability (Primary Containment Isolation Valve TECH SPEC 3.6.1.3).

Several conditions can cause this situation:

- Ball valve will not close
- TIP Detector will not retract (stuck)

Given the need to isolate the guide tube, the Shear Valve is capable of being closed by operating the key lock switch (S-1) at the Valve Control Monitor. The Shear Valve itself is not a PCIS Valve.

Operators need to be aware that a Technical Specification LCO needs to be initiated if either of the following conditions occurs:

a. The TIP Detector is inserted beyond the TIP Ball Valve and the associated TIP Machine power is turned off. The TIP logic is defeated in this condition and a Group 2 isolation signal will not occur on this TIP probe.

b. A TIP Detector becomes stuck beyond the TIP Ball Valve.

1.0 PURPOSE

This procedure provides the guidance for operation of the Traversing Incore Probe System.

2.0 SCOPE

This procedure provides the prerequisites, precautions, limitations, and instructional guidance for the Startup, Normal Operation, Shutdown, and Infrequent Operation of the Traversing Incore Probe System.

3.0 PRECAUTIONS AND LIMITATIONS

1. The amount of time the detector is in the core is minimized to limit detector and cable activation. .... ☐

2. The Mode switch is **NOT** to be placed in OFF while the probe is inserted past the Ball Valve to ensure the isolation logic is **NOT** defeated. .... ☐

3. The Mode switch is **NOT** to be placed in OFF until Ball Valve Closed position indicating light is ON. .... ☐

4. The TIP Ball Valve will **NOT** close and the TIP Probe will **NOT** stop if the shield proximity switch fails to actuate while retracting the TIP Probe from the indexer to the IN-SHIELD position. .... ☐

4.0 GENERAL INFORMATION

1. The TIP Probes will automatically retract to the IN-SHIELD position, and the Guide Tube Ball Valves will automatically close upon receipt of the following signals during normal operation of the TIP System:

   - Low Level 1 (B21-LTM-N017A-1, B-1)
   - High drywell pressure (C72-PTM-N002A-1, B-1)

2. The following Technical Specifications requirements are observed for the TIP System:

   - Section 3.3.6.1, Primary Containment Isolation Instrumentation.
   - Section 3.6.1.3, Primary Containment Isolation Valves.

3. Attachment 2, Tip Location Chart provides additional TIP information.

3. **IF AT ANY TIME** the probe becomes stuck beyond the shield, **THEN notify** the Unit CRS that the Primary Containment Isolation Logic is defeated for the associated Ball Valve...................................... _________

3.6 CONTAINMENT SYSTEMS

3.6.1.3 Primary Containment Isolation Valves (PCIVs)

LCO 3.6.1.3 Each PCIV, except reactor building-to-suppression chamber vacuum breakers, shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3,

When associated instrumentation is required to be OPERABLE per LCO 3.3.6.1, "Primary Containment Isolation Instrumentation."

ACTIONS

------------------------------------NOTES------------------------------------

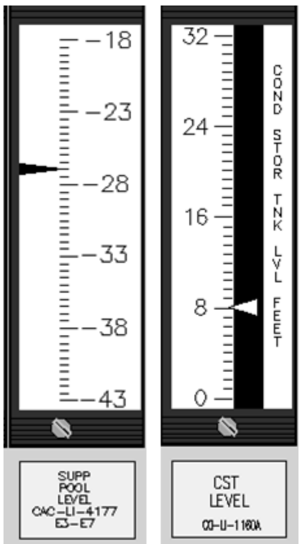
1. Penetration flow paths may be unisolated intermittently under administrative controls.
2. Separate Condition entry is allowed for each penetration flow path.
3. Enter applicable Conditions and Required Actions for systems made inoperable by PCIVs.
4. Enter applicable Conditions and Required Actions of LCO 3.6.1.1, "Primary Containment," when PCIV leakage results in exceeding overall containment leakage rate acceptance criteria.

C. ----------NOTE---------- Only applicable to penetration flow paths with only one PCIV. --------------------------- One or more penetration flow paths with one PCIV inoperable.	C.1 Isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve, closed manual valve, or blind flange. AND	8 hours except for excess flow check valves (EFCVs) AND 12 hours for EFCVs
	C.2 ----------NOTE---------- Isolation devices in high radiation areas may be verified by use of administrative means. --------------------------- Verify the affected penetration flow path is isolated.	Once per 31 days

(continued)

80. S217000 1
RCIC is being used to control level on Unit Two with the following plant conditions:

CST indication is lowering at a foot per minute.
Suppression Pool indication is rising at an inch per minute.



Which one of the following completes the statements below?

The RCIC system suction sources will swap in ___(1)___ .

The bases of the TS 3.6.2.2, Suppression Pool Water Level, upper level limit is to ___(2)___ during a DBA LOCA.

A. (1) 2 minutes
   (2) prevent excessive clearing loads from SRV discharges and excessive pool swell loads

B. (1) 2 minutes
   (2) prevent the cyclic condensation of steam at the downcomer openings of the drywell vents due to chugging

C. (1) 5 minutes
   (2) prevent excessive clearing loads from SRV discharges and excessive pool swell loads

D. (1) 5 minutes
   (2) prevent the cyclic condensation of steam at the downcomer openings of the drywell vents due to chugging

Answer: C

K/A:
217000 Reactor Core Isolation Cooling System
G2.4.47 Ability to diagnose and recognize trends in an accurate and timely manner utilizing the appropriate control room reference material. (CFR: 41.10 / 43.5 / 45.12)

RO/SRO Rating: 4.2/4.2

Pedigree: New

Objective: LOI-CLS-LP-016, Objective 14
Given plant conditions, predict the RCIC System response to the following conditions:
f. High/low Suppression Pool water level

Reference: None

Cog Level: High

Explanation: The RCIC system will transfer to the torus on low level in the CST (~3 feet 1 inch). The HPCI system also transfer to the torus on low CST level but additionally will transfer on high torus water level (-25 inches).
The TS bases documents lists the reason for the high torus water level limitation. The distractor, although not a TS Basis, is a condensation phenomenon that can occur in the Tours due to non-condensible gases being pushed from the Drywell to the Torus.

Distractor Analysis:

Choice A: Plausible because HPCI transfers at -25 inches. The second part is correct.

Choice B: Plausible because HPCI transfers at -25 inches. The second part describes a condensation phenomenon that can occur at the exit of the downcomers.

Choice C: Correct Answer, see explanation

Choice D: Plausible because the first part is correct and the second part describes a condensation phenomenon that can occur at the exit of the downcomers.

SRO Basis: Facility operating limitations in the TS and their bases. [10 CFR 55.43(b)(2)]

3.2 RCIC Pump Suction Control (Figures 16-2, 20, 25)

Normally, the RCIC System is in Standby, with the pump suction aligned to the CST via the Condensate Storage Tank Suction Valve, E51-F010. The Suppression Pool Suction Valves, E51-F029 and E51-F031, are normally closed.

On a low level in the CST, the pump suction automatically transfers to the suppression pool by the opening of the Suppression Pool Suction Valves. Once both suppression pool valves are full open, the Condensate Storage Tank Suction Valve automatically closes. The setpoint for the RCIC automatic suction transfer is as follows:

CST Level Low Tech Spec: ≥23'0" elev. (3' tank level)
Actual: 23'1" elev. (3'1" tank level)

Table 19-6 - HPCI Suppression Pool Suction Transfer Signals		
Signal	Setpoint	Tech Spec
CST Level Low	23'5" elev. (3'5" tank level)	≥23'4" elev. (≥3'4" tank level)
Suppression Pool Level	-25"	≤-2'

B 3.6 CONTAINMENT SYSTEMS

B 3.6.2.2 Suppression Pool Water Level

BASES

BACKGROUND

The suppression chamber is a toroidal shaped, steel pressure vessel containing a volume of water called the suppression pool. The suppression pool is designed to absorb the energy associated with decay heat and sensible heat released during a reactor blowdown from safety/relief valve (SRV) discharges or from a Design Basis Accident (DBA). The suppression pool must quench all the steam released through the downcomer lines during a loss of coolant accident (LOCA). This is the essential mitigative feature of a pressure suppression containment, which ensures that the peak containment pressure is maintained below the maximum allowable pressure for DBAs (62 psig). The suppression pool must also condense steam from the steam exhaust lines in the turbine driven systems (i.e., High Pressure Coolant Injection (HPCI) System and Reactor Core Isolation Cooling (RCIC) System) and provides the main emergency water supply source for the reactor vessel. The suppression pool volume ranges between 86,450 ft^3 at the low water level limit of -31 inches and 89,750 ft^3 at the high water level limit of -27 inches.

If the suppression pool water level is too low, an insufficient amount of water would be available to adequately condense the steam from the SRV quenchers, main vents, or HPCI and RCIC turbine exhaust lines. Low suppression pool water level could also result in an inadequate emergency makeup water source to the Emergency Core Cooling System. The lower volume would also absorb less steam energy before heating up excessively. Therefore, a minimum suppression pool water level is specified.

If the suppression pool water level is too high, it could result in excessive clearing loads from SRV discharges and excessive pool swell loads during a DBA LOCA. Therefore, a maximum pool water level is specified. This LCO specifies an acceptable range to prevent the suppression pool water level from being either too high or too low.

0EOP-UG

11. **Chugging:** An intermittent condensation phenomenon which occurs at the downcomer exit when the drywell is pressurized due to a small high energy (steam) leak inside the drywell. When a steam bubble collapses at the exit of the downcomers, the rush of water filling the void (some of it drawn up into the downcomer pipe) induces severe stress at the junction of the downcomer vent header. Repeated application of this stress can cause these joints to experience fatigue failure (i.e., crack) thereby creating a pathway which bypasses the pressure suppression function of the containment.

0OI-37.8:

The Torus Spray Initiation Pressure is defined to be the lowest torus pressure which can occur when 95% of the noncondensibles in the drywell have been transferred to the airspace of the torus. This pressure is utilized to preclude chugging: the cyclic condensation of steam at the downcomer openings of the drywell vents.

When a steam bubble collapses at the exit of the downcomers, the rush of water filling the void (some of it drawn up into the downcomer pipe) induces a severe stress at the junction of the downcomer and the vent header. Repeated application of this stress can cause these joints to experience fatigue failure (i.e., crack) thereby creating a pathway which bypasses the pressure suppression function of the containment. Subsequent steam discharges through the downcomers would directly pressurize the torus airspace rather than being discharged to and condensed in the torus.

81. S219000 1

An event on Unit One has resulted in the following plant conditions:

Reactor pressure	1000 psig
Reactor Water Level	120 inches
Control Rod Positions	All unknown
APRMs	Downscale
Drywell pressure	3 psig
Torus pressure	2 psig
Torus water temp	150° F
Torus water level	-4 feet

Which one of the following identifies the status of the Heat Capacity Temperature Limit (HCTL) and the required procedure for reactor level control?

(Reference provided)

	HCTL	Level Control Leg of Procedure
A.	has been exceeded	RVCP
B.	has been exceeded	ATWS
C.	has NOT been exceeded	RVCP
D.	has NOT been exceeded	ATWS

Answer: B

K/A:
219000 RHR/LPCI: Torus/Suppression Pool Cooling Mode
G2.1.25 Ability to interpret reference materials, such as graphs, curves, tables, etc.
(CFR: 41.10 / 43.5 / 45.12)

RO/SRO Rating: 3.9/4.2

Pedigree: 2010-1 NRC Exam (modified for EOP R3, changed Pressure Control leg to Level leg)

Objective: LOI-CLS-LP-300-L, Objective 05a
Given the PCCP, determine the appropriate actions if any of the following limits are approached or exceeded: Heat Capacity Temperature Limit

Reference: 0EOP-01-UG, Attachment 7 (HCTL only)

Cog Level: High

Explanation: HCTL has been exceeded. Select the graph line immediately below torus water level as the limit. With rods unknown the operator would be in ATWS Control Procedure.

Distractor Analysis:

Choice A: Plausible because although first part is correct, rods are unknown which would determine ATWS.

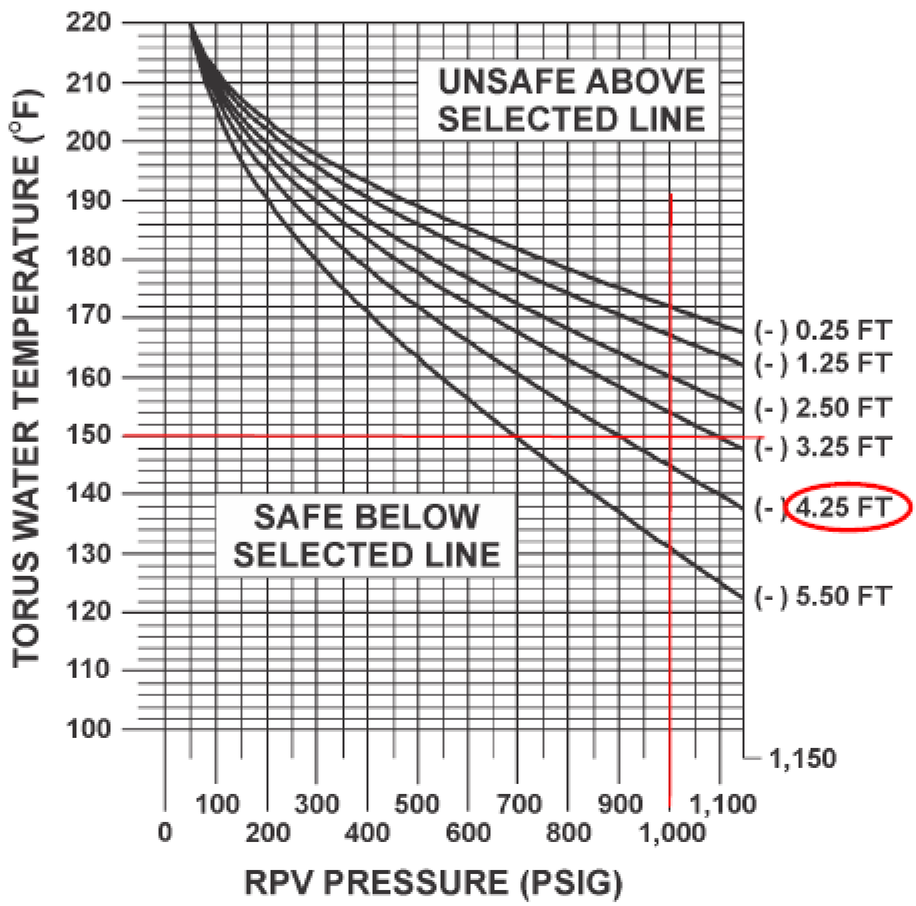
Choice B: Correct Answer, see explanation.

Choice C: Plausible because HCTL graph must be correctly read (cannot interpolate graph for the torus level must use the line below the level), and HCTL has been exceeded. With rods unknown, ATWS would be the correct procedure.

Choice D: Plausible because HCTL graph must be correctly read (cannot interpolate graph for the torus level must use the line below the level), and HCTL has been exceeded. Second part is correct.

SRO Basis: Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations (43(b)(5)

Heat Capacity Temperature Limit



82. S261000 1

Unit Two is operating at rated power when the following sequence of events occurs:

12/13 @ 0100, 2B SBGT is declared Inoperable for scheduled maintenance
12/15 @ 1230, 2A SBGT is declared Inoperable due to fan failure
12/15 @ 1430, 2B SBGT is declared Operable

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One SGT subsystem inoperable in MODE 1, 2 or 3.	A.1 Restore SGT subsystem to OPERABLE status.	7 days

IAW Technical Specifications, which one of the following identifies the applicable completion time to restore 2A SBGT train to operable status?

A. 12/20 @ 0100

B. 12/21 @ 0100

C. 12/22 @ 1230

D. 12/23 @ 1230

Answer: B

K/A:
261000 Standby Gas Treatment System
A2 Ability to (a) predict the impacts of the following on the STANDBY GAS TREATMENT SYSTEM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: (CFR: 41.5 / 45.6)
05 Fan trips

RO/SRO Rating: 3.0/3.1

Pedigree: New

Objective: LOI-CLS-LP-010, Objective 11
Given plant conditions associated with the Standby Gas Treatment system, determine the required action(s): b. to be taken in accordance with Technical Specifications, TRM, and COLR. (LOCT) (SRO/STA Only)

Reference: None

Cog Level: High

Explanation: No discriminatory value in determining the impact of the fan failure, so only wrote the question to the use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations part of the K/A.

Completion Times (Tech Spec 1.3) states that for the failure of 2A that you must use the more restrictive of its completion time or the completion time of the first train failure with an extension of 24 hours. This would make the completion time of 2B SBGT train 12/20 @ 0100 plus 24 hours the more restrictive time.

Distractor Analysis:

Choice A: Plausible because this is the expiration time of the 2B SBGT train, with no extension applied for the failure of 2A SBGT.

Choice B: Correct Answer, see explanation.

Choice C: Plausible because this is the completion time for 2A SBGT train if the 2B was not already failed or if this would have been more restrictive than the allowed 24 hour extension

Choice D: Plausible because this is the completion time for 2A SBGT train plus the 24 hour extension.

SRO Basis: Facility operating limitations in the TS and their bases. [10 CFR 55.43(b)(2)]

However, when a subsequent division, subsystem, component, or variable expressed in the Condition is discovered to be inoperable or not within limits, the Completion Time(s) may be extended. To apply this Completion Time extension, two criteria must first be met. The subsequent inoperability:

a. Must exist concurrent with the first inoperability; and

b. Must remain inoperable or not within limits after the first inoperability is resolved.

The total Completion Time allowed for completing a Required Action to address the subsequent inoperability shall be limited to the more restrictive of either:

a. The stated Completion Time, as measured from the initial entry into the Condition, plus an additional 24 hours; or

b. The stated Completion Time as measured from discovery of the subsequent inoperability.

The above Completion Time extension does not apply to those Specifications that have exceptions that allow completely separate re-entry into the Condition (for each division, subsystem, component or variable expressed in the Condition) and separate tracking of Completion Times based on this re-entry. These exceptions are stated in individual Specifications.

The above Completion Time extension does not apply to a Completion Time with a modified "time zero." This modified "time zero" may be expressed as a repetitive time (i.e., "once per 8 hours," where the Completion Time is referenced from a previous completion of the Required Action versus the time of Condition entry) or as a time modified by the phrase "from discovery . . ." Example 1.3-3 illustrates one use of this type of Completion Time. The 10 day Completion Time specified for Condition A and B in Example 1.3-3 may not be extended.

83. S290002 1

Unit One is performing a reactor shutdown in preparation for a refueling outage.

Which one of the following completes the statements below IAW Technical Specifications?

(Reference provided)

The maximum cooldown rate is limited to ___(1)___ change in any one hour period.

Violation of this limit is a ___(2)___ IAW 0OI-1.07, Notifications.

A. (1) 30°F
   (2) 8-hour Report

B. (1) 30°F
   (2) Safety Significance Concern

C. (1) 100°F
   (2) 8-hour Report

D. (1) 100°F
   (2) Safety Significance Concern

Answer: D

K/A:
290002 Reactor Vessel Internals
A2 Ability to (a) predict the impacts of the following on the REACTOR VESSEL INTERNALS ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: (CFR: 41.5 / 45.6)
04 Excessive heatup/cooldown rate

RO/SRO Rating: 3.7/4.1

Pedigree: New

Objective: LOI-CLS-LP-001, Objective 14
Given plant conditions and Technical Specifications, including the Bases, TRM, ODCM and COLR, determine the required action(s) to be taken in accordance with Technical specifications associated with the Reactor Vessel and Internals System. (SRO/STA only)

Reference: None

Cog Level: Fund

Explanation: Predicting the impact of violating the heatup rate (brittle fracture) is in the bases for TS 3.4.9. The normal heatup / cooldown rate is 100°F but during hydrostatic testing it is reduced to 30°F in any 1 hour period. This requirement is listed in the surveillance testing requirements/bases and is not listed "above the line" of the TS.

Distractor Analysis:

Choice A: Plausible because the heatup/cooldown rate during RCS in-service leak and hydrostatic testing is 30°F and an event or condition that results in the nuclear power plant being in an unanalyzed condition that significantly degrades plant safety is an 8-hour report is one example.

Choice B: Plausible because the heatup/cooldown rate during RCS in-service leak and hydrostatic testing is 30°F and second part is correct.

Choice C: Correct Answer, see explanation.

Choice D: Plausible because the normal heatup rate is 100°F and an event or condition that results in the nuclear power plant being in an unanalyzed condition that significantly degrades plant safety is an 8-hour report is one example.

SRO Basis: Facility operating limitations in the TS and their bases. [10 CFR 55.43(b)(2)
Conditions and limitations in the facility license. [10 CFR 55.43(b)(1)]

LCO The elements of this LCO are:

a. RCS pressure and temperature are within the applicable limits specified in Figures 3.4.9-1 and 3.4.9-2, and heatup or cooldown rates are ≤ 100°F in any 1 hour period, during RCS heatup and cooldown;

b. RCS pressure and temperature are within the applicable limits in Figures 3.4.9-3, 3.4.9-4, or 3.4.9-5 and heatup or cooldown rates are ≤ 30°F in any 1 hour period, during RCS inservice leak and hydrostatic testing;

c. The temperature difference between the reactor vessel bottom head coolant and the RPV coolant is ≤ 145°F during recirculation pump startup;

d. The temperature difference between the reactor coolant in the respective recirculation loop and in the reactor vessel is ≤ 50°F during recirculation pump startup;

e. RCS pressure and temperature are within the criticality limits specified in Figure 3.4.9-2, prior to achieving criticality; and

f. The reactor vessel flange and the head flange temperatures are ≥ 70°F when tensioning the reactor vessel head bolting studs.

(continued)

Brunswick Unit 1 B 3.4.9-3 Revision No. 38

The LCO establishes operating limits that provide a margin to brittle failure of the reactor vessel and piping of the reactor coolant pressure boundary (RCPB). The vessel (including its appurtenances) is the component most subject to brittle failure. Therefore, the LCO limits apply mainly to the vessel (including its appurtenances).

The consequence of violating the LCO limits is that the RCS has been operated under conditions that can result in brittle failure of the RCPB, possibly leading to a nonisolable leak or loss of coolant accident. In the event these limits are exceeded, an evaluation must be performed to determine the effect on the structural integrity of the RCPB components. ASME Code, Section XI, Appendix E (Ref. 9), provides a recommended methodology for evaluating an operating event that causes an excursion outside the limits.

ATTACHMENT 4
Page 1 of 3

Initial Safety Significance Determination Checklist

NOTE

If the answer to any of the below-listed questions is YES, a safety significance concern might exist and further analysis might be necessary.

1. ______ RPV pressure greater than or equal to safety/relief valve set pressure.
YES NO

2. ______ RPV temperature has exceeded a maximum cooldown of 100°F in any 1 hour period.
YES NO

8 HOUR REPORTABILITY			
ITEM #	**YES**	**NO**	**DESCRIPTIVE QUESTION**
3.1			Has the event or condition resulted in the condition of the nuclear power plant, including its principal safety barriers, being seriously degraded? [10 CFR 50.72(b)(3)(ii)(A)]
3.2			Has the event or condition resulted in the nuclear power plant being in an unanalyzed condition that significantly degrades plant safety? [10 CFR 50.72(b)(3)(ii)(B)]

84. S295003 1

Unit Two was operating at rated power when an electrical fault occurred.

After the event the RO performed the Scram immediate operator actions, observed DG3 is running, DG4 is locked out, and the following electrical indications:

Which one of the following completes the statements below?

The cause of the electrical transient is from a ___(1)___ lockout.

If DG4 becomes available, the CRS will direct it to be started IAW ___(2)___.

A. (1) SAT
(2) 0OP-39, Diesel Generator Operating Procedure

B. (1) SAT
(2) 0AOP-36.1, Loss of any 4160V Buses or 480V E-Buses

C. (1) Main Generator
(2) 0OP-39, Diesel Generator Operating Procedure

D. (1) Main Generator
(2) 0AOP-36.1, Loss of any 4160V Buses or 480V E-Buses

Answer: B

K/A:
295003 Partial or Complete Loss of A.C. Power
AA2 Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF A.C. POWER : (CFR: 41.10 / 43.5 / 45.13)
04 System Lineups

RO/SRO Rating: 3.5/3.7

Pedigree: New

Objective: LOI-CLS-LP-050, Objective 20
Given plant conditions predict the changes in Unit 1 and/or Unit 2 parameters associated with the operation of the following equipment:
b. SAT lockout relay **(LOCT)**

Reference: None

Cog Level: High

Explanation: The key to determining the cause is Bus 2B, being de-energized from the SAT must be a lockout of the SAT. With only one diesel even though it is on Unit 2 they would enter AOP-36.1. The SBO directs if a DG becomes available to start IAW op-39, while AOP-36.1 has an attachment to start a DG that becomes available.

Distractor Analysis:

Choice A: Plausible because an SAT lockout is correct and if SBO is entered it would direct restarting the DG IAW the operating procedure.

Choice B: Correct Answer, see explanation

Choice C: Plausible because on a main generator lockout the UAT to 2D and 2C buses would be open (but the 2B buss would not) and if SBO is entered it would direct restarting the DG IAW the operating procedure.

Choice D: Plausible because on a main generator lockout the UAT to 2D and 2C buses would be open (but the 2B buss would not) and the second part is correct.

SRO Basis: Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations. [10 CFR 55.43(b)(5)]

4.2.1 Actions Determination (continued)

NOTE

Resetting the diesel generator lockout relay may result in an automatic start of the diesel generator. ☐

b. **Depress** Lockout Reset pushbutton on local diesel generator control panel. ☐

c. **Perform** Section 4.2.12 on page 62 concurrently with this section. ☐

While in this procedure:

IF	THEN
SAT <u>OR</u> UAT becomes available	1. Coordinate plant alignment and recovery of electrical system with ERO 2. Energize switchyard per OP-50 3. Energize selected BOP buses per OP-50 4. Energize selected E-buses per OP-50.1
An EDG becomes available	1. Coordinate plant alignment and recovery of E-bus with ERO 2. <u>WHEN</u> EDG support MCC available, <u>THEN</u> start EDG per OP-39 3. Energize associated E-bus per OP-50.1

SBO-2

85. S295015 1

Unit Two has just scrammed with the following plant conditions:

Reactor power indication	IRM's inserted, on range 2 and slowly lowering
Drywell pressure	1.3 psig
Reactor pressure	800 psig
Reactor water level	Unknown
Six control rods	Between 02 and 08

Which one of the following completes the statements below?

The CRS ___(1)___ required to exit RSP.

The CRS will perform ___(2)___ of ATWS concurrently with RxFP.

A. (1) is
(2) all legs

B. (1) is
(2) ONLY the power leg

C. (1) is NOT
(2) all legs

D. (1) is NOT
(2) ONLY the power leg

Answer: B

K/A:
295015 Incomplete SCRAM
G2.4.5 Knowledge of the organization of the operating procedures network for normal, abnormal, and emergency evolutions. (CFR: 41.10 / 43.5 / 45.13)

RO/SRO Rating: 3.7/4.3

Pedigree: New

Objective: LOI-CLS-LP-300-E, Objective 13a
Given plant conditions and the Level/Power Control Procedure, determine if any of the following are appropriate or required: Reactor Flooding (LOCT)

Reference: None

Cog Level: High

Explanation: On a scram the RSP is entered, with 6 rods out beyond 02 the ATWS procedure is entered, (criteria is no more than 10 rods beyond position 02) and with level unknown the override in the ATWS procedure states to exit the P and L legs of ATWS and perform RxFP concurrently with the Q leg of the ATWS procedure.

Distractor Analysis:

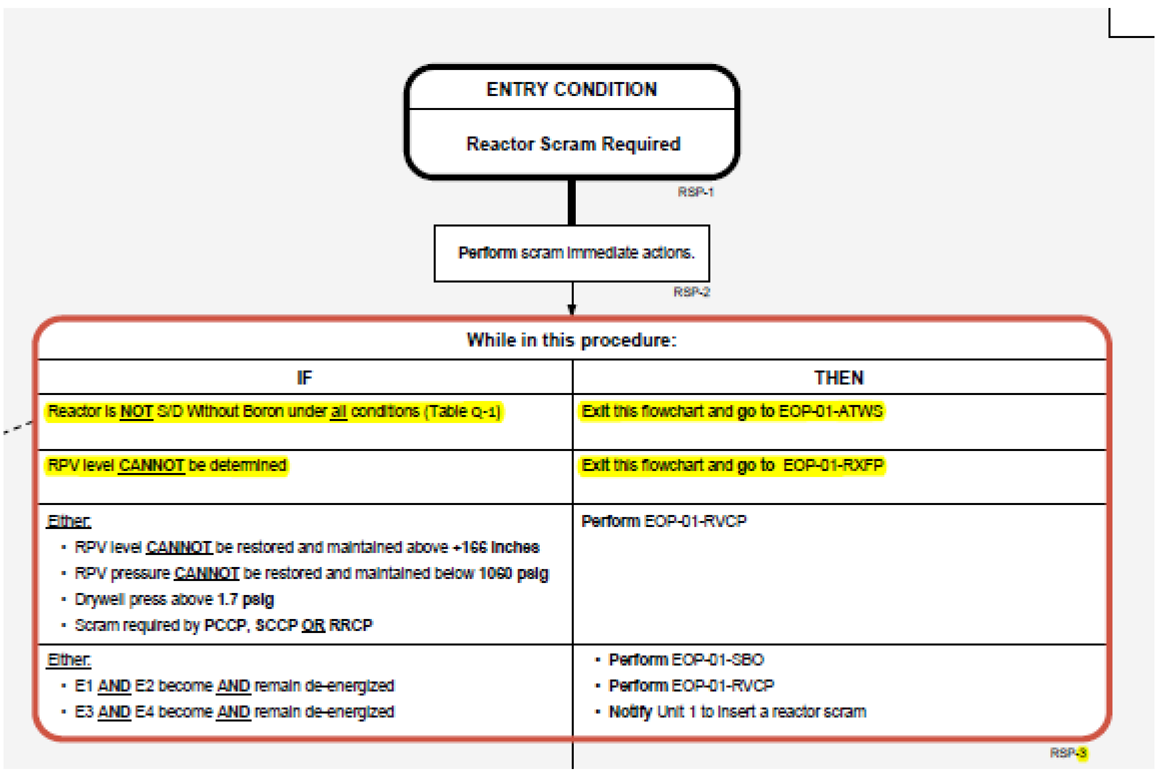
Choice A: Plausible because the RSP is exited and the student might think that the RxFP will perform actions for level so they would stay in the RVCP pressure leg.

Choice B: Correct Answer, see explanation.

Choice C: Plausible because there is a reason to be in each of these procedures.

Choice D: Plausible because there is a reason to be in each of these procedures.

SRO Basis: Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations. [10 CFR 55.43(b)(5)]



START

ATWS-1

While in this procedure:

IF	THEN
RPV level __CANNOT__ be determined	Exit RC/L __AND__ RC/P flowpaths and **go to** EOP-01-RXFP
Emergency depressurization is __OR__ has been required	• **Proceed to** 6 • **Exit** RC/P flowpath and **go to** EOP-01-EDP
Reactor is S/D Without Boron under __all__ conditions (Table Q-1)	1. **Terminate** boron injection __NOT__ required by other EOPs 2. **Exit** this flowchart and **go to** EOP-01-RVCP

ATWS-2

86. S295016 1

Unit Two was operating at rated power when the following occurred:

Time	Event
0800	Fire/smoke in the Control Building is reported
0803	Main control room evacuation completed
0820	Fire is reported to be extinguished with visible damage in the cable spread area
0825	RSDP is staffed
0826	One SRV is open maintaining reactor pressure constant Torus Cooling is maintaining Torus temperature below HCTL

Which one of the following completes the statements below?

(Reference provided)

Reactor thermal power is currently approximately ___(1)___ .

IAW 0PEP-02.1, BNP Initial Emergency Actions, the highest EAL classification that is required for these conditions is ___(2)___.

A. (1) 6%
   (2) an Alert

B. (1) 6%
   (2) a Site Area Emergency

C. (1) 9%
   (2) an Alert

D. (1) 9%
   (2) a Site Area Emergency

Answer: B

K/A:
295016 Control Room Abandonment
G2.1.7 Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior, and instrument interpretation.
(CFR: 41.5 / 43.5 / 45.12 / 45.13)

RO/SRO Rating: 4.4/4/7

Pedigree: New

Objective: LOI-CLS-LP-301B, Objective 9
Given a hypothetical abnormal event and plant operating mode, use 0PEP-02.1 to properly classify or re-classify the event.

Reference: EAL flowchart PEP-02.1

Cog Level: High

Explanation: A SRV is ~830,000 Mlbs/hr steam flow. At 100% power steam flow is 12,781,000 Mlbs/hr. This makes each SRV ~6.5% power. Fire not extinguished in 15 minutes is a UE, visible damage is an Alert. Control Room Evacuation is an Alert. Not having RSDP capabilities in 15 minutes is a SAE.

Distractor Analysis:

Choice A: Plausible because ~6% is correct and the visible damage is an Alert, but remote shutdown not established in 15 minutes is a SAE.

Choice B: Correct Answer, see explanation

Choice C: Plausible because the examinee could conclude that the 11 SRVs are capable of handling 100% power which would make each SRV capable of ~9% and the visible damage is an Alert, but remote shutdown not established in 15 minutes is a SAE.

Choice D: Plausible because the examinee could conclude that the 11 SRVs are capable of handling 100% power which would make each SRV capable of ~9% and a SAE is correct.

SRO Basis: Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations. [10 CFR 55.43(b)(5)]

SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
None	HA2 Fire or explosion affecting the operability of plant safety systems required to establish or maintain safe shutdown 1 | 2 | 3 | 4 | 5 | DEF **HA2.1** Fire or explosion resulting in visible damage to any Table H-1 area containing safety systems or components or Control Room indication of degraded performance of those safety systems	HU2 Fire within the Protected Area not extinguished within 15 minutes of detection or explosion within Protected Area 1 | 2 | 3 | 4 | 5 | DEF **HU2.1** Fire not extinguished within 15 min. (Note 3) of control room notification or verification of a control room fire alarm in any Table H-1 or Table H-3 areas **HU2.2** Explosion within Protected Area boundary
None	HA3 Access to a vital area is prohibited due to toxic, corrosive, asphyxiant or flammable gases which jeopardize operation of operable equipment required to maintain safe operations or safely shutdown the reactor 1 | 2 | 3 | 4 | 5 | DEF **HA3.1** Access to any Table H-1 area is prohibited due to toxic, corrosive, asphyxiant, or flammable gases which jeopardize operation of systems required to maintain safe operations or safely shutdown the reactor (Note 4)	HU3 Release of toxic, corrosive, asphyxiant or flammable gases deemed detrimental to normal plant operations 1 | 2 | 3 | 4 | 5 | DEF **HU3.1** Toxic, corrosive, asphyxiant or flammable gases in amounts that have or could adversely affect normal plant operations **HU3.2** Recommendation by local, county or state officials to evacuate or shelter site personnel based on offsite event

HS5 Control Room evacuation has been initiated and plant control cannot be established 1 | 2 | 3 | 4 | 5 | DEF	HA5 Control Room evacuation has been initiated 1 | 2 | 3 | 4 | 5 | DEF	
HS5.1 Control Room evacuation has been initiated **AND** Control of the plant cannot be established within 15 min.	**HA5.1** Control Room evacuation has been initiated	None

ATTACHMENT 3
Page 1 of 1

Characteristics Of Injection Sources And Pressure Control Systems

- 1 Mlbm/hr is equivalent to approximately 2000 gpm (conversion factor of 500 lbm/hr per gpm)
- 1 Mlbm/hr feed flow/steam flow mismatch changes RPV level approximately 13 inches/min (assuming 150 gal/in in the 182 inches to 192 inches normal operating range)
- Post scram CRD flow maximization for one CRD pump, raises RPV level approximately 1 inch/min if neglecting inventory losses due to decay heat. The injection rate can vary by as much as 25 gpm between 700 psig and 1000 psig.
- Sustained injection flow rates of >1 Mlb/hr of cold water injection will result in a swell of approximately 2.5 times the observed RPV level rise due to the injection (e.g. if HPCI used at full flow to raise level from +160 inches to +180 inches, level will swell by approximately 50 inches).
- An SRV can pass steam flow equivalent to 6 – 7% power at 1000 psig.
- Opening an SRV with no injection can cause approximately 10 inches of swell

87. S295018 1

Following a LOCA on Unit One, with NO RBCCW pumps in service, the following peak Drywell air temperatures were obtained:

CAC-TR-778, Primary Containment Air Temp recorder	347°F @ 88 ft elevation
CAC-TR-4426, Torus and Drywell Temp Div I(II) recorders	347°F @ 23 ft elevation

Which one of the following completes the statements below for RBCCW pump restart IAW 1OP-21, Reactor Building Closed Cooling Water System Operating Procedure?

(Reference Provided)

Attachment ___(1)___ is used to determine when the RBCCW pumps may be started.

The RBCCW pumps may be restarted ___(2)___ after the peak local temperatures have cooled to ≤230°F.

A. (1) 5, RBCCW Pump Restart Determination Using CAC-TR-4426
   (2) 30 minutes

B. (1) 5, RBCCW Pump Restart Determination Using CAC-TR-4426
   (2) 4 hours and 21 minutes

C. (1) 6, RBCCW Pump Restart Determination Using CAC-TR-778
   (2) 30 minutes

D. (1) 6, RBCCW Pump Restart Determination Using CAC-TR-778
   (2) 4 hours and 21 minutes

Answer: B

K/A:
295018 Partial or Complete Loss of Component Cooling Water
G2.1.23 Ability to perform specific system and integrated plant procedures during all modes of plant operation. CFR: 41.10 / 43.5 / 45.2 / 45.6)

RO/SRO Rating: 4.3/4/4

Pedigree: 2010 NRC Exam

Objective: CLS-LP-21 Objective. 12
Given plant conditions and OP-21, determine any limitations to restart of RBCCW pumps due to high drywell temperature.

Reference: 1OP-21 Attachment 5, 6 & 7

Cog Level: high

Explanation: NRC Generic Letter 96-06 pertains to issues that could affect containment integrity and equipment operability during accident conditions. In particular, the GL discusses cooling water systems serving containment air coolers (RBCCW supply to the drywell coolers) which may be exposed to the hydrodynamic effects of waterhammer during either a loss-of-coolant accident (LOCA), or a main steam line break (MSLB). These cooling water systems were not designed to withstand the effects of waterhammer. At BNP, the method used to address this concern is to establish a philosophy for restoring RBCCW flow to the drywell, in a controlled manner, after the RBCCW pumps have tripped or have been secured. While SBO is outside the scope of GL 96-06, and control of RBCCW pump restart due to this event is not required to resolve GL 96-06, prevention of inappropriate pump restart under this condition will contribute to preservation of system piping and containment integrity. At the beginning of the event, or in any case, prior to restoration of power, the pump control switches should be placed to OFF. This action will prevent potential waterhammer caused by an automatic pump restart when power is restored and worst-case local drywell temperatures have met or exceeded the threshold. If the RBCCW pumps can be restarted during or after this event, prior to drywell temperatures meeting or exceeding the threshold of 260 degrees F, then waterhammer will not be a concern. If the temperature threshold has been reached, then pump restart should be controlled to prevent damage to plant structures and equipment. If the pumps cannot be restarted, containment integrity will not be compromised since no credit is currently taken for RBCCW pump operation during this event.

The 4426 recorders are used unless they are inoperative, which makes Attachment 5 the correct attachment. Based on attachment 5, with the highest temperature being above the 29' elevation and >260'F, Table 1 on Attachment 6 is used. With temperatures in the range of 300 - 350°F the correct time allotment is 4 hours and 21 minutes.

Distractor Analysis:

Choice A: Plausible because Attachment 5 is correct but 30 minutes would be correct if Table 1 was used.

Choice B: Correct Answer, see explanation

Choice C: Plausible because attachment 5 would be correct if the data was not available from CAC-TR-4426. 30 minutes would be correct if Table 1 was used.

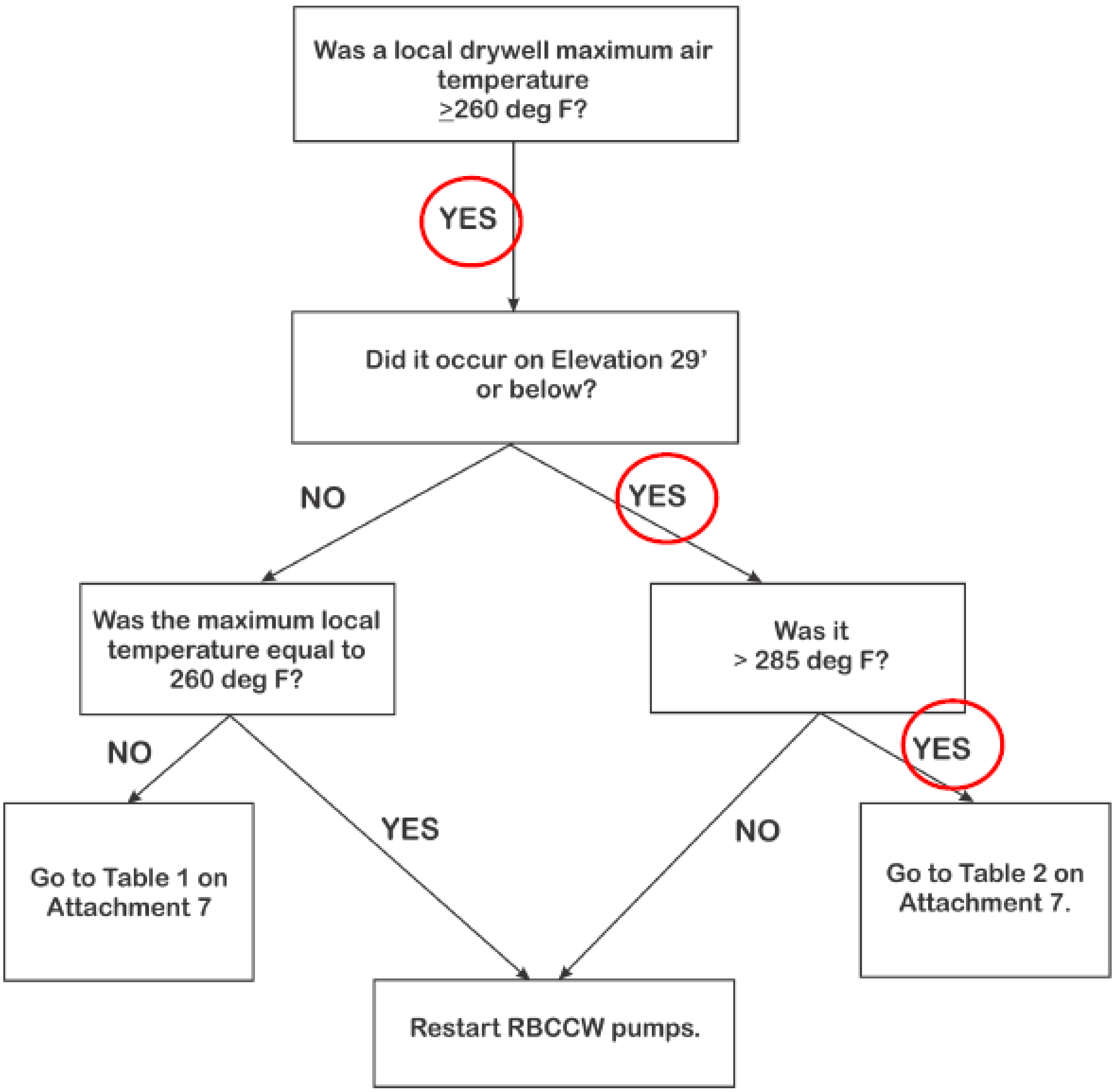
Choice D: Plausible because attachment 5 would be correct if the data was not available from CAC-TR-4426 and 4 hrs and 21 min is correct.

SRO Basis: Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations (10 CFR 55.43(b)(5))

ATTACHMENT 5
Page 1 of 1

RBCCW Pump Restart Determination Using CAC-TR-4426 [Torus And Drywell Temp Div I(II)]

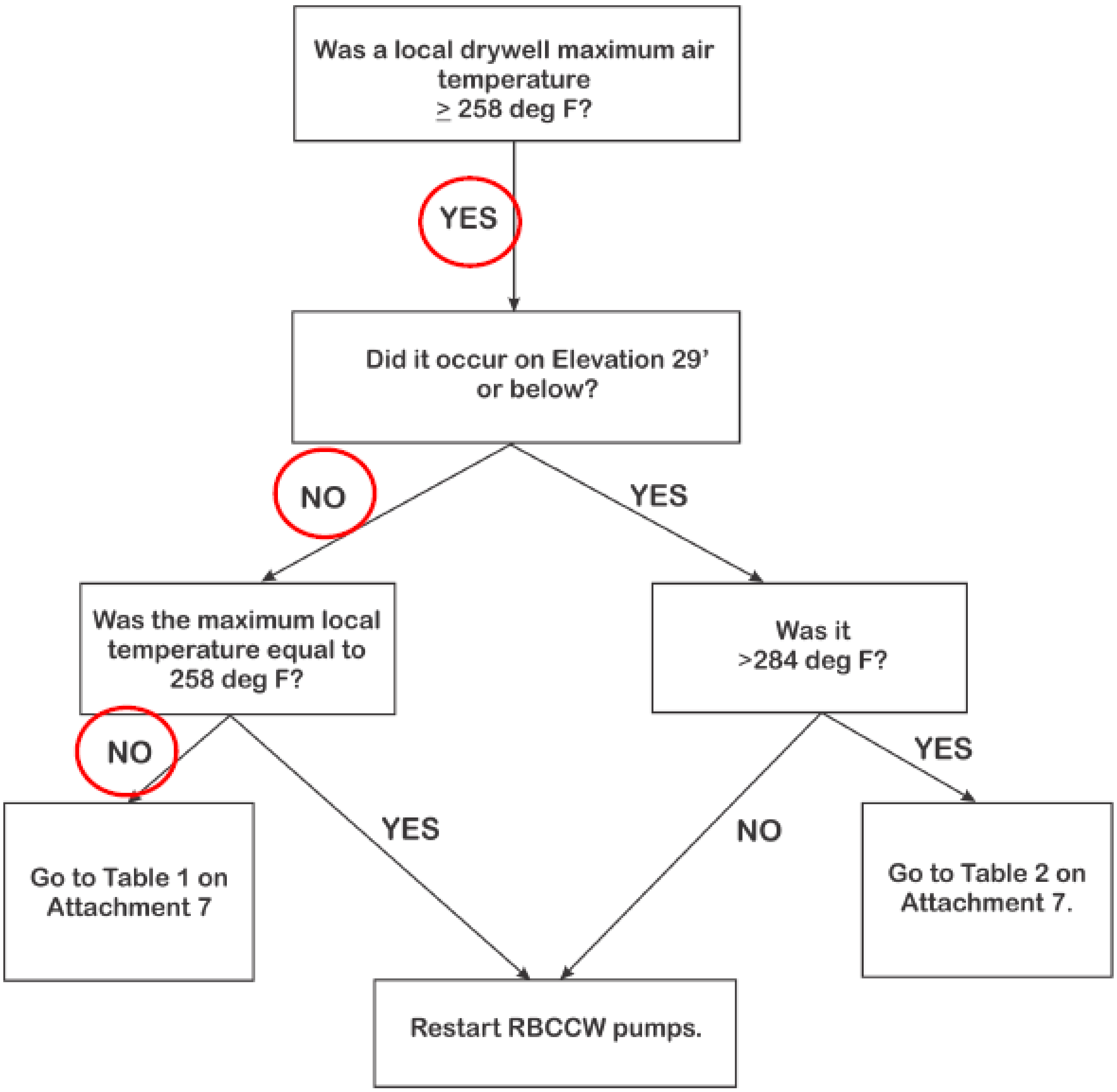
{8.1.2}



ATTACHMENT 6
Page 1 of 1

RBCCW Pump Restart Determination Using CAC-TR-778 (Primary Containment Air Temp)

{8.1.2}



ATTACHMENT 7
Page 1 of 1

Required Drywell Cooldown Time Prior to RBCCW Pump Restart

{8.1.2}

1. Using each peak local temperature identified in Attachment 5 or Attachment 6 requiring evaluation per this attachment, **determine** required Drywell cooldown time.

2. **WHEN** the peak local temperature for each temperature indicator has been less than or equal to 230°F for the required cooldown time,
THEN go to Section 6.3.7 Step 5.

TABLE 1

Peak Temp	>450°F	>400°F and ≤450°F	>350°F and ≤400°F	>300°F and ≤350°F	CAC-TR-4426 [Torus And Drywell Temp Div I(II)]: >260°F and ≤300°F CAC-TR-778 (Primary Containment Air Temp): >258°F and ≤300°F
Cooldown	43 minutes	39 minutes	36 minutes	30 minutes	22 minutes

TABLE 2

Peak Temp	>450°F	>400°F and ≤450°F	>350°F and ≤400°F	>300°F and ≤350°F	CAC-TR-4426 [Torus And Drywell Temp Div I(II)]: >285°F and ≤300°F CAC-TR-778 (Primary Containment Air Temp): >284°F and ≤300°F
Cooldown	7 hr 4 min	6 hr 23 min	5 hr 30 min	4 hr 21 min	2 hr 27 min

6.3.7 Restarting RBCCW Pumps in RBCCW Mode with High Drywell Temperature (continued)

c. **Evaluate** each local drywell temperature indicator to determine if RBCCW pumps may be restarted as follows:

- **IF** CAC-TR-4426 [Torus And Drywell Temp Div I(II)] recorders were available, **THEN determine** when RBCCW pumps may be restarted per Attachment 5, RBCCW Pump Restart Determination Using CAC-TR-4426 [Torus And Drywell Temp Div I(II)]. .... _________

- **IF** CAC-TR-4426 [Torus And Drywell Temp Div I(II)] recorders were **NOT** available, **THEN determine** when RBCCW pumps may be restarted per Attachment 6, RBCCW Pump Restart Determination Using CAC-TR-778 (Primary Containment Air Temp). .... _________

5. **WHEN** directed by Attachment 7, Required Drywell Cooldown Time Prior to RBCCW Pump Restart to restart RBCCW pumps, **THEN perform** the following for the pump to be started in RBCCW Mode:.. _________

88. S295021 1

Unit One is performing a shutdown with the following plant conditions:

Reactor mode switch	Shutdown
Reactor water level	195 inches
RCS temperature	200°F

A loss of Off-Site power occurs with all DG starting and loading.
Thirty minutes later, Shutdown Cooling has been returned to service with the following plant conditions:

Reactor water level	225 inches
RCS temperature	244°F

Which one of the following completes the statements below?

(Reference provided)

A MODE change ___(1)___ occurred.

The highest EAL classification for this event is an ___(2)___.

A. (1) has
   (2) Unusual Event

B. (1) has
   (2) Alert

C. (1) has NOT
   (2) Unusual Event

D. (1) has NOT
   (2) Alert

Answer: B

K/A:
295021 Loss of Shutdown Cooling
AA2 Ability to determine and/or interpret the following as they apply to LOSS OF SHUTDOWN COOLING: (CFR: 41.10 / 43.5 / 45.13)
06 Reactor Pressure

RO/SRO Rating: 3.2/3/3

Pedigree: New

Objective: LOI-CLS-LP-301-B, Objective 9
Given a hypothetical abnormal event and plant operating mode, use 0PEP-02.1 to properly classify or re-classify the event.

Reference: EAL flowchart PEP-02.1, Steam Tables

Cog Level: High

Explanation: The student will have to determine reactor pressure using the steam tables to answer this question. The current conditions show the reactor in Mode 4, when temperature increases above 212°F then the reactor will be in Mode 3. Using the steam table reactor pressure would have risen above 10 psig during an unplanned event so the EAL call will be an alert (CA3.1). The power loss is an UE.

Distractor Analysis:

Choice A: Plausible because a mode change has occurred but an Unusual Event is not the highest classification.

Choice B: Correct Answer, see explanation

Choice C: Plausible because a mode change has occurred and an Unusual Event does apply but is not the highest classification.

Choice D: Plausible because a mode change did occur and the Alert is correct.

SRO Basis: Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations. [10 CFR 55.43(b)(5)]

Table 1.1-1 (page 1 of 1)
MODES

MODE	TITLE	REACTOR MODE SWITCH POSITION	AVERAGE REACTOR COOLANT TEMPERATURE (°F)
1	Power Operation	Run	NA
2	Startup	Refuel[a] or Startup/Hot Standby	NA
3	Hot Shutdown[a]	Shutdown	> 212
4	Cold Shutdown[a]	Shutdown	≤ 212
5	Refueling[b]	Shutdown or Refuel	NA

(a) All reactor vessel head closure bolts fully tensioned.

(b) One or more reactor vessel head closure bolts less than fully tensioned.

ALERT	UNUSUAL EVENT
CA1 Loss of all offsite and all onsite AC power to emergency buses for 15 minutes or longer 4 | 5 | DEF	CU1 AC power capability to emergency buses reduced to a single power source for 15 minutes or longer such that any additional single failure would result in loss of all AC power to emergency buses 4 | 5
CA1.1 Loss of **all** offsite and **all** onsite AC power to Emergency 4 KV Buses E1(E3) and E2(E4) for ≥ 15 min. (Note 3)	**CU1.1** AC power capability to Emergency 4 KV Buses E1(E3) and E2(E4) reduced to a single power source for ≥ 15 min. (Note 3) such that **any** additional single failure would result in loss of **all** AC power to Emergency Buses
CA3 Inability to maintain plant in cold shutdown 4 | 5	CU3 Unplanned loss of decay heat removal capability with irradiated fuel in the RPV 4 | 5
CA3.1 **Any** unplanned event results in RCS temperature > 212°F for > Table C-3 duration **OR** Unplanned RPV pressure increase > 10 psig due to loss of RCS cooling	**CU3.1** **Any** unplanned event resulting in RCS temperature > 212°F due to loss of decay heat removal capability **CU3.2** Loss of **all** RCS temperature and RPV level indication for ≥ 15 min. (Note 3)

89. S295022 1

Unit One is operating at rated power when the following observations are made:

0800	1A CRD pump trips (1B under clearance)
0810	Annunciator A-07 (6-1), *CRD Accum Lo Press/Hi Level,* is received with HCU 12-19 amber light illuminated on the full core display

Which one of the following completes the statements below?

When the local panel pushbutton is depressed, if the local alarm light remains lit, the annunciator is due to ___(1)___.

If the Reactor Building AO reports an accumulator pressure of 925 psig, then the control rod scram accumulator is ___(2)___.

A. (1) low pressure
   (2) Operable

B. (1) low pressure
   (2) Inoperable

C. (1) high water level
   (2) Operable

D. (1) high water level
   (2) Inoperable

Answer: B

K/A:
295022 Loss of CRD Pumps
AA2. Ability to determine and/or interpret the following as they apply to LOSS OF CRD PUMPS: (CFR: 41.10 / 43.5 / 45.13)
01 Accumulator pressure

RO/SRO Rating: 3.5/3.6

Pedigree: New

Objective: LOI-CLS-LP-008, Objective 10
Given plant conditions, determine proper operator actions if no CRD pumps are operating. (LOCT)

Reference: None

Cog Level: High

Explanation: The local alarm panel red light for the HCU will illuminate for the particular HCU, if the light is depressed and the light remains lit this determines the cause of the alarm to be from a low pressure condition, If the lit extinguishes then this determines that the alarm is from a hi water condition.

S.R. 3.1.5.1 defines operable control rod scam accumulator pressure of $\geq$940 psig.

The AOP contains the required action to ensure charging water pressure is equal to or greater than 940 psig if two or more amber lights on the full core display illuminate when reactor pressure is greater than 950 psig.

Distractor Analysis:

Choice A: Plausible because the first part is correct. Tech Spec actions are based on reactor pressure above or below 950 psig. The alarm setpoint for accumulator low pressure is 955 psig.

Choice B: Correct Answer, see explanation

Choice C: Plausible because the light extinguishing is for a high water condition and the alarm indicates either high water or low pressure. Tech Spec actions are based on reactor pressure above or below 950 psig. The alarm setpoint for accumulator low pressure is 955 psig.

Choice D: Plausible because the light extinguishing is for a high water condition and the alarm indicates either high water or low pressure. The second part is correct.

SRO Basis: Facility operating limitations in the TS and their bases. [10 CFR 55.43(b)(2)]

WHITE

6-1

CRD ACCUM LO PRESS/HI LEVEL **Page 1 of 1**

1.0 OPERATOR ACTIONS:

1.1 **CONFIRM** which CRD is causing the annunciator by observation of the amber light on the affected HCU on the Full Core Display.

1.2 **OBSERVE** Automatic Functions:

1.2.1 At the local CRD HCU Panel, the red indication light is ON for the affected HCU.

1.3 **PERFORM** Corrective Actions:

> **NOTE:** Accumulator pressure less than 940 psig will render the accumulator inoperable.
>
> **NOTE:** **IF** this annunciator is sealed in, **THEN** the other accumulator alarms will be masked, **AND** contingency plans should be made to monitor the other accumulator alarms at the discretion of the Unit CRS.

1.3.1 **DETERMINE** if alarm is due to low pressure or high water level in the HCU by depressing the lighted indication on the local HCU panel and observing the status of the light (light out indicates water).

4.2 Supplementary Actions (continued)

8c. **IF** reactor pressure is greater than or equal to 950 psig,
AND two or more HCU low pressure alarms (A-07 6-1,
confirmed by amber light on Full Core Display),
THEN ensure CRD charging header pressure is restored to
greater than or equal to 940 psig within 20 minutes. .... ☐

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.5.1	Verify each control rod scram accumulator pressure is $\geq$ 940 psig.	7 days

2.2 Low nitrogen charge pressure.

3.0 DEVICES: **SETPOINT:**

3.1	Level Switch C11-LDSH-129 (Each HCU)	60 cc
3.2	Pressure Switch C11--PSL-130 (Each HCU)	955 psig

4.0 REFERENCES:

4.1 LL-93064 - 98

4.2 Technical Specification 3.1.5

4.3 1OP-08, Control Rod Drive Hydraulic System Operating Procedure

90. S295023 1

A fuel bundle was dropped in the spent fuel pool and the following alarms are received:

UA-03 (3-7) *Area Rad Refuel Floor High*
UA-03 (4-5) *Process Rx Bldg Vent Rad Hi*
UA-03 (2-3) *Rx Bldg Roof Vent Rad High*
UA-03 (3-3) *Turb Bldg Vent Rad High*

Turbine Vent Rad levels have been determined to be above the ALERT level.

Which one of the following completes the statements below?

Secondary Containment ___(1)___automatically isolated.

Execute 0AOP-5.0, Radioactive Spills, High Radiation, and Airborne Activity, 0AOP-5.4, Radiological Release, and ___(2)___ RRCP, Radiological Release Control.

A. (1) has
(2) do NOT execute

B. (1) has
(2) concurrently execute

C. (1) has not
(2) do NOT execute

D. (1) has not
(2) concurrently execute

Answer: D

K/A:
295023 Refueling Accidents
2.4.11 Knowledge of abnormal condition procedures. (CFR: 41.10 / 43.5 / 45.13)

RO/SRO Rating: 4.0/4.2

Pedigree: New

Objective: LOI-CLS-LP-302-J, Objective 1
Given plant conditions, determine if the AOP-5.0 should be entered.

Reference: None

Cog Level: High

Explanation: All four of these alarms are symptoms for AOP entry. Entry into EOP RRCP requires an ALERT level per the EALs. This is a new requirement for entry into RRCP. Unlike 0AOP-14, when an entry condition exists for the EOP, you do not exit the AOP, instead it is completed concurrently with the EOP. Conditions do not exist for SCI (SBGT start, Group VI, and RBV isolation). CREV should be manually started but no auto start signal exists. This is SRO level because it is not just about entry conditions, but also covers the rules of execution for AOPs and EOPs which are different depending on the AOP as indicated above.

Distractor Analysis:

Choice A: Plausible because AOP-5.0 and 5.4 should be executed, but also EOP-RRCP should be entered. SCI signal does not exist.

Choice B: Plausible because if RB Vent Hi Hi was in this would be a correct answer. SCI signal does not exist. Second part is correct.

Choice C: First part is correct. EOP-RRCP should be entered, but AOP-5.0 and 5.4 should not be exited.

Choice D: Correct Answer, see explanation.

SRO Basis: Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations. [10 CFR 55.43(b)(5)]

1.0 PURPOSE

1. This procedure lists symptoms and provides operator actions to mitigate a radiological release.

2.0 SYMPTOMS

1. Possible Indications:

   - Rising radiation indicated on D12-R601 (SJAE Rad Monitor) recorder on Panel XU-3
   - Rising radiation indicated on D12-RR-R600B (Stack Rad Monitor) recorder on Panel XU-3
   - Rising radiation indicated on D12-R603 (Main Steam Line Rad Monitor) recorder on Panel XU-3
   - Rising radiation indicated on D12-R605 (Reactor Bldg Vent Rad Monitor) recorder on Panel XU-3
   - Rising radiation indicated on D12-R604 (Service Water & RBCCW Rad Monitors) recorder on Panel XU-3
   - Rising radiation indicated on D12-RR-R001A(B) (Radwaste Effluent Rad Monitor) recorder on Panel XU-3
   - Elevated reactor building roof vent release rate indicated on CAC-AR-1264 (Rx Bldg Roof Vent Mon Recorder) on Panel XU-55

2.0 SYMPTOMS (continued)

- UA-03 4-1, Process OG Timer Initiated
- UA-03 6-4, Process OG Vent Pipe Rad Hi
- UA-03 5-4, Process OG Vent Pipe Rad Hi Hi
- UA-23 2-6, Main Steam Line Rad Hi
- UA-23 3-6, Main Steam Line Rad Hi-Hi/Inop
- UA-03 2-3, Rx Bldg Roof Vent Rad High
- UA-03 4-5, Process Rx Bldg Vent Rad Hi
- UA-03 3-5, Process Rx Bldg Vent Rad Hi-Hi
- UA-03 3-3, Turb Bldg Vent Rad High
- UA-03 5-5, Service Wtr Effluent Rad High
- UA-03 2-8, Radwaste Effluent Rad Hi-Hi

1.0 PURPOSE

1. This procedure lists symptoms and automatic actions and provides immediate and supplementary operator actions to mitigate radioactive spills, high radiation, and airborne activity.

2.0 SYMPTOMS

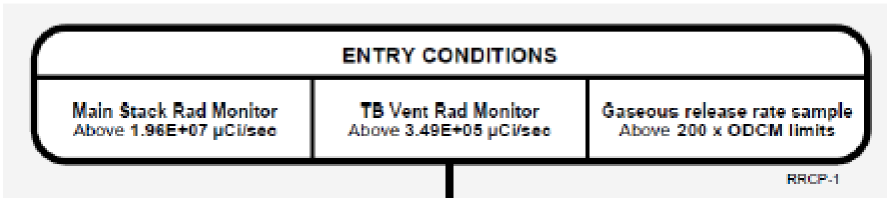
1. Possible indications:

    - Turbine Building WRGM indicates elevated (higher than expected or an unanticipated increase) activity
    - An unexplained or uncontrolled rise in area radiation, contamination, or airborne activity as determined by routine surveys and further actions are required to mitigate the effects
    - Report of spill or leak
    - Report of potential damage to new or spent fuel
    - Continuous Air Monitor (CAM) alarming

2. Possible annunciators:

    - UA-03 3-7, Area Rad Refuel Floor High
    - UA-03 4-7, Area Rad New Fuel Storage High
    - UA-03 4-5, Process Rx Bldg Vent Rad Hi
    - UA-03 3-5, Process Rx Bldg Vent Rad Hi-Hi
    - UA-03 3-3, Turb Bldg Vent Rad High

5.1 Step RRCP-1



The entry conditions for the RRCP guideline correspond to the gaseous Alert levels defined in the site Emergency Plan. These levels are sufficiently high that they are not expected to occur during normal plant operation but sufficiently low such that the condition does not pose an immediate threat to the health and safety of the public.

The Site Emergency Plan specifies Alert action levels for liquid as well as gaseous offsite radioactivity releases. However, it is not possible for a primary system (as the term is defined in the EOPs) to generate a liquid offsite radioactivity release. Since this guideline is based on a primary system discharging into an area outside the primary and secondary containments, the Alert entry condition need only include gaseous offsite radioactivity releases.

91. S295025 1

Unit One has been operating at rated power for the last 18 months.
A Loss of Off-site Power (LOOP) occurs and cannot be restored for 3 hours.

Which one of the following completes the statements below?

The bases for RVCP procedure Step RC/P-3, 'If any SRV is cycling, Then open SRVs until pressure drops', is to ___(1)___.

HPCI in pressure control ___(2)___ capable of providing sufficient steam flow to stabilize reactor pressure (within the first 10 minutes).

A. (1) conserve SRV accumulator pressure
(2) is

B. (1) conserve SRV accumulator pressure
(2) is NOT

C. (1) minimize dynamic loads/stresses imposed on the RPV
(2) is

D. (1) minimize dynamic loads/stresses imposed on the RPV
(2) is NOT

Answer: D

K/A:
295025 High Reactor Pressure
EA2 Ability to determine and/or interpret the following as they apply to HIGH REACTOR PRESSURE: (CFR: 41.10 / 43.5 / 45.13)
05 Decay Heat Generation

RO/SRO Rating: 3.4/3.6

Pedigree: new

Objective: LOI-CLS-LP-300-E, Objective 11
Given plant conditions, the Reactor Vessel Control Procedure, and which steps have been completed, determine the required operator actions. (LOCT)

Reference: None

Cog Level: high

Explanation: The bases for Step RC/P-3 is to minimize Significant dynamic loads/stresses imposed on the RPV, on SRV tail pipes and supporting structures, and on primary containment structures.

The amount of decay heat added depends on the power history of the reactor and the amount of time since the reactor was shut down. The number of fissions that have occurred determines the number of fission fragments in the core. Initial Decay Heat generation is equivalent to approximately 7% (beyond the capacity of HPCI) of the equilibrium power prior to the scram. 1 hour following the scram, Decay Heat generation is equivalent to approximately 1% power (within the capacity of HPCI and maybe RCIC).

Distractor Analysis:

Choice A: Plausible because conserving accumulator pressure is a subsequent step bases. HPCI can provide pressure control but not initially.

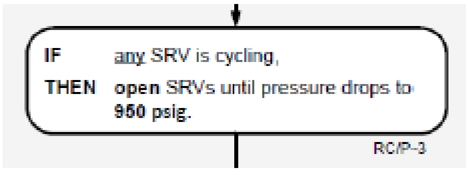
Choice B: Plausible because conserving accumulator pressure is a subsequent step bases and second part is correct.

Choice C: Plausible because this is the correct bases and HPCI can provide pressure control, but not initially.

Choice D: Correct Answer, see explanation

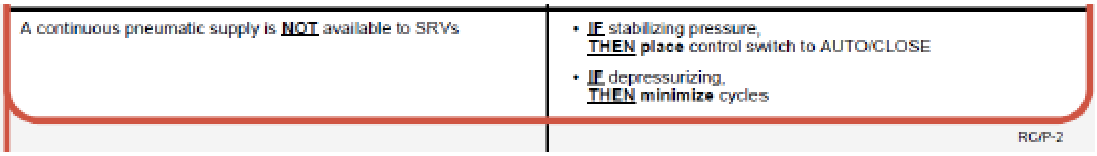
SRO Basis: Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations [10 CFR 55.43(b)(5)]
This question measures the SRO's assessment of high RPV pressure conditions and the knowledge of EOP symptom based steps used to prevent SRV cycling under high RPV pressure conditions.

5.16 Step RC/P-3



SRV cycling is defined as multiple, closely sequenced valve actuations with valve opening being initiated in response to RPV pressure increasing to/above the lift setpoint, and valve closure being governed by RPV pressure decreasing to/below the reset point. Potential severe consequences associated with SRV cycling require prompt manual action to reduce RPV pressure below the SRV lift setpoint. Actions to prevent SRV cycling will minimize:

- Significant dynamic loads/stresses imposed on the RPV, on SRV tail pipes and supporting structures, and on primary containment structures.
- Fluctuating RPV level (shrink occurring when the valves close as RPV pressure increases and swell occurring when the valves open as RPV pressure rapidly decreases).
- Repeated challenges to SRV operability (potential failure of a valve to open on demand or to close once it has opened).



If SRVs are being used to depressurize (i.e. cooldown) the RPV, sustained SRV opening conserves accumulator pressure. Reducing the number of cycles on SRVs prolongs SRV availability should more degraded conditions later require SRVs be opened for rapid depressurization of the RPV. The cooldown rate LCO of 100°F/hr is not allowed to be exceeded.

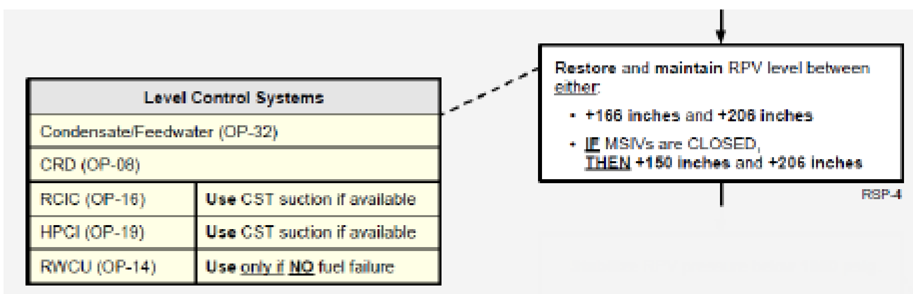
5.3.1 General Strategies for Effective Control (continued)

- With HPCI operating in pressure control, it is likely that SRV operation will be required for a short period of time after the Group 1 isolation.

- RCIC is designed to slowly restore RPV level following a Group 1 isolation. It is acceptable for RPV level to be below the +166 to +206 inch control band with RCIC injecting and slowly restoring RPV level.

- Use RWCU as necessary to lower RPV level and restore RPV level to the established control band.

5.4 Step RSP-4



The preferred strategy is to restore and maintain RPV level in a normal band of +170 inches to +200 inches using the listed systems. An allowed band of +166 inches to +206 inches is specified based on the low RPV level scram and the high RPV level turbine trips.

If the MSIVs are closed, RPV pressure will initially be controlled using SRVs. Due to "shrink" (drop in indicated level when SRVs are closed) RPV level will likely drop below +166 inches; therefore an alternate level band with a lower limit of +150 inches, is allowed. The widened RPV level band provides added operational flexibility while still assuring adequate core cooling through core submergence.

92. S295034 1

Unit Two is operating at rated power when the following annunciators are received:

UA-03 (4-5) *Process Rx Bldg Vent Rad High*
UA-03 (3-5) *Process Rx Bldg Vent Rad Hi-Hi*
UA-03 (2-7) *Area Rad Rx Bldg High*

Which one of the following completes the statements below?

(Reference provided)

The cause of these radiation alarms is due to a ___(1)___.

IAW 0OI-01.07, Notifications, this event meets the conditions for reportability to the NRC within ___(2)___ hours.

A. (1) RWCU line leak in the triangle room
(2) 4

B. (1) RWCU line leak in the triangle room
(2) 8

C. (1) RHR heat exchanger leak
(2) 4

D. (1) RHR heat exchanger leak
(2) 8

Answer: B

K/A:
295034 Secondary Containment Ventilation High Radiation
EA2 Ability to determine and/or interpret the following as they apply to SECONDARY CONTAINMENT VENTILATION HIGH RADIATION: (CFR: 41.10 / 43.5 / 45.13)
02 Cause of high radiation levels

RO/SRO Rating: 3.7/4.2

Pedigree: 10-2 NRC Exam
(modified second part to ask notifications requirement instead of procedure use)

Objective: CLS-LP-300-N Objective 19
Given plant conditions and 0EOP-04-RRCP, determine the following:
a. Release path (LOCT)
b. Required actions to be taken. (LOCT)
c. If a leak is on a primary system. (LOCT)
LOI-CLS-LP-201-D, Objective 12
Given plant conditions and an event, determine any applicable reporting requirements per OI-01.07, Notifications. (LOCT)

Reference: 0OI-01.07, Attachment 1

Cog Level: High

Explanation: Alarms coming in is indication of RB HVAC isolating, SBGT actuation. Damaged fuel may release a substantial amount of radioactive noble gases, halogens, and other fission products into the secondary containment, but this would not occur from a new fuel bundle. The RWCU system leak in the triangle room would be a HELB condition causing these alarms.
The actuation of SBGT/isolation of RBHVAC is an 8 hour notification. The release is within the Rx Bldg so no release is in progress (which would be a 4 hour notification).

Distractor Analysis:

Choice A: Plausible because RWCU is a primary system. If the student interprets it to be a release then 4 hour reportability would be correct.

Choice B: Correct Answer, see explanation.

Choice C: Plausible because an RHR leak would cause hi rad conditions if it was external to the heat exchanger. If the student interprets it to be a release then 4 hour reportability would be correct.

Choice D: Plausible because an RHR leak would cause hi rad conditions if it was external to the heat exchanger and an 8 hour notification is correct.

SRO Basis: Conditions and limitations in the facility license. [10 CFR 55.43(b)(1)]

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ATTACHMENT 1
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Reportability Evaluation Checklist

NOTE

- If the answer to any of the following questions is YES, the event is reportable within 4 hours.
- If all answers to the following questions are NO, the event is not reportable within 4 hours.

4 HOUR REPORTABILITY			
ITEM #	**YES**	**NO**	**DESCRIPTIVE QUESTION**

2.4			**NOTE** • Such an event may include an on-site fatality or an inadvertent release of radioactively contaminated materials. • Outside Government Agency notifications (e.g., North Carolina Wildlife Resource Commission) as a result of sea turtle takes resulting in injury or death that are determined to be causally related to BSEP operations are reportable to the NRC under 10 CFR 50.72(b)(2)(xi). Is the event a situation, as related to the health and safety of the public or on-site personnel, or protection of the environment, for which a news release is planned or notification to other government agencies has been or will be made? [10 CFR 50.72(b)(2)(xi)] [10 CFR 72.75(b)(2)]

Reportability Evaluation Checklist

8 HOUR REPORTABILITY			
ITEM #	**YES**	**NO**	**DESCRIPTIVE QUESTION**
3.3.2			General containment isolation signals affecting containment isolation valves in more than one system or multiple main steam isolation valves (MSIVs). • Main Steam Isolation. • Main Steam Line Drain Isolation. • HPCI Steam Line Isolation. • RCIC Steam Line Isolation. • RWCU Suction Isolation. • Primary Containment Isolation. • Secondary Containment Isolation. • SGTS Actuation. • Combustible Gas Control (CAD). [10 CFR 50.72(b)(3)(iv)(B)(2)]

93. S295037 1

Unit One was operating at rated power when the following sequence of events occurred:

0800	Loss of Off-Site power occurs DG's all auto start and tie onto their respective busses Manual Scram pushbuttons depressed Mode Switch placed in shutdown
0801	25 control rods not full in HPCI started for level control ARI initiated
0802	All Rods In reported A-01 (3-5)(4-5) *HPCI Isol Trip Sys A(B) Initiated* reported
0803	A-01 (5-4) *HPCI Valves Mtr Overload* reported
0804	E41-F002(F003), Steam Supply Inboard(Outboard) Isol Vlv, both red lights are lit and green lights are extinguished
0805	The following room temperatures are observed on ERFIS: NRHR 113°F SRHR 111°F Mini Stm Tnl 195°F 20 Ft 105°F 50 Ft 100°F
0806	Drywell Rad Monitors indicate 0.08×10^2 R/hr

Based on the information above, which one of the following identifies the highest required EAL classification IAW 0PEP-02.1, Brunswick Nuclear Plant Initial Emergency Actions?

(Reference provided)

A. Unusual Event

B. Alert

C. Site Area Emergency

D. General Emergency

Answer: C

K/A:
295037 SCRAM Condition Present and Reactor Power Above APRM Downscale or Unknown
EA2 Ability to determine and/or interpret the following as they apply to SCRAM CONDITION PRESENT AND REACTOR POWER ABOVE APRM DOWNSCALE OR UNKNOWN:
(CFR: 41.10 / 43.5 / 45.13)
07 Containment conditions/isolations

RO/SRO Rating: 4.0/4.2

Pedigree: New

Objective: LOI-CLS-LP-301-B, Objective 9
Given a hypothetical abnormal event and plant operating mode, use 0PEP-02.1 to properly classify or re-classify the event

Reference: 0PEP-02.1, 0EOP-01-NL Table 3-B

Cog Level: High

Explanation: The loss of offsite power equates to an Unusual event, the ATWS is an alert because ARI worked, the unisolable leak with the mini steam tunnel greater than max normal is a SAE, If DW RM are greater than 2000 then this would be a GE.

Distractor Analysis:

Choice A: Plausible because the loss of offsite power is a UE, but this is not the highest classification (SU1.1).

Choice B: Plausible because a failure to auto scram (LOOP closes MSIVs) and manual actions shutdown the reactor (ARI) is an Alert classification, but this is not the highest classification. (SA2.1)

Choice C: Correct Answer, see explanation

Choice D: Plausible because if the DW RM would be greater than 2000 R/hr then a GE would be declared. Student has to convert the given reading.

SRO Basis: Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations. [10 CFR 55.43(b)(5)]

PLANT AREA	PLANT LOCATION DESCRIPTION	MAX NORM OPERATING VALUE (°F)	MAX SAFE OPERATING VALUE (°F)	AUTO GROUP ISOLATION
N CORE SPRAY	N CORE SPRAY ROOM	120	175	N/A
S CORE SPRAY	S CORE SPRAY ROOM	120	175	N/A
RWCU	PMP ROOM A PMP ROOM B HX ROOM	140	225	3
N RHR	N RHR EQUIP ROOM	175	295	N/A
S RHR	S RHR EQUIP ROOM RCIC EQUIP ROOM	175 165	295 295	N/A 5
HPCI	HPCI EQUIP ROOM	165	165	4
STEAM TUNNEL	RCIC STM TUNNEL HPCI STM TUNNEL	190 190	295 295	5 4
20 FT	2O FT NORTH 20 FT SOUTH	140 140	200 200	N/A N/A
50 FT	50 FT NW 50 FT SE	140 140	200 200	N/A N/A
REACTOR BLDG	MULTIPLE AREAS ANNUN. A-02 5-7	ALARM SETPOINT	N/A	3, 4, AND/OR 5
REACTOR BLDG	MSIV PIT ANNUN. A-06 6-7	ALARM SETPOINT	N/A	1

ALERT	UNUSUAL EVENT
SA1 AC power capability to emergency buses reduced to a single power source for 15 minutes or longer such that **any** additional single failure would result in loss of **all** AC power to emergency buses 1 2 3	SU1 Loss of **all** offsite AC power to emergency buses for 15 minutes or longer 1 2 3
SA1.1 AC power capability to Emergency 4 KV Buses E1(E3) and E2(E4) reduced to a single power source for $\geq$ 15 min. (Note 3) such that **any** additional single failure would result in loss of **all** AC power to emergency buses	**SU1.1** Loss of **all** offsite AC power to Emergency 4 KV Buses E1(E3) and E2(E4) for $\geq$ 15 min. (Note 3)
SA2 Automatic Scram fails to shutdown the reactor and the manual actions taken from the reactor control console are successful in shutting down the reactor. 1 2	SU2 Inadvertent criticality 3
SA2.1 Automatic scram fails to reduce reactor power < 2% (APRM downscale) **AND** Manual scram actions taken at the reactor control console (Manual PBs, Mode Switch, ARI) successfully shutdown the reactor as indicated by reactor power < 2% (APRM downscale)	**SU2.1** **Any** unplanned sustained positive period observed on nuclear instrumentation

GENERAL EMERGENCY	SITE AREA EMERGENCY
FG1.1 1 2 3	FS1.1 1 2 3
Loss of any two barriers **AND** Loss or potential loss of third barrier (Table F-1)	Loss or potential loss of any two barriers (Table F-1)

Reactor Coolant System Barrier		
Loss	**Potential Loss**	
3. Release pathway exists outside primary containment resulting from isolation failure in any of the following (excluding normal process system flowpaths from an unisolable system): - Main steam line - HPCI steam line - RCIC steam line - RWCU - Feedwater 4. Emergency Depressurization is required	1. RCS leakage > 50 gpm inside the drywell 2. Unisolable primary system discharge outside primary containment as indicated by Secondary Containment area radiation or temperature above any Maximum Normal Operating Limit (0EOP-03-SCCP Tables 3, 1)	3. Failure of any valve in any one line to close **AND** Direct release pathway to the environment outside PC exists after PC isolation signal (manual or automatic) 4. Intentional PC venting per EOPs 5. Unisolable primary system discharge outside primary containment as indicated by Secondary Containment area radiation or temperature above any Maximum Safe Operating Limit (0EOP-03-SCCP Tables 3,1)

94. SG2.1.26 1

Local conditions at a valve requiring independent verification are as follows:

Area Temperature:	110°F
Oxygen Content:	16.5%
Radiation Level:	250 mr/hr
Location/Elevation:	Valve is eight feet overhead; accessible via installed ladder. Independent verification is expected to take two minutes.

Which one of the following identifies the criteria that will allow a waiver of the independent verification requirements?

A. Excessive radiation exposure.

B. Area temperature is too high.

C. Valve is too high above floor level.

D. Hazards potentially dangerous to health are present.

Answer: D

K/A:
G2.1.26 Knowledge of industrial safety procedures (such as rotating equipment, electrical, high temperature, high pressure, caustic, chlorine, oxygen and hydrogen). (CFR: 41.10 / 45.12)

RO/SRO Rating: 3.4/3.6

Pedigree: Bank

Objective: LOI-CLS-LP-201-C, Objective 12b
Describe the following regarding OPS-NGGC-1303, Verification Practices: Exemptions from Independent Verification.

Reference: None

Cog Level: Fund

Explanation: The oxygen levels are below normal which would be a hazard dangerous to health. Rad exposure would be 250 mr/hr divided by 60 min = 4.16 mr/min in which the job takes 2 minutes thus 8.33 mr, which would be within the limits allowed. Temperature is slightly less than the exemption requirement. The valve is accessible via an installed ladder and fall protection would be required at this height.

Distractor Analysis:

Choice A: Plausible because high radiation is a reason for not performing the independent verification.

Choice B: Plausible because high temperature is a reason for not performing the independent verification.

Choice C: Plausible because if the ladder was not installed this would be correct.

Choice D: Correct Answer, see explanation

SRO Basis: Facility licensee procedures required to obtain authority for design and operating changes in the facility. [10 CFR 55.43(b)(3)]

9.5 Exceptions to Independent Verification

I
Information
Use

9.5.3 INDEPENDENT VERIFICATION requirements may be waived if:

1. Excessive radiation exposures would result. As a guideline, an exposure of greater than 10 mrem to conduct the INDEPENDENT/CONCURRENT VERIFICATION would be considered excessive. Individual situations should be determined on a case-by-case basis by the respective supervisor. In these situations, an alternate means such as FUNCTIONAL VERIFICATION not involving radiation exposure (such as observing process parameters) should be utilized.

2. Entry into any area where personnel safety is compromised or jeopardized due to the presence of extreme temperatures (greater than 120°F), or other hazards potentially dangerous to health are present.

3. Manipulated equipment have required positions controlled by valve and equipment lineup sheets and current plant operational conditions do not require the system to be operable. In these situations, prior to the time operability is required, valve and equipment lineup check sheets with INDEPENDENT or CONCURRENT VERIFICATION shall be completed.

4. Waivers for the performance of INDEPENDENT VERIFICATION shall not be made without Supervisory approval. Such approval should be annotated in the Notes, Comments or appropriate section of the controlling document.

95. SG2.1.42 1

Refueling is being performed per 0FH-11. 0FH-11 prohibits control rod withdrawal during the core load sequence until a neutronic bridge is established.

Which one of the following completes the statement below to meet the core loading sequence to establish a neutronic bridge as described in 0FH-11?

Four fuel bundles are loaded around ___(1)___, then fuel is loaded in all fuel cells in a line between ___(2)___.

A. (1) SRMs A and D ONLY
   (2) SRMs A and D.

B. (1) SRMs B and D ONLY
   (2) SRMs B and D.

C. (1) each of the four SRMs
   (2) SRMs A and D.

D. (1) each of the four SRMs
   (2) SRMs B and D

Answer: D

K/A:
G2.1.42 Knowledge of new and spent fuel movement procedures. (CFR: 41.10 / 43.7 / 45.13)

RO/SRO Rating: 2.5/3.4

Pedigree: Bank

Pedigree: 2010-1 NRC Exam

Objective: LOI-CLS-LP-305-E, Objective 19a
State the purpose of the following fuel handling procedures: FH-11, Refueling

Reference: None

Cog Level: Fund

Explanation: Definition in 0FH-11. See notes section.

Distractor Analysis:

Choice A: Plausible because loading fuel around 2 SRMs and a line between them would establish a neutron bridge (between those 2 SRMs) but not IAW 0FH-11 and A&D are adjacent.

Choice B: Plausible because loading fuel around 2 SRMs and a line between them would establish a neutron bridge (between those 2 SRMs) but not IAW 0FH-11 B&D would be on opposite sides of the core and the line of loaded fuel cells would intersect the center, but part 1 does not satisfy the definition.

Choice C: Plausible because loading fuel around all SRMs is correct but A&D are adjacent.

Choice D: Correct Answer, see explanation.

SRO Basis: Fuel handling facilities and procedures. [10 CFR 55.43(b)(7)]

REFUELING	0FH-11
	Rev. 103
	Page 8 of 61

3.0 PRECAUTIONS AND LIMITATIONS (continued)

22. To help ensure that an unmonitored criticality will **NOT** occur, control rod withdrawal is **NOT** allowed during the core reload sequence until after the neutronic bridge is established. The neutronic bridge ensures that two SRMs are neutronically coupled, thus monitoring the loaded area of the core. The reload sequence has three basic steps. Four fuel bundles are loaded around each of the four SRMs, the neutronic bridge is established and a spiral reload of the other fuel bundles completes the sequence. The neutronic bridge is established by loading fuel in all fuel cells in a line between two SRMs. These SRMs must be on opposite sides of the core and the line of loaded fuel cells must intersect the center of the core. .... ☐

96. SG2.2.17 1

Which one of the following completes the statements below IAW AD-WC-ALL-0200, On-Line Work Management?

The work week schedule is locked/frozen at the ___(1)___ Schedule Freeze Meeting.

Any work added after schedule freeze (not performed by the FIN team) is treated as ___(2)___ work.

A. (1) T-3
   (2) Critical Activity

B. (1) T-3
   (2) Emergent

C. (1) T-10
   (2) Critical Activity

D. (1) T-10
   (2) Emergent

Answer: B

K/A:
G2.2.17 Knowledge of the process for managing maintenance activities during power operations, such as risk assessments, work prioritization, and coordination with the transmission system operator. (CFR: 41.10 / 43.5 / 45.13)

RO/SRO Rating: 2.6/3.8

Pedigree: Bank (Browns Ferry)

Objective: LOI-CLS-LP-201-E, Objective 1
Describe the Work Management process from initiation of a W/R to the filing of completed documents in the Vault, including the following per ADM-NGGC-0104, 0AP- 025, and AD-WC-ALL-0200: d. Determine the requirements for authorizing on-line system outages (SRO Only)

Reference: None

Cog Level: Fundamental

Explanation: The schedule freeze occurs at T-3 weeks. The Scope freeze is at T-10 weeks. Any work added is treated as emergent work. Critical Activity work represents a substantial challenge to Nuclear, Operational, Industrial, Rad, or environmental risk (defined in AD-WC-ALL-0410).

Distractor Analysis:

Choice A: Plausible because T-3 is correct and a critical activity is a definition in the work control procedures.

Choice B: Correct Answer, see explanation

Choice C: Plausible because T-10 is the scope freeze time and a critical activity is a definition in the work control procedures.

Choice D: Plausible because T-10 is the scope freeze time and it is an emergent work.

SRO Basis: Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations. [10 CFR 55.43(b)(5)]
Operations involvement with respect to reviewing preliminary schedules in the Work Control Center is an SRO function.

Schedule Freeze: A predefined point in the process at which the work week schedule is locked/frozen. Changes to the locked/frozen schedule undergo a process that includes documentation and required signatures needed to control the changes to the schedule for the work week.

Scope Freeze: A predefined point in the process at which the work week scope is locked/frozen. Changes to the locked/frozen scope undergo a process that includes documentation and required signatures needed to control the addition and deletion of scope from a given work week.

Emergent Work: Any work added after schedule freeze **NOT** performed by the FIN/SPOC team.

T-10 Week

a. The T-10 schedule scope freeze/constraints review meeting is chaired by the Cycle Scheduler or Intermediate WWM using Attachment 5, T-10 Scope Freeze/Restraint Meeting Agenda.

b. The purpose of the T-10 review is to finalize the scope of work for the designated work week.

T-3 Week

a. The WWM chairs the T-3 Schedule Freeze Meeting using Attachment 7, T-3 Schedule Freeze Meeting Agenda to conduct a supervisory level review of the proposed schedule.

b. The purpose of the T-3 meeting is to perform a final review of the work week schedule to identify any potential schedule execution issues and to provide commitments from all groups to support the schedule.

97. SG2.2.22 1

Unit One is operating at 88% power with the following conditions:

Jet Pump Flow Loop A (B21-R611A)	29 Mlbs/hr
Jet Pump Flow Loop B (B21-R611B)	33 Mlbs/hr
Total Core Flow (U1CPWTCF)	62 Mlbs/hr

IAW Technical Specifications, which one of the following completes the statements below?

The current Jet Pump Flow Mismatch ____(1)____ within limits.

When Jet Pump Flows are not matched within limits, the loop with the ___(2)____ must be considered not in operation.

A. (1) is
(2) lower flow

B. (1) is
(2) higher flow

C. (1) is NOT
(2) lower flow

D. (1) is NOT
(2) higher flow

Answer: C

K/A:
G2.2.22 Knowledge of limiting conditions for operations and safety limits. (CFR: 41.5 / 43.2 / 45.2)

RO/SRO Rating: 4.0/4.7

Pedigree: 10-1 NRC Exam

Objective: LOI-CLS-LP-002, Objective 34
Given plant conditions and Technical Specifications, including the Bases, TRM, ODCM, and COLR determine the required action(s) to be taken in accordance with Technical Specifications associated with the Reactor Recirculation System. (SRO/STA only)

Reference: None

Cog Level: High

Explanation: Two recirculation loops are normally required to be in operation with their flows matched within the limits specified in SR 3.4.1.1 to ensure that that during a LOCA caused by a break of the piping of one recirculation loop the assumptions of the LOCA analysis are satisfied. Jet pump loop flow mismatch should be maintained within the following limits:

- jet pump loop flows within 10% (maximum indicated difference <u>7.5 $x10^6$ lbs/hr</u>) with total core flow less than <u>58 $x10^6$ lbs/hr</u>
- jet pump loop flows within 5% (maximum indicated difference 3.5 $x10^6$ lbs/hr) with total core flow greater than or equal to 58 $x10^6$ lbs/hr

Distractor Analysis:

Choice A: Plausible because flow mismatch is within limits for lower reactor power level.

Choice B: Plausible because flow mismatch is within limits for lower reactor power level and because the belief that the higher flow loop will experience excessive vibration could cause them to select the "higher flow" response

Choice C: Correct Answer, see explanation

Choice D: Plausible because the belief that the higher flow loop will experience excessive vibration could cause them to select the "higher flow" response

SRO Basis: Facility operating limitations in the TS and their bases. [10 CFR 55.43(b)(2)]

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.1 Recirculation Loops Operating

LCO 3.4.1 Two recirculation loops with matched flows shall be in operation,

<u>OR</u>

One recirculation loop may be in operation provided the following limits are applied when the associated LCO is applicable:

a. LCO 3.2.1, "AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)," single loop operation limits specified in the COLR;

b. LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)," single loop operation limits specified in the COLR;

c. LCO 3.2.3, "LINEAR HEAT GENERATION RATE (LHGR)," single loop operation limits specified in the COLR; and

d. LCO 3.3.1.1, "Reactor Protection System (RPS) Instrumentation," Function 2.b (Average Power Range Monitors Simulated Thermal Power—High), Allowable Value of Table 3.3.1.1-1 is reset for single loop operation.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Requirements of the LCO not met.	A.1 Satisfy the requirements of the LCO.	6 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Required Action and associated Completion Time of Condition A not met. OR No recirculation loops in operation.	B.1 Be in MODE 3.	12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.1.1	-------------------------------NOTE---------------------------- Not required to be performed until 24 hours after both recirculation loops are in operation. --- Verify recirculation loop jet pump flow mismatch with both recirculation loops in operation: a. ≤ 10% of rated core flow when operating at < 75% of rated core flow; and b. ≤ 5% of rated core flow when operating at ≥ 75% of rated core flow.	24 hours

BASES

APPLICABLE SAFETY ANALYSES (continued)	For AREVA fuel, the COLR presents single loop operation APLHGR limits in the form of a multiplier that is applied to the two loop operation APLHGR limits. The transient analyses of Chapter 15 of the UFSAR have also been evaluated for single recirculation loop operation. The evaluation concludes that results of the transient analyses are not significantly affected by the single recirculation loop operation. There is, however, an impact on the fuel cladding integrity SL since some of the uncertainties for the parameters used in the critical power determination are higher in single loop operation. The net result is an increase in the MCPR operating limit. During single recirculation loop operation, modification to the Reactor Protection System (RPS) average power range monitor (APRM) Simulated Thermal Power—High Allowable Value is required to account for the different analyzed limits between two-recirculation drive flow loop operation and operation with only one loop. The APRM channel subtracts the ΔW value from the measured recirculation drive flow to effectively shift the limits and uses the adjusted recirculation drive flow value to determine the APRM Simulated Thermal Power—High Function trip setpoint. Recirculation loops operating satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii) (Ref. 4).
LCO	Two recirculation loops are normally required to be in operation with their flows matched within the limits specified in SR 3.4.1.1 to ensure that during a LOCA caused by a break of the piping of one recirculation loop the assumptions of the LOCA analysis are satisfied. Alternately, with only one recirculation loop in operation, modifications to the required APLHGR limits (LCO 3.2.1, "AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)"), MCPR limits (LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)"), LHGR limits (LCO 3.2.3, "LINEAR HEAT GENERATION RATE (LHGR)"), and APRM Simulated Thermal Power—High Allowable Value (LCO 3.3.1.1), as applicable, must be applied to allow continued operation. The COLR defines adjustments or modifications required for the APLHGR, MCPR, and LHGR limits for the current operating cycle.

(continued)

BASES

APPLICABILITY	In MODES 1 and 2, requirements for operation of the Reactor Coolant Recirculation System are necessary since there is considerable energy in the reactor core and the limiting design basis transients and accidents are assumed to occur. In MODES 3, 4, and 5, the consequences of an accident are reduced and the coastdown characteristics of the recirculation loops are not important.
ACTIONS	A.1 With the requirements of the LCO not met, the recirculation loops must be restored to operation with matched flows within 6 hours. A recirculation loop is considered not in operation when the pump in that loop is idle or when the mismatch between total jet pump flows of the two loops is greater than the required limits. The loop with the lower flow must be considered not in operation. Should a LOCA occur with one recirculation loop not in operation, the core flow coastdown and resultant core response may not be bounded by the LOCA analyses. Therefore, only a limited time is allowed to restore the inoperable loop to operating status. Alternatively, if the single loop requirements of the LCO are applied to operating limits and RPS setpoints, as applicable, operation with only one recirculation loop would satisfy the requirements of the LCO and the initial conditions of the accident sequence. The 6 hour Completion Time is based on the low probability of an accident occurring during this time period, on a reasonable time to complete the Required Action (i.e., reset the applicable limits or setpoints for single recirculation loop operation), and on frequent core monitoring by operators allowing abrupt changes in core flow conditions to be quickly detected. This Required Action does not require tripping the recirculation pump in the lowest flow loop when the mismatch between the total jet pump flows of the two loops is greater than the required limits. However, in cases where large flow mismatches occur, low flow or reverse flow can occur in the low flow loop jet pumps, causing vibration of the jet pumps. If zero or reverse flow is detected, the condition should be alleviated by changing pump speeds to re-establish forward flow.

(continued)

BASES

ACTIONS (continued)

B.1

With no recirculation loops in operation or the Required Action and associated Completion Time of Condition A not met, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to MODE 3 within 12 hours. In this condition, the recirculation loops are not required to be operating because of the reduced severity of DBAs and minimal dependence on the recirculation loop coastdown characteristics. The allowed Completion Time of 12 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.4.1.1

This SR ensures the recirculation loops are within the allowable limits for mismatch. At low core flow (i.e., < 75% of rated core flow), the MCPR requirements provide larger margins to the fuel cladding integrity Safety Limit such that the potential adverse effect of early boiling transition during a LOCA is reduced. A larger flow mismatch can, therefore, be allowed when core flow is < 75% of rated core flow. The recirculation loop jet pump flow, as used in this Surveillance, is the summation of the flows from all of the jet pumps associated with a single recirculation loop.

The mismatch is measured in terms of the percent of rated core flow. If the flow mismatch exceeds the specified limits, the loop with the lower flow is considered not in operation. The SR is not required when both loops are not in operation since the mismatch limits are meaningless during single loop or natural circulation operation. The Surveillance must be performed within 24 hours after both loops are in operation. The 24 hour Frequency is consistent with the Surveillance Frequency for jet pump OPERABILITY verification and has been shown by operating experience to be adequate to detect off normal jet pump loop flows in a timely manner.

REFERENCES

1. UFSAR, Section 5.4.1.3.
2. UFSAR, Chapter 15.
3. NEDC-31776P, Brunswick Steam Electric Plant Units 1 and 2 Single Loop Operation, February 1990.
4. 10 CFR 50.36(c)(2)(ii).

98. SG2.3.6 1
The BSEP Radioactive Liquid Release Permit is being approved with the following step filled out on the permit:

9. **Confirm** the following instrumentation is OPERABLE:

a. Liquid Radwaste Radioactivity Monitor, 2-D12-RM-K604 .... CRS
....

b. Liquid Radwaste Effluent Flow Measurement Device,
2-G16-FIT-N057 .... INOP

Which one of the following completes the statements below?

The minimum required approval to commence any liquid release is(are) ___(1)___.

The Radioactive Liquid Release ___(2)___.

A. (1) Unit CRS ONLY
(2) can still occur if ODCM compensatory actions are implemented

B. (1) Unit CRS ONLY
(2) is NOT allowed unless 2-G16-FIT-N057 is operable

C. (1) Unit CRS and Shift Manager
(2) can still occur if ODCM compensatory actions are implemented

D. (1) Unit CRS and Shift Manager
(2) is NOT allowed unless 2-G16-FIT-N057 is operable

Answer: C

K/A:
G2.3.6 Ability to approve release permits. (CFR: 41.13 / 43.4 / 45.10)

RO/SRO Rating: 2.0/3.8

Pedigree: 2012 NRC Exam

Objective: LOI-CLS-LP-6.3, Objective 8a
State the actions required for the following conditions: Performing a release with the D12-RM-K604 Liquid Radwaste Effluent Radiation Monitor inoperable.

Reference: none

Cog Level: High

Explanation: If the effluent flow monitor is not available, ODCM 7.3.1 requires compensatory measure to estimate the flow rate at least once /4 hours during actual releases. 0OP-06.4 requires Unit CRS and Shift Manager signatures for releases.

Distractor Analysis:

Choice A: Plausible because ODCM compensatory measures are required.

Choice B: Plausible because CRS signature is required.

Choice C: Correct Answer, see explanation

Choice D: Plausible because CRS and Shift Manager signatures are required.

SRO Basis: Radiation hazards that may arise during normal and abnormal situations, including maintenance activities and various contamination conditions. [10 CFR 55.43(b)(4)]

0OP-06.4:

NOTE

ODCM 7.3.1, Radioactive Liquid Effluent Monitoring Instrumentation, contains compensatory requirements if the Liquid Radwaste Radioactivity Effluent Monitor and/or Liquid Radwaste Effluent Flow Measuring Device is INOPERABLE. ☐

9. **Confirm** the following instrumentation is OPERABLE:

   a. Liquid Radwaste Radioactivity Monitor, 2-D12-RM-K604 .... ______ CRS

   b. Liquid Radwaste Effluent Flow Measurement Device, 2-G16-FIT-N057 .... ______ CRS

BSEP Radioactive Liquid Release Permit

Special instructions for release: ____

____

____

____

____

10. Approval to release____

(Unit CRS) (Date/TIME)

11. Approval to release: ____

(Shift Manager) (Date/Time)

7.3.1 RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

ODCMS 7.3.1 The radioactive liquid effluent monitoring instrumentation channels in Table 7.3.1-1 shall be OPERABLE.

NOTE
The annunciator function may be removed from operation for performance of troubleshooting for up to 30 minutes provided the associated function maintains monitoring capability

APPLICABILITY: In accordance with Table 7.3.1-1.

COMPENSATORY MEASURES

NOTE
Separate Condition entry is allowed for each required channel.

COMPENSATORY MEASURES (continued)

CONDITION	REQUIRED COMPENSATORY MEASURE		COMPLETION TIME
C. As required by Required Compensatory Measure A.1 and referenced in Table 7.3.1-1.	C.1	Estimate the flow rate through the associated pathway using pump performance curves or tank level indicators.	Once per 4 hours during releases through the associated line
	AND		
	C.2	Restore the channel to OPERABLE status.	30 days

99. SG2.4.37 1

During accident conditions, an auxiliary operator is needed to enter the reactor building for local emergency actions to prevent fuel damage. Due to elevated reactor building radiation levels, it is estimated the operator will receive 7.5 rem.

Which one of the following completes the statements below?

The estimated dose of 7.5 rem ___(1)___ exceed EPA-400 limits.

The Site Emergency Coordinator ___(2)___ authorize exceeding 10CFR20 limits IAW 0PEP-3.7.6, Emergency Exposure Controls.

A. (1) will not
   (2) can

B. (1) will not
   (2) cannot

C. (1) will
   (2) can

D. (1) will
   (2) cannot

Answer: A

K/A:

G2.4.37 Knowledge of the lines of authority during implementation of the Emergency Plan. (CFR: 41.10 / 45.13)

RO/SRO Rating: 3.0/4.1

Pedigree: 2014 NRC Exam

Objective: CLS-LP-102-A, Objective 11:
State the emergency worker exposure limits listed in EPA 400 for each of the following conditions: b. Protection of valuable property

Reference: None

Cog Level: Fundamental

Explanation: Per PEP-03.7.6, emergency limits follow EPA-400 guidelines of 10 rem for protection of valuable property and 25 rem for life saving action. Exceeding 10 CFR 20 limits (5 rem) requires authorization of the SEC for onsite activities.

Distractor Analysis:

Choice A: Correct Answer, see explanation

Choice B: Plausible because part 1 is correct, protection of personnel is often reserved for the plant manager, VP or above. In this case, the SEC is the highest line of authority during an emergency.

Choice C: Plausible because on-site radiation dose levels are governed by 10CFR20. EPA-400 guidelines are higher, and part 2 is correct.

Choice D: Plausible because on-site radiation dose levels are governed by 10CFR20. Protection of personnel is often reserved for the plant manager, VP or above. In this case, the SEC is the highest line of authority during an emergency.

SRO Basis: Radiation hazards that may arise during normal and abnormal situations, including maintenance activities and various contamination conditions. [10 CFR 55.43(b)(4)]

4.0 RESPONSIBILITIES

4.2 The Site Emergency Coordinator is responsible for authorization of exposures in excess of 10CFR20 limits and approval of the administration of potassium iodide (KI) for station ERO personnel performing onsite functions.

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ATTACHMENT 2

Emergency Exposure Guidelines

Exposure guidelines in this attachment are consistent with EPA Emergency Worker and Lifesaving Activity Protective Action Guides described in EPA 400-R-92-001.

Every reasonable effort will be used to ensure that an emergency is handled in such a manner that no worker exceeds the normal exposure limits, including the administering of radioprotective drugs. In emergency situations, workers may receive exposure under a variety of circumstances in order to assure safety and protection of others and of valuable property. These exposures will be justified if the maximum risks or costs by the actions outweigh the risks to which the workers are subjected. The Emergency Worker Dose Limit Guidelines are as follows:

Dose Limit (Rem TEDE)	Activity	Condition
5	All	
10	Protecting valuable property	Lower dose not practicable
25	Lifesaving or protection of large populations	Lower dose not practicable
> 25	Lifesaving or protection of large populations	Only on a voluntary basis to persons fully aware of the risks involved.

100. SG2.4.46 1

A partial loss of drywell cooling on Unit One has occurred.
Drywell pressure 1.5 psig and slowly rising.

Which one of the following completes the statements below?

An expected alarm for this condition is ___(1)___.

The CRS will direct venting containment IAW ___(2)___.

A. (1) A-05 (5-5) *Pri Ctmt Hi/Lo Press*
   (2) 1OP-24, Containment Atmosphere Control System

B. (1) A-05 (5-5) *Pri Ctmt Hi/Lo Press*
   (2) 1OP-10, Standby Gas Treatment System Operating Procedure

C. (1) A-03 (4-9) *RHR High Drywell Press*
   (2) 1OP-24, Containment Atmosphere Control System

D. (1) A-03 (4-9) *RHR High Drywell Press*
   (2) 1OP-10, Standby Gas Treatment System Operating Procedure

Answer: B

K/A:
G2.4.46 Ability to verify that the alarms are consistent with the plant conditions.
(CFR: 41.10 / 43.5 / 45.3 / 45.12)

RO/SRO Rating: 4.2/4.2

Pedigree: Bank

Objective: LOI-CLS-LP-302D Obj 2
Given plant conditions and AOP-14.0, determine the required supplementary actions. (LOCT)

Reference: None

Cog Level: High

Explanation: A-05 alarm indicates that DW pressure is 1.5 psig while the A-03 alarm indicates pressure is above 1.7 psig. AOP-14 would be entered and venting will be directed before pressure reaches 1.7 psig. Venting is performed IAW OP-10. OP-24 is directed in AOP-14 and will provide guidance for operation of ventilation, inerting and de-inerting but not for venting.

Distractor Analysis:

Choice A: Plausible because the first part is correct and OP-24 is directed in AOP-14 but not for venting of containment.

Choice B: Correct Answer, see explanation

Choice C: Plausible because this is a high DW press alarm and OP-24 is directed in AOP-14 but not for venting of containment.

Choice D: Plausible because this is a high DW press alarm and OP-10 is correct.

SRO Basis:
Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations. [10 CFR 55.43(b)(5)]

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4.2.3 Primary Containment Pressure High (continued)

e. **Vent** the drywell as necessary in accordance with 1OP-10(2OP-10), Standby Gas Treatment System Operating Procedure. ☐

f. **IF** drywell pressure continues to rise, **THEN** before drywell pressure reaches 1.7 psig, **perform** the following: ☐

(1) **Insert** a manual scram. ☐

(2) **Enter** 1EOP-01-RSP(2EOP-01-RSP), Reactor Scram Procedure. ☐

3. **IF** drywell pressure is greater than 1.7 psig, **THEN go to** 0EOP-02-PCCP, Primary Containment Control Procedure. ☐