2013 PERRY NUCLEAR POWER PLANT

INITIAL LICENSE EXAMINATION

ADMINISTRATIVE FILES

Perry Nuclear Power Plant
10 Center Road
P.O. Box 97
Perry, Ohio 44081

FENOC
First Energy Nuclear Operating Company

March 15, 2013

ATTN: Mr. Michael Bielby
Chief Examiner, Operations Branch, Region III
U. S. Nuclear Regulatory Commission
2443 Warrenville Road, Suite 210
Lisle, IL 60532-4352

SUBJECT:
Perry Nuclear Power Plant
NRC Post-Examination Material, Perry License Class 1-LOT-13

In accordance with the guidance provided in NUREG-1021, Operator Licensing Examination Standards for Power Reactors (Revision 9 Supplement 1), ES-501, Initial Post-Examination Activities, Section C.1, the following materials are submitted in support of the Perry Nuclear Power Plant written operator license examination conducted on March 6, 2013.

1. Form ES 201-3, Examination Security Agreement sheets

All other post exam documentation has been sent to you under correspondence L-13-117 dated March 5, 2013. This completes the necessary documentation for the Perry Unit 1 Initial Licensing Examination.

If there are any questions or if additional information is required, please contact Ray Torres at (440) 280-5277.

Sincerely,

Ray Torres

Ray Torres

Attachments:
Form ES-201-3, Examination Security Agreement sheets (6 sheets)

MAR 18 2013

ORIGINAL

1. **Pre-Examination**

I acknowledge that I have acquired specialized knowledge about the NRC licensing examinations scheduled for the week(s) of 2/25/13 & 3/4/13 as of the date of my signature. I agree that I will not knowingly divulge any information about these examinations to any persons who have not been authorized by the NRC chief examiner. I understand that I am not to instruct, evaluate, or provide performance feedback to those applicants scheduled to be administered these licensing examinations from this date until completion of examination administration, except as specifically noted below and authorized by the NRC (e.g., acting as a simulator booth operator or communicator is acceptable if the individual does not select the training content or provide direct or indirect feedback). Furthermore, I am aware of the physical security measures and requirements (as documented in the facility licensee's procedures) and understand that violation of the conditions of this agreement may result in cancellation of the examinations and/or an enforcement action against me or the facility licensee. I will immediately report to facility management or the NRC chief examiner any indications or suggestions that examination security may have been compromised.

2. **Post-Examination**

To the best of my knowledge, I did not divulge to any unauthorized persons any information concerning the NRC licensing examinations administered during the week(s) of 2/25/13 & 3/4/13. From the date that I entered into this security agreement until the completion of examination administration, I did not instruct, evaluate, or provide performance feedback to those applicants who were administered these licensing examinations, except as specifically noted below and authorized by the NRC.

	PRINTED NAME	JOB TITLE / RESPONSIBILITY	SIGNATURE (1)	DATE	SIGNATURE (2)	DATE	NOTE
1.	RJ Torres	FitzExam Team Lead (PY)	[illegible]	5-23-12	[illegible]	3/7/13	
2.	DW O'Donnell	Facility Reviewer	[illegible]	6·12·12	[illegible]	3·7·13	
3.	Michael Garnett	Procedure Writer	[illegible]	7/1/12	[illegible]	3/7/13	
4.	Gary G. Rhoads	Exam Developer	[illegible]	7/16/12	[illegible]	3/7/13	
5.	Cynthia L Miklacic	Simulator Support	Cynthia L Miklacic	7/23/12	Cynthia L Miklacic	3/7/13	
6.	Stephen J. Kapostasy	Technical Reviewer	[illegible]	8/17/12	[illegible]	3/8/13	
7.	James R Cross	Reactor operator	[illegible]	10/3/12	[illegible]	3/8/13	
8.	John L Flowers	Reactor operator	John L Flowers	10/3/12	John L Flowers	3/7/13	
9.	David Dossin	SRO	[illegible]	10/3/12	[illegible]	3/7/13	
10.	Christopher M. Elliott	SRO	[illegible]	10/3/12	[illegible]	3/12/13	
11.	James A. Gerber	H/W tech Sim	[illegible]	10/15/12	[illegible]	3/7/12	
12.	Douglas J. Pierce	SRO Cert. Instructor	[illegible]	10-15/12			①
13.	Dustin Reese	SRO	[illegible]	10/15/12	[illegible]	3/8/12	
14.	Robert McCreary	RO	[illegible]	10/15/12	[illegible]	3/12/13	
15.	Doug Shorten	SRO	[illegible]	10/15/12	[illegible]	3/12/13	

NOTES: SHEET 1 OF 6

① SIGNATURE ON COPY OF SECURITY AGREEMENT.

ORiginaL

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	PRINTED NAME	JOB TITLE / RESPONSIBILITY	SIGNATURE (1)	DATE	SIGNATURE (2)	DATE	NOTE
1.	Richard J. Brooks	Fleet Exam Lead		11/19/12			①
2.	Paul K Hetrick	US/SRO instructor		12/10/12		3/7/13	
3.	JOHN F. PELCIC	PY REGULATORY COMPLIANCE		12/19/12		3/7/13	
4.	Rick Strehl	Trng superintendent		1/26/13		3/7/13	
5.	MATTHEW NELSON	RO/SRO		2/4/13		3/7/13	
6.	Lance M. Bump	RO		2/5/13		3/7/13	
7.	Jim CASE	SRO		2/5/13		3/7/13	
8.	Joe Evans	RO		2/5/13		3-8-13	
9.	DALE RICHMOND	SIMULATOR		2/5/13		3/7/13	NONE
10.	DANIEL RONIGER	RO		2/7/13		3/7/13	
11.	PAUL EISENMANN	FLEET EXAM DEVELOPER		2/11/13			①
12.	JAMES KELLY	SUPV ILO		2/25/13		3/7/13	
13.	CARLOS N CHAVEZ	Instructor		2/25/13		3/11/13	
14.	mike Brown	OPERATIONS		2/27/13		3/7/13	
15.	JAMES E AGNEW	INSTRUCTOR		3/4/13		3/7/13	

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Sheet 2 of 6

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	PRINTED NAME	JOB TITLE / RESPONSIBILITY	SIGNATURE (1)	DATE	SIGNATURE (2)	DATE	NOTE
1.	David P Johnson	Instructor	David P Johnson	3/5/13	David P Johnson	3/7/13	
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NOTES: SHEET 3 OF 6

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1.	Richard J. Brooks	Fleet Exam Lead	[signature]	11/19/12			
2.	Paul K Hetrick	US/SRO instructor	[signature]	12/10/12	[signature]	3/7/13	
3.	JOHN F. PELCIC	PY REGULATORY COMPLIANCE	[signature]	12/19/12			
4.	Rick Strehl	Trng superintendent	[signature]	1/26/13			
5.	MATTHEW NELSON	RO/SRO	[signature]	2/4/13			
6.	Lance M. Bump	RO	[signature]	2/5/13			
7.	Jim CASE	SRO	[signature]	2/5/13			
8.	Joe Evans	RO	[signature]	2/5/13			
9.	DALE RICHMOND	SIMULATOR	[signature]	2/5/13	[signature]	3/7/13	None
10.	DANIEL ROISIGER	RO	[signature]	2/7/13			
11.	PAUL EISENMANN	FLEET EXAM DEVELOPER	[signature]	2/11/13	[signature]	3/7/13	1
12.	JAMES KELLY	SUPV ILO	[signature]	2/25/13			
13.	CARLOS N CHAVEZ	Instructor	[signature]	2/25/13			
14.	Mike Brown	OPERATIONS	[signature]	2/27/13			
15.	JAMES E AGNEW	INSTRUCTOR	[signature]	3/4/13			

NOTES: 1 SIGNATURE ON COPY OF SECURITY AGREEMENT. [initials]

sheet ~~2~~ 4 of 6
RSI 3/12/13

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1.	Richard J. Brooks	Fleet Exam Lead		11/19/12		3/8/13	1
2.	Paul K Hetrick	US/SRO instructor		12/10/12		3/7/13	
3.	JOHN F. PELCIC	PY REGULATORY COMPLIANCE		12/19/12			
4.	Rick Strohl	Trng superintendent		1/26/13			
5.	MATTHEW NELSON	RO/SRO		2/4/13			
6.	Lance M. Bump	RO		2/5/13			
7.	JIM CASE	SRO		2/5/13			
8.	Joe Evans	RO		2/5/13			
9.	DALE RICHMOND	SIMULATOR		2/5/13		3/7/13	None
10.	DANIEL ROMIGER	RO		2/7/13			
11.	PAUL EISENMANN	FLEET EXAM DEVELOPER		2/11/13			
12.	JAMES KELLY	SUPV ILO		2/25/13			
13.	CARLOS J CHAVEZ	Instructor		2/25/13			
14.	Mike Brosnan	OPERATIONS		2/27/13			
15.	JAMES E AGNEW	INSTRUCTOR		3/4/13			

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Sheet ~~2~~ 5 of 6
RST 3/12/13

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	PRINTED NAME	JOB TITLE / RESPONSIBILITY	SIGNATURE (1)	DATE	SIGNATURE (2)	DATE	NOTE
1.	RJ Tomes	Fit Exam Team Lead (PY)	[illegible]	5-23-12	[illegible]	3/7/13	
2.	DW O'Donnell	Facility Reviewer	[illegible]	6-12-12	[illegible]	3-7-13	
3.	Michael Garrett	Procedure Writer	[illegible]	7/16/12	[illegible]	3/7/13	
4.	GARY G. RHOADS	Exam DEVELOPER	[illegible]	7/16/12	[illegible]	3/7/13	
5.	Cynthia L Miklacic	Simulator Support	Cynthia L Miklacic	7/23/12	Cynthia L Miklacic	3/7/13	
6.	Stephen J. Kopostary	Technical Reviewer	[illegible]	8/17/12	[illegible]	3/8/13	
7.	James R Cross	Reactor operator	[illegible]	10/3/12	[illegible]	3/8/13	
8.	John L Flowers	Reactor operator	John L Flowers	10/3/12	John L Flowers	3/7/13	
9.	David Doesing	SRO	[illegible]	8/3/12	[illegible]	3/7/13	
10.	Christopher M. Elliott	SRO	[illegible]	10/3/12	[illegible]	3/12/13	
11.	James A. Gerber	H/W tech Sim	[illegible]	10/15/12	[illegible]	3/7/13	
12.	Douglas J. Pierce	SRO Cert. Instructor	[illegible]	10-15-12	[illegible]	3/15/13	1
13.	Dustin Reese	SRO	[illegible]	10/15/12	[illegible]	3/8/13	
14.	Robert McLeary	RO	[illegible]	10/15/12	[illegible]	3/12/13	
15.	Doug Shorter	SRO	[illegible]	10/15/12	[illegible]	3/12/13	

NOTES: SHEET ~~1~~ 6 OF 6

RJT 3/15/13

1 SIGNED COPY OF SECURITY AGREEMENT RJT

Written Examination Grading Quality Checklist

Facility: **PERRY** Date of Exam: **2/25/2013 thru 3/8/2013** Exam Level: RO ☒ SRO ☒

	Item Description	Initials a	Initials b	Initials c
1.	Clean answer sheets copied before grading	RJT	RS	MEB
2.	Answer key changes and question deletions justified and documented	RJT	RS	MEB
3.	Applicants' scores checked for addition errors (reviewers spot check > 25% of examinations)	RJT	RS	MEB
4.	Grading for all borderline cases (80 ±2% overall and 70 or 80, as applicable, ±4% on the SRO-only) reviewed in detail	RJT	RS	MEB
5.	All other failing examinations checked to ensure that grades are justified	RJT	RS	MEB
6.	Performance on missed questions checked for training deficiencies and wording problems; evaluate validity of questions missed by half or more of the applicants	RJT	RS	MEB

	Printed Name/Signature	Date
a. Grader	Raymond J. Torres	3/11/13
b. Facility Reviewer(*)	Richard Strohl	3/11/13
c. NRC Chief Examiner (*)	Michael Bielby	4/5/13
d. NRC Supervisor (*)	Hironori Peterson	4/5/13

(*) The facility reviewer's signature is not applicable for examinations graded by the NRC; two independent NRC reviews are required.

FENOC™
FirstEnergy Nuclear Operating Company

Perry Nuclear Power Plant
P.O. Box 97
10 Center Road
Perry, Ohio 44081

Vito A. Kaminskas
Vice President

440-280-5382
Fax: 440-280-8029

March 13, 2013
L-13-117

10 CFR 55

ATTN: Mr. Michael Bielby
Chief Examiner, Operations Branch, Region III
U. S. Nuclear Regulatory Commission
2443 Warrenville Road, STE 210
Lisle, IL 60532-4352

SUBJECT:
Perry Nuclear Power Plant
Docket No. 50-440, License No. NPF-58
NRC Post-Examination Material, Perry License Class 1LOT13

In accordance with the guidance provided in NUREG-1021, Operator Licensing Examination Standards for Power Reactors (Revision 9, Supplement 1), ES-501, Initial Post-Examination Activities, Section C.1, the following materials are submitted in support of the Perry Nuclear Power Plant (PNPP) written operator license examination conducted on March 6, 2013:

1. Graded written examinations (i.e., each applicant's original answer and examination cover sheets) and two clean copies of each applicant's answer sheet
2. Master examination answer key (No changes were made to the examinations during administration and grading.)
3. All questions asked by the applicants during the administration of the examination and the proctor's responses to those questions
4. Substantive comments made by the applicants following the written examination
5. Written Examination Seating Chart
6. Form ES-403-1, Written Examination Grading Quality Checklist
7. Scenario #2 forms ES-D-1 and ES-D-2 (modified during exam week)
8. Form ES-301-1 (SRO) (modified during exam week)
9. JPM OT-3701-ADM-027_SRO (added during exam week)

The post-examination performance analysis identified potential areas of weakness. Several questions on the Reactor Operator and Senior Reactor Operator written examination were missed by 50 percent or more of the applicants. This condition was entered in the Corrective Action Program for further investigation under Condition Report 2013-03592.

All individuals signed onto Form ES-201-3, Examination Security Agreement, have not yet completed the post examination signature. When form ES-201-3 has been completed, it will be forwarded to you, thus completing the necessary documentation for the PNPP Initial Licensing Examination.

There are no regulatory commitments contained in this letter. If there are any questions or if additional information is required, please contact Mr. Raymond Torres at (440) 280-5277 or Mr. Anthony E. Mueller, Jr. at (440) 280-5535.

Sincerely,

Vito A. Kaminskas

Attachments:

1. Original Graded Written Examinations and Applicant's Answer Sheets
2. Master Examination Answer Key
3. Questions asked and answers given to the applicants during the written examination
4. Substantive comments made by the applicants following the written examination
5. Written Examination Seating chart
6. Form ES-403-1, Written Examination Grading Quality Checklist
7. Scenario #2 forms ES-D-1 and ES-D-2 (modified during exam week)
8. Form ES-301-1 (SRO) (modified during exam week)
9. JPM OT-3701-ADM-027_SRO (added during exam week)

cc: NRC Project Manager (w/o attachments)
NRC Resident Inspector (w/o attachments)
NRC Region III Branch Chief (w/o attachments)
NRC Document Control Desk (w/o attachments)

Attachment 1
L-13-117

Original Graded Written Examinations and Applicant's Answer Sheets
(105 pages)

Attachment 2
L-13-117

Master Examination Answer Key
(1 page)

Attachment 3
L-13-117

Questions asked and answers given to the applicants during the written examination
(5 pages)

Perry License Class 1-LOT-13 NRC 2013 Exam - Questions from candidates
3/6/2013

	Q#	Candidate	The candidate asked.	The proctor answered.
1	RO 29	Douglass	The question has 2 parts. In the first part, does the fire affect plant safety?	Make no assumptions. Read each part as a separate question.
2	RO 26	Waggel	Is 1B21-N0673C a RCIC instrument?	Answer the question the best you can.
3	SRO 7	Vaughn	Is ICS available?	Answer the question the best you can.
4	RO 34	Smith D	Have control rods moved before the operator manipulates the Recirc pumps?	Re-read the initial conditions.
5	RO 23	Hallmark	Are the scram times already compensated for the Rx pressure value per SVI-C11-T1006?	Don't make any assumptions.
6	SRO 7	Hallmark	Per EAL JU1, is the last part of the EAL met by the initial condition, "the shift manager was able to recruit off shift personnel for additional control room monitoring"?	Answer the question the best you can.
7	RO 35	Jones	Does 'division' refer to divisional valves of division of SDC?	Re-read the question.

Perry License Class 1-LOT-13 NRC 2013 Exam - Questions from candidates
3/6/2013

	Q#	Candidate	The candidate asked.	The proctor answered.
8	RO 49	Wiley	Have the contingency steps at the beginning of EOP-SPI 6.6 been completed (i.e., remove the trip units, etc)?	Answer the question the best you can with the information provided.
9	RO 50	Douglass	Does the word 'LOCA' imply an actual LOCA initiation or is it used as a generic term?	A valid LOCA initiation has occurred. Placed on white board for all students.
10	RO 56	Jones	What is the definition of 'immediate'?	Provided candidate with dictionary.
11	SRO 9	Vaughn	Was the reset of alarm P680-5-E5 valid or not?	Answer the question the best you can with the information provided.
12	RO 53	Jones	Do the first 2 bullets represent initial plant conditions?	Re-read the question. Make no assumptions.
13	RO 56	Jones	Does the LOCA cause fuel damage by uncovering the fuel?	Re-read the question. 'LOCA' means a valid 'LOCA' initiation signal has been received.
14	SRO 15	Vaughn Vriezen	Did RCIC initiate or not?	Answer the question the best you can with the information provided.

Perry License Class 1-LOT-13 NRC 2013 Exam - Questions from candidates
3/6/2013

	Q#	Candidate	The candidate asked.	The proctor answered.
15	RO 67	Wiley	How can we have indicated H2 concentrations if water level has not been below TAF for >30 minutes?	Answer the question the best you can.
16	RO 62	Jones	What causes P52-F050 to close? What was the malfunction? Should I assume an air leak?	Re-read the question. Make no assumptions.
17	SRO 2	Waggle	Am I trying to maintain proficiency as the US, SM, or an SRO?	Answer the question the best you can.
18	SRO 22	Vaughn	Am I allowed to have a copy of OAI-1701?	The question does not provide any references, Answer the question the best you can.
19	RO 14	Smith T	Are the answers testing the Tech Spec Bases for the various RPS functions?	Re-read the question. Answer the question the best you can.
20	SRO 15	Douglass	Am I predicting the correct answer based on the effects of the 'shrink' or 'swell'?	Re-read the question. Answer the question the best you can.
21	RO 75	Kuntz	What impact did the malfunction have on the SRV's other than stated?	Re-read the question. Make no assumptions.

Perry License Class 1-LOT-13 NRC 2013 Exam - Questions from candidates
3/6/2013

	Q#	Candidate	The candidate asked.	The proctor answered.
22	RO 35	Chiarelli	Does the question ask what is the impact on RHR SDC due to the loss of RPS bus A or just the overall impact due to the loss of RPS bus A?	Re-read the conditions and question in order to answer the best you can.
23	RO 33	Vaughn	Is the question asking how much time is left to get the SRV closed before a scram is required?	Re-read the conditions and question in order to answer the best you can.
24	RO 69	Vaughn	After the reactor scram, were any other operator actions taken?	Make no assumptions.
25	SRO 8	Vaughn	Have all my Tech Spec Actions been completed for the inoperable OPRM's?	Unable to provide more information. Answer the question the best you can.
26	RO 67	Douglass	Is this question testing my knowledge of the SAGs?	Unable to provide more information. Answer the question with the information provided.
27	RO 30	Vriezen	Is the Div 1 DG in parallel with the grid?	Answer the question the best you can.
28	RO 47	Vaughn	What is the definition of 'deplete'?	Provided candidate with dictionary.

Perry License Class 1-LOT-13 NRC 2013 Exam - Questions from candidates
3/6/2013

	Q#	Candidate	The candidate asked.	The proctor answered.
29	SRO 5	Douglass	Where is the Offgas Vent Pipe rad monitor located in regards to the Offgas effluent flow path?	Unable to provide more information. Answer the question with the information provided.
30	RO 48	Vaughn	Are you asking for all conditions that will make the APRM inop or just the automatic conditions (in regard to LPRMs)?	Evaluate operability under all conditions.
31	RO 62	Vaughn	In the answer, does low IA header pressure mean the same as low IA receiver pressure?	Unable to provide more information. Answer the question with the information provided.
32	RO 72	Vaughn	Does the word 'adjust' include taking 1N21-R475 to 'manual'?	Answer the question with the information provided.
33	RO 75	Douglass	Did the reactor scram?	Unless stated otherwise, assume all other equipment functions as designed.

Attachment 4
L-13-117

Substantive comments made by the applicants following the written examination
(65 pages)

Perry Nuclear Power Plant Comments

Question RO #40

1. **Recommendation** – Amend the Answer Key to list 'Answer B' as the only correct answer since we have provided "newly discovered technical information that supports a change in the answer key."
2. **Justification -** The question asks what the response of the LPCS pump will be following a LOOP/LOCA signal if the LPCS pump initiation logic had been previously overridden. The plant drawings show the LPCS pump response will be to immediately start when power is restored to the emergency bus by the diesel generator with out time delay. See Attachment #1 for details.
3. **References** – Plant drawings 208-0060-00004 revision AA and 208-0060-00008 revision DD.

Question RO #66

1. **Recommendation** – Amend the Answer Key to list 'Answer A' as the only correct answer since we have provided "newly discovered technical information that supports a change in the answer key."
2. **Justification** –The question asks what the temperature difference is between bulk reactor coolant and the RPV metal temperature and whether the Technical Specification cooldown rate has been exceeded when lowering reactor pressure at a constant rate. Technical data shows the Technical Specification cooldown rate will be exceeded when pressure is reduced at the rate prescribed in the question stem. See Attachment #2 for details.
3. **References** – ABB Steam Tables

Question SRO #1

1. **Recommendation** – Amend the Answer Key to list 'Answer D' as a correct answer in addition to 'Answer B' since we have provided "newly discovered technical information that supports a change in the answer key."
2. **Justification -** The question asks the required Shift Manager actions after the determination that adequate core cooling no longer exists and the required status of the Emergency Operating Procedures when transitioning to the Severe Accident Guidelines. The SAG-1 bases states "The requirements for Primary Containment flooding in the RPV Control, EOP-01, Level Power Control, EOP-01A, RPV Flooding, EOP-04-4, flowcharts are directed through concurrent transitions to the Primary Containment Flooding, SAG-1 and the RPV, Primary Containment, and Radioactivity Release Control, SAG-2 Flowcharts. Through these transitions, all parameter control paths transfer from the EOP flowcharts to the SAGs." See Attachment # 3 for details.
3. **References** – EOP-1 pg 61, SAG-1 bases pg 6 and SAG-2 pg 8

Question SRO #8

1. **Recommendation** – Delete question from exam, there is no correct answer since this was "a question with an unclear stem that confused the applicants or did not provide all the necessary information."
2. **Justification** – The question asks what actions are required for a trip of a Reactor Recirculation pump per ONI-C51. The stem of the question does not provide all the necessary information to answer the question. ONI-C51 FLOWCHART requires the operator to determine the actual core flow using core plate ΔP which was not provided. See Attachment # 4 for details.
3. **References** – ONI-C51, UNPLANNED CHANGE IN REACTOR POWER OR REACTIVITY, ONI-C51 Flowchart

Question SRO #14

1. **Recommendation** – The station staff does not support the candidate's basis and this question does, in fact, have only one correct answer.
   Justification - With the conditions stated in the stem there are additional actions that can be taken to restore and maintain RPV pressure within the limits of HCL, Figure 4 and should be directed first by the Unit Supervisor which requires opening an additional SRV to lower RPV pressure. See attachment #5 for details.
3. **References** –EOP-SPI Supplement, EOP Bases, PYBP-POS-0030, OAI-1703

Question SRO #15

1. **Recommendation** – The station staff does not support the candidate's basis and this question does, in fact, have only one correct answer.
2. **Justification** - While it is true over the next ten minutes RPV Water level both rises and lowers, the causes indicate there is only one correct answer. The injection of cold water from HPCS causes level to rise due to swell as the cold water injected begins to heat up and expand making answer A the correct answer. RPV Water Level begins to lower later in the ten minute period however the cause is a loss of inventory due to open main steam line drains which makes C an incorrect answer. See Attachment #6.
3. **References** – None provided by candidate

Question SRO #18

1. **Recommendation** – Amend the Answer Key to list 'Answer A' as a correct answer, in addition to 'Answer D' since we have provided "newly discovered technical information that supports a change in the answer key."
2. **Justification** – The question asks how the LPCI mode of operation is impacted when a train of the Residual Heat Removal System is placed in the Shutdown Cooling Mode of operation. In Mode 4, only 2 ECCS injection/spray subsystems shall be OPERABLE (TS 3.5.2). Per TS 3.5.2 Bases, one LPCI subsystem may be considered operable during alignment or operation for decay heat removal in Mode 4 or 5, if capable of being manually realigned to the LPCI mode. ALL ECCS systems are OPERABLE therefore LPCI A is NOT required to be OPERABLE. See Attachment #7 for details.
3. **References** – TS 3.5.2, TS 3.5.2 bases

<u>ATTACHMENT #1</u>

QUESTION RO 40

The plant was operating at rated power.

An inadvertent initiation of Low Pressure Core Spray occurred due to failure of DW pressure trip units.

<u>Only</u> the Immediate Actions of ONI-E12-1, Inadvertent Initiation of ECCS or RCIC were performed and were successful.

Subsequently, a loss of offsite power occurred coincident with a LOCA.

When power is restored to the divisional buses by the diesel generators, LPCS will ____.

A. <u>not</u> automatically restart

B. automatically restart immediately

C. automatically restart in 10 seconds

D. automatically restart in 15 seconds

QUESTION RO 40

	Level:	RO	SRO
	Tier #	2	
	Group #	1	
	K/A#	209001	K2.01
	Importance Rating	3.0	

1.0 K&A: KNOWLEDGE OF ELECTRICAL POWER SUPPLIES TO THE FOLLOWING: PUMP POWER

LPCS

Explanation: **Answer A** – The Immediate Action for an inadvertent initiation of LPCS is to override the pump to OFF. Since the LPCS pump has been overridden off, K13 is energized, preventing the pump from restarting.

B – Incorrect – This would be true if the pump had not been previously overridden off.

C – Incorrect – This is the time allowed for the DG to reenergize the bus.

D – Incorrect – This is the 'normal' time delay for the LPCS pump to start without a LOOP.

Technical Reference(s): OT-COMBINED-E21 LP (PowerPoint) Rev 1	Reference Attached: OT-COMBINED-E21 LP (PowerPoint) slides 39-41

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OT-COMBINED-E21-F

Question Source:	Bank #	
	Modified Bank #	Monticello 2009
	New	
Question History:	Previous NRC Exam	Monticello 2009 #RO-31
Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	x
10 CFR Part 55 Content:	55.41	b.7
	55.43	

Comments: Level of Difficulty = x

Candidate Comments:

1. The question asks how the LPCS pump will respond following a LOOP/LOCA actuation signal if the pump initiation logic had been previously overridden.
2. The answer key incorrectly lists distracter 'A' as the correct answer
3. Per plant drawings 208-0060-00004 revision AA and 208-0060-00008 revision DD the LPCS Override Logic Seal-in will de-energize during a Loss of Offsite Power and when power is restored the LPCS pump will automatically restart with no time delay
4. Amend the answer key to list distracter 'B' as the correct answer

Candidate Justifications for answers:

A. Incorrect answer – The LPCS pump will automatically restart, the override seal-in logic de-energizes during a LOOP

B. Correct answer – The LPCS pump will automatically restart without time delay

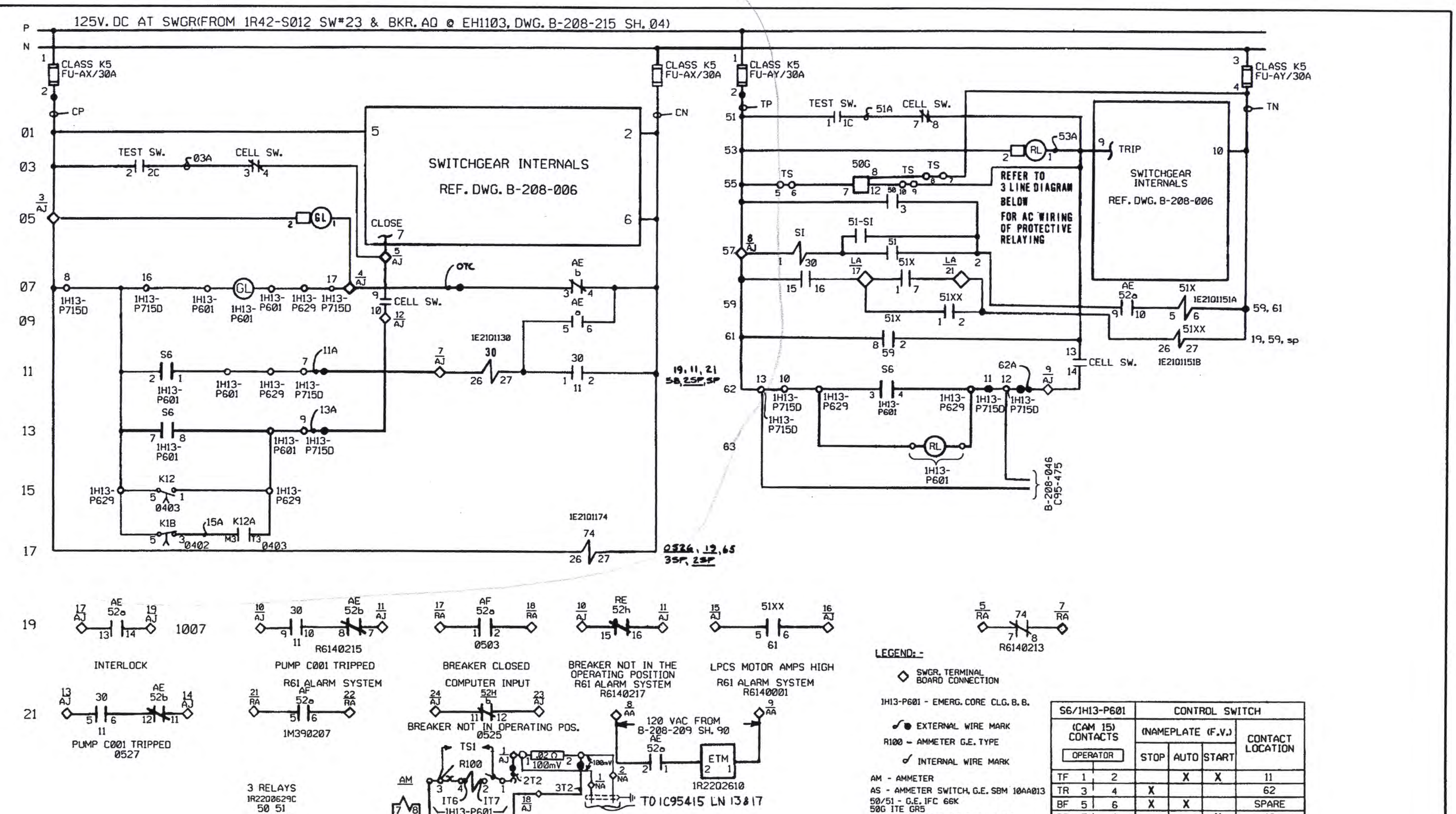
C. Incorrect answer – The LPCS pump will automatically restart without time delay, 10 seconds is the time allowed for the DG to energize the bus.

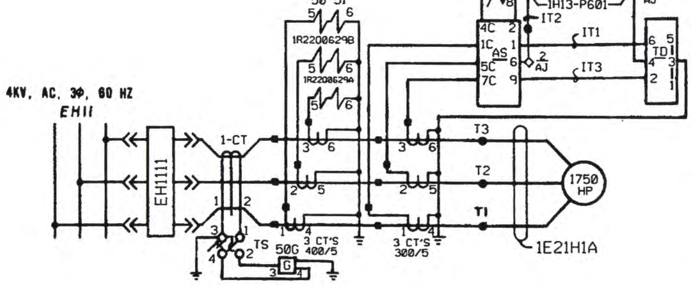
D. Incorrect answer – The LPCS pump will automatically restart without time delay. The 15 second time delay is for a LOCA only without a LOOP.

Recommendation: Amend the Answer Key to list 'Answer B' as the correct answer.

References: Plant drawings 208-0060-00004 revision AA and 208-0060-0008 revision DD

Station Proposed Resolution:
Station agrees with the above candidate Comments, Justifications and Recommendation.





NOTES:

1. ALL DEVICES WITHOUT LOCATION MARKS ARE LOCATED ON SWGR.
2. ALPHABETICAL WIRE MARKS ARE PREFIXED BY: SUB-SYSTEM NO., SHEET NO. AND LINE NO. (EX. E21A 08 07A)
3. POTENTIAL WIRE MARKS ARE PREFIXED BY: SUB-SYSTEM NO. AND SHEET NO. (EX. E21A08CP)
4. DEVICE DESIGNATIONS ARE PREFIXED BY: UNIT AND SYSTEM NO. (EX. 1E21-C001), EXCEPT AS NOTED (SEE NOTE 6).
5. FOR GRAPHIC STANDARDS REFER TO DWG. B-208-001.
6. THE FOLLOWING DEVICES ARE PREFIXED BY 1E21A:
   S6 - SWITCH
   51X - AUX. RELAY
   TS1 - TEST SWITCH
   R100 - AMMETER
7. FOR C-H RELAY STANDARDS REFER TO DWG. B-208-002.
8. FOR ITE RELAY STANDARDS REFER TO DWG. B-208-005.
9. SWITCHGEAR SHOWN WITH BREAKER RACKED OUT.

LEGEND: -

◇ SWGR. TERMINAL BOARD CONNECTION

1H13-P601 - EMERG. CORE CLG. B.B.

EXTERNAL WIRE MARK

R100 - AMMETER G.E. TYPE

INTERNAL WIRE MARK

AM - AMMETER
AS - AMMETER SWITCH, G.E. SBM 10AA013
50/51 - G.E. IFC 66K
50G ITE GR5
TS1 - SUPERIOR 2 POLE AM TEST SW. ASSY.'S 2-11
TS - 2 CURRENT, 3 POTENTIAL
TD - TEST DEVICE G.E. PACK-2, 6 POLE
30 G.E. CR120BD07041
51X - G.E. CO. RELAY TYPE HMA24A
1H13-P715D - TERMINATION CABINET TM004
74 G.E. CR120BD07041
ETM - ELAPSED TIME METER
1H13-P629 - DIV. 1 AUX. RELAY PNL.
51XX G.E. CR120BD07041

REFERENCE: -
G.E. CO. DRAWING 828F535CA SHEET 05

S6/1H13-P601		CONTROL SWITCH			
(CAM 15) CONTACTS		(NAMEPLATE (F.V.)			CONTACT LOCATION
OPERATOR		STOP	AUTO	START	
TF 1	2		X	X	11
TR 3	4	X			62
BF 5	6	X	X		SPARE
BR 7	8			X	13
TFT	10		X	X	SPARE
TRT	12	X			0405
TANDEM BLOCK CR2940-U206		X - CLOSED CONTACTS			
SPRING RETURN TO AUTO					

RECEIVED
APR 07 2011

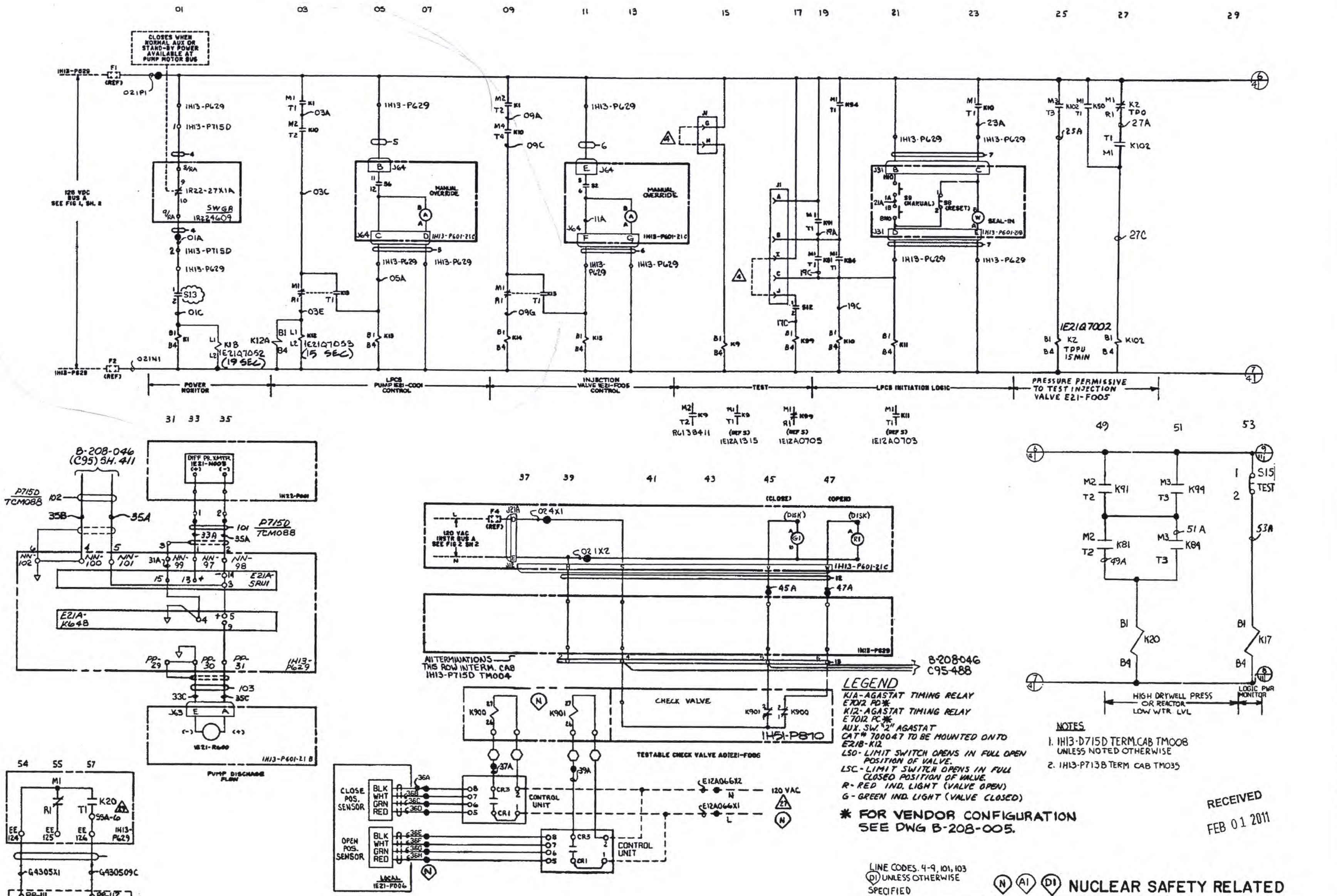
NUCLEAR SAFETY RELATED

THIS AS-BUILT DRAWING REVISED TO INCORPORATE: DUN 11-0126-001-015.

PERRY NUCLEAR POWER PLANT
10 CENTER RD., PERRY, OHIO 44081

ELECTRICAL		ELEMENTARY DIAGRAM	
LOW PRESSURE CORE SPRAY SYSTEM			
LPCS PUMP C001			
MADE	CHECKED	ENGINEER APPROVALS	DATE
S.M. LINDQUIST	Dressler		4/7/2011
SCALE NONE	B	208-0060-00008	DD
SYSTEM E21	SIZE	LEVEL 1	REV

16-MAR-2011 E:\208's\208\208-0060-00008.dgn



(N) (A1) (D1) NUCLEAR SAFETY RELATED

THIS AS-BUILT DRAWING REVISED TO INCORPORATE: DUN 10-0823-001-008.

PERRY NUCLEAR POWER PLANT

10 CENTER RD., PERRY, OHIO 44081

ELECTRICAL			ELEMENTARY DIAGRAM	
LOW PRESSURE CORE SPRAY SYSTEM				
RELAY LOGIC & TESTABLE CHECK VALVE F006				
MADE	CHECKED	ENGINEER APPROVALS		DATE
S.M. LINDQUIST	Dressler	—	[signature]	1/25/2011
SCALE N/A	B	208-0060-00004		AA
SYSTEM E21A	SIZE	LEVEL 1		REV

03-JAN-2011 E:\200's\208\208-0060-00004.dgn

INADVERTENT INITIATION OF ECCS/RCIC

Effective Date: 2-28-13

Preparer: Michael Garnett / 2-11-13
Date

Approver: David Duesing / 2-18-13
Date

TABLE OF CONTENTS PAGE

1.0 ENTRY CONDITIONS

1.1 Alarms

- HPCS PUMP AUTO START RECEIVED
- HPCS DW PRESS HIGH
- HPCS RX LEVEL LO L2
- LPCS AUTO START RECEIVED
- LPCS & LPCI A DW PRESS HIGH
- LPCS & LPCI A RX LEVEL LO L1
- LPCI A AUTO START RECEIVED
- LPCI B & C AUTO START RECEIVED
- LPCI B & C DW PRESS HIGH
- LPCI B & C RX LEVEL LO L1
- RCIC AUTO START RECEIVED
- ADS A PERMISSIVE LPCS/RHR A RUN
- ADS B PERMISSIVE RHR B/C RUN

1.2 Parameters

> NOTE
>
> With reactor pressure above 280 psig for the RHR Pumps or 450 psig for the LPCS Pump, an initiation of LPCI or LPCS, respectively, will not result in cold water actually being injected into the reactor vessel.

- If the initiation results in cold water injection into the RPV, then an Increase in reactor power will occur.
- Increase in reactor vessel water level.

- FEEDWATER FLOW & MAIN STEAM FLOW mismatch.
- Diesel start/running indications.

2.0 AUTOMATIC ACTIONS

2.1 Reduction in feed flow to compensate for emergency system injection.

2.2 Main and Reactor Feed Pump Turbines trip with RCIC initiation signal (4.5 min. time delay if Reactor power >95% as measured by Turbine 1st Stage Pressure)

2.3 Possible reactor scram on high flux or power.

2.4 At a reactor vessel water level of 219 inches the following occurs:

- Reactor Feedwater Pump Turbines trip.
- Main Turbine trips.
- Motor Feed Pump trips.
- Reactor scrams (Reactor System Mode Switch in RUN).
- RCIC Steam Shutoff, 1E51-F045 closes.
- RCIC Injection Vlv, 1E51-F013 closes.
- HPCS Injection Valve, 1E22-F004 closes.

3.0 IMMEDIATE ACTIONS

NOTE

Unit Supervisor concurrence is required to override safety system operation. (reference PAP-0205)

CAUTION

An Emergency Core Cooling System shall **NOT** be manually over-ridden unless one of the following is confirmed:

- Initiation is proven incorrect (beyond a reasonable doubt by two independent indications)
- Continued operation is no longer necessary
- Misoperation in automatic is confirmed.

NA ❑	3.1	**IF** HPCS Initiation is incorrect **OR** misoperation in automatic is confirmed, **THEN TAKE** the HPCS PUMP TO **STOP**.	**1E22-C001**
NA ❑	3.2	**IF** LPCS Initiation is incorrect **OR** misoperation in automatic is confirmed, **THEN TAKE** the LPCS PUMP to **STOP**.	**1E21-C001**
NA ❑	3.3	**IF** LPCI Initiation is incorrect **OR** misoperation in automatic is confirmed, **THEN TAKE** the RHR PUMP to **STOP**.	1E12-C002A 1E12-C002B 1E12-C002C

NA 3.4 ❑ **IF** RCIC Initiation is incorrect **OR** misoperation in automatic is confirmed, **THEN MOMENTARILY DEPRESS** the RCIC TURBINE REMOTE TRIP pushbutton. **1E51-S17**

NA 3.5 **IF** permissives for ADS initiation were met, **THEN MOMENTARILY DEPRESS** the following pushbuttons on P601:

❑ • ADS A LOGIC SEAL IN RESET. **1B21C-S13A**

❑ • ADS B LOGIC SEAL IN RESET. **1B21C-S13B**

> NOTE
>
> The intent of the following step is to only inhibit the logic channel associated with the inadvertent initiation signal.

NA 3.6 ❑ **IF** required to prevent an ADS initiation, **THEN PLACE** affected channel ADS LOGIC INHIBIT switch to INHIBIT. **1B21-S34A** **1B21-S34B**

4.0 SUPPLEMENTAL ACTIONS

NA 4.1 ❑ **IF** HPCS Initiation is incorrect **OR** misoperation in automatic is confirmed, **THEN VERIFY CLOSED** HPCS INJECTION VALVE. **1E22-F004**

NA 4.2 ❑ **IF** LPCS Initiation is incorrect **OR** misoperation in automatic is confirmed, **THEN VERIFY CLOSED** LPCS INJECTION VALVE. **1E21-F005**

NA ❑	4.3	**IF** LPCI Initiation is incorrect **OR** misoperation in automatic is confirmed, **THEN VERIFY CLOSED** LPCI INJECTION VALVE.	**1E12-F042A**	**1E12-F042B**	**1E12-F042C**

NOTE

Placing the SUPR POOL MAKE-UP LOGIC switch in OFF requires a 7-day ALCO in accordance with Tech Spec 3.6.2.4, SPMU System.

NA ❑	4.4	**IF** SPMU TRAIN TIMER RUNNING is in alarm, **AND** initiation is incorrect is confirmed, **THEN PLACE** the SUPR POOL MAKE-UP LOGIC switch in OFF.	**1G43-S6**	**1G43-S8**

NOTE

Opening Brkr D1B08 removes non divisional power to Main and RFPT trip logic for a RCIC initiation. Opening Brkr D1B08 will preclude an inadvertent plant trip if power falls below 95% (as sensed by first stage shell pressure) prior to resetting of the RCIC initiation logic.

NA	4.5	**IF** RCIC has been tripped, **THEN PERFORM** the following:	
❑	4.5.1	**OPEN** the RCIC TURBINE GLAND SEAL COMP Brkr at Bus D-1-B	1E51-C004 D1B08
❑	4.5.2	**REFER TO** SOI-E51 and **PLACE** RCIC in Secured Status.	
❑	4.6	**REFER TO** IOI-18, Emergency Operating Procedure And Isolation Restoration and **PERFORM** RHR LOCA ISOLATION (Level 1 / 1.68#) Isolation Restoration.	

NA 4.7 **IF** an injection occurred,
THEN PERFORM the following:

- ❑ **NOTIFY** the reactor engineer.
- ❑ **DIRECT** Chemistry to evaluate reactor water and Off-Gas Pre-treatment samples to ensure Operational Requirements Manual parameters have not been exceeded (pH, conductivity, chlorides).
- NA ❑ **IF** the ICS is available, **THEN DIRECT** the SE to archive data to aid in evaluating the event.

4.8 **REFER TO** the following Technical Specifications:

- ❑ 3.3.5.1, Emergency Core Cooling System Instrumentation
- ❑ 3.3.5.2, Reactor Core Isolation Cooling System Instrumentation
- ❑ 3.5.1, ECCS - Operating
- ❑ 3.5.2, ECCS - Shutdown
- ❑ 3.6.2.2, Suppression Pool Water Level
- ❑ 3.6.2.1, Suppression Pool Average Temperature
- ❑ 3.5.3, RCIC System
- ❑ 3.6.2.4, SPMU System

❑ 4.9 **RESTORE** the Condensate Storage Tank to normal level.

❑ 4.10 **RESTORE** the Suppression Pool to normal level.

5.0 REFERENCES

None

6.0 RECORDS

The following records are completed/generated by this document:

Quality Assurance Records

None

Non-Quality Assurance Records

None

7.0 SCOPE OF REVISION

Rev. 11 1. Step 4.4 – added a Note to alert the operator that performance of the step requires the initiation of an ALCO per Tech Specs. (600811150, 13-01812)

8.0 ATTACHMENTS

None

ATTACHMENT #2

QUESTION RO 66

The plant was operating at rated power when the following occurred:

- The reactor was scrammed 20 minutes ago due to a problem with the pressure regulator system
- During the transient, both Reactor Recirculation Pumps tripped off
- Current RPV pressure is 600 psig and lowering at 2.5 psig/minute.
- The scram has just been reset
- RPV Bottom Head Drain temperature is 445°F
- RPV Vessel Head Flange temperature is 500°F

Based on these conditions, bulk RPV water temperature currently is approximately__(1)__. If cooldown is allowed to continue at the present rate, Tech Spec cooldown rate __(2)__ Exceeded

Reference Provided:

	__(1)__	__(2)__
A.	44°F > Bottom Head Drain	will
B.	16°F < Vessel Head Flange	will
C.	44°F > Bottom Head Drain	will not
D.	16°F < Vessel Head Flange	will not

This is what we used to come up with our answer.

Time		Pressure	Temp	Delta
T-0		1024	549	60
T+20 min		600	489	19
T+1 hr		500	470	34
T+2 hr	150	350	436	48
T=3 hr	150	200	388	90
T=4 hr	150	50	298	49
T=5 hr		15	249	37
T-6 hr		0	212	

	Level:	RO	SRO
	Tier #	2	
	Group #	2	
	K/A#	216000	K5.11
	Importance Rating	3.2	

1.0 K&A: KNOWLEDGE OF THE OPERATIONAL IMPLICATIONS OF THE FOLLOWING CONCEPTS AS THEY APPLY TO NUCLEAR BOILER INSTRUMENTATION: INDICATED VESSEL TEMPERATURE RESPONSE DURING RAPID HEATUPS OR COOLDOWNS

Nuclear Boiler Inst.

Explanation: **Answer C** – This is a steam table question. The saturated temperature of 600 psig (~615 psia) is ~490°F (489), which is about 44°> the bottom head temp in this case. Additionally, with pressure lowering at 2.5 psig/min, the pressure at 1 hr after the scram will be 500 psig which corresponds to ~470°F. With initial pressure at 1024 psig, Tsat is 549°F. Thus not exceeding 100°F/hr Each hour thereafter, the cool-down rate is l<100°F/hr

A – Incorrect – Cool-down rate will not be exceeded.

B – Incorrect – Cool-down rate will not be exceeded. And, 16°F < Vessel Head Flange is temperature calculated if psia is calculated backwards.

D – Incorrect – 16°F < Vessel Head Flange is temperature calculated if psia is calculated backwards.

Technical Reference(s): ABB Steam Tables	Reference Attached: x

Proposed references to be provided to applicants during examination: **Steam Tables**

Learning Objective (As available): OT-COMBINED-B21(INST)-C

Question Source:	Bank #	
	Modified Bank #	
	New	x
Question History:	Previous NRC Exam	
Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	x
10 CFR Part 55 Content:	55.41	b.5
	55.43	

Comments: Level of Difficulty = x

Candidate Comments:

1. The question asks what the temperature difference is between bulk reactor coolant and the RPV skin and whether the Technical specification cooldown rate has been exceeded when cooling down at 2.5 psig per minute. The answer key should be changed for newly discovered technical information.
2. A cooldown rate maintained at a constant 2.5 psig/minute (150 psig/hr) will exceed the Technical Specification cooldown limit of <100 degrees F per hour.
3. During the one hour period pressure is reduced from 175 psig to 25 psig the cooldown rate is 110 degrees F exceeding the Technical Specification Limit.

Time	Pressure (psig)	Temperature (°F)	Delta (°F)
T+ 2 hr 10 min	325	429	
T+ 3 hr 10 min	175	377	52
T+ 4 hr 10 min	25	267	110

Candidate Justifications for answers:

		__(1)__	__(2)__
A.	**Correct Answer**	**44°F > Bottom Head Drain**	**will**
	Justification	**The saturated temperature of 600 psig (~615 psia) is ~490°F (489), which is about 44°> the bottom head temp in this case.**	**Tech Spec cooldown rate Exceeded 110°F**
B.	Incorrect Answer	16°F < Vessel Head Flange	**will**
	Justification	16°F < Vessel Head Flange is temperature calculated if psia is calculated backwards	**Tech Spec cooldown rate Exceeded 110°F**
C.	Incorrect Answer	**44°F > Bottom Head Drain**	will not
	Justification	**The saturated temperature of 600 psig (~615 psia) is ~490°F (489), which is about 44°> the bottom head temp in this case.**	Tech Spec cooldown rate Exceeded 110°F
D.	Incorrect Answer	16°F < Vessel Head Flange	will not
	Justification	16°F < Vessel Head Flange is temperature calculated if psia is calculated backwards	Tech Spec cooldown rate Exceeded 110°F

Recommendation: Amend the Answer Key to list 'Answer A' as the only correct answer.

References: ABB Steam Tables

Station Proposed Resolution:
Station agrees with the above candidate Comments, Justifications and Recommendation.

ATTACHMENT #3

QUESTION SRO 1

A loss of Hot Surge Tank level occurred.

The following conditions now exist:

- Operating in EOP-01, RPV Control
- RPV level is - 46 inches and lowering
- RPV pressure is 5 psig
- HPCS pump shaft has broken.
- LPCS pump is tagged out for motor replacement with motor removed
- RHR A pump is degraded
- EH12 has a Bus lockout
- No Alternate Injection Subsystems can be lined up
- The EOF is operational

As the Shift Manager, you would notify the Emergency Response Organization that entry into __(1)__ is required.

EOP actions are __(2)__ after the SAGs are entered.

	__(1)__	__(2)__
A.	SAG-1, Primary Containment Flooding	Continued
B.	SAG-1, Primary Containment Flooding	Exited
C.	SAG-2, RPV, Containment, and Radioactivity Release Control	Continued
D.	SAG-2, RPV, Containment, and Radioactivity Release Control	Exited

QUESTION SRO 1

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #		3
	Group #		
	K/A#		2.1.20
	Importance Rating		4.6

K&A: Ability to interpret and execute procedure steps.

Generic

Explanation: **Answer** B – SRO must know that with the given conditions, EOP-01 requires entry into SAG-1 is required. The shift manager is responsible for notifying the ERO that entry into SAGs is required. EOP actions are exited when SAGs are entered.

A – incorrect – the EOP Actions are discontinued when SAGs are entered.

C – incorrect – SAG-2 entry is not required. EOP actions are exited when SAGs are entered.

D incorrect - SAG-2 entry is not required.

Technical Reference(s): EOP-01 Bases Rev 3	Reference Attached: EOP-01 p 61

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OT-3410-01-A.3

Question Source:	Bank # Modified Bank # New	 x
Question History:	Previous NRC Exam	
Question Cognitive Level:	Memory or Fundamental Knowledge Comprehension or Analysis	 x
10 CFR Part 55 Content:	55.41 55.43	 b.5

Comments: Level of Difficulty = x - E. Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations. [10 CFR 55.43(b)(5)] Knowledge of diagnostic steps and decision points in the emergency operating procedures (EOP) that involve transitions to event specific subprocedures or emergency contingency procedures

Candidate Comments:

1. The question asks the required Shift Manager actions after the determination that adequate core cooling no longer exists and the required status of the Emergency Operating Procedures when transitioning to the Severe Accident Guidelines.
2. The answer key states SAG-1, Primary Containment Flooding is the required SAG actions.
3. The bases for EOP-1, step ALC-15 states that "Transition to the SAGs is completed by the Shift Manager notification to the ERO that Primary Containment Flooding is required".
4. The bases for SAG-1 states that "The requirements for Primary Containment flooding in the RPV Control, EOP-1, Level Power Control, EOP-1A, RPV Flooding, EOP04-4, flowcharts are directed through concurrent transitions to the Primary Containment Flooding, SAG-1 and RPV, Primary Containment, and Radioactivity Release Control, SAG-2 Flowcharts."
5. The SAG-1 bases clearly states "The RPV and Primary Containment Flooding guideline and the Primary Containment and Radioactivity Release Control guideline are entered and executed concurrently.
6. Answer 'D' SAG-2, RPV, Containment, and Radioactivity Release Control for part 1 and Exited for part 2 is a correct answer since we have "newly discovered technical information that supports a change in the answer key."
7. References, EOP-1 pg 61, SAG-1 bases pg 6 and SAG-2 pg 8 are attached.

Candidate Justifications for answers:

As the Shift Manager, you would notify the Emergency Response Organization that entry into __(1)__ is required.

EOP actions are __(2)__ after the SAGs are entered.

		__(1)__	__(2)__
A.	Incorrect answer	SAG-1, Primary Containment Flooding Correct - Primary Containment (SAG-1) and Radioactivity Release Control guideline (SAG-2) are entered and executed concurrently	Continued Incorrect -EOP Actions are discontinued when SAGs are entered.
B.	**Correct answer**	SAG-1, Primary Containment Flooding **Correct - Primary Containment (SAG-1) and Radioactivity Release Control guideline (SAG-2) are entered and executed concurrently**	Exited **Correct -EOP Actions are discontinued when SAGs are entered.**
C.	Incorrect answer	SAG-2, RPV, Containment, and Radioactivity Release Control Correct - Primary Containment (SAG-1) and Radioactivity Release Control guideline (SAG-2) are entered and executed concurrently	Continued Incorrect -EOP Actions are discontinued when SAGs are entered.
D.	**Correct answer**	SAG-2, RPV, Containment, and Radioactivity Release Control **Correct - Primary Containment (SAG-1) and Radioactivity Release Control guideline (SAG-2) are entered and executed concurrently**	Exited **Correct -EOP Actions are discontinued when SAGs are entered.**

Recommendation: Amend the Answer Key to list 'Answer D' as a correct answer also, in addition to 'Answer B'.

References: EOP bases and SAG bases, See attached

Station Proposed Resolution:
Station agrees with the above candidate Comments, Justifications and Recommendation. The question did not ask for all the required procedures. Therefore, the justification supports 2 correct answers.

RPV CONTROL

Effective Date: 6-13-12

Preparer: Dan Roniger / 12-16-11
Date

Approver: David Duesing / 6-6-12
Date

If RPV water level cannot be restored and maintained above -25 inches (Minimum Steam Cooling RPV Water Level) using only preferred injection sources and spray cooling cannot be established, alternate injection subsystems must be used, if not already in service. While earlier steps permitted use of the alternate injection subsystems, injection may not have been previously required if preferred subsystems were available. RHR loops in service to cool the suppression pool are required to be realigned at this step as adequate core cooling is not assured and RHR injection is required to establish adequate core cooling if possible.

To RO ATC will normally have the responsibility to maintain RPV level and to use injection systems as required.

Step ALC-15

If RPV water level cannot be restored and maintained above -25 inches (MSCRWL) with alternate injection subsystems and spray cooling cannot be established, primary containment flooding is required. If a primary system break exists, flooding the primary containment will backfill the RPV through the break. While flooding will be possible only if a primary containment fill source of sufficient capacity is available, immediate transfer to the SAGs is prescribed in anticipation of possible core geometry changes and so that instructions appropriate to the condition will be in effect when the necessary injection or fill systems are available.

Transition into SAGs is completed by the Shift Manager notification to the ERO that Primary Containment Flooding is Required. The ERO will notify the Control Room that SAGs are entered and to discontinue the actions of the EOPs. Until SAGs are entered by the ERO the actions the EOPs shall be continued in an effort to restore RPV level to provide Adequate core cooling as soon as possible. In the event the Emergency Response Organization cannot be activated the implementation of SAGs would fall upon the Control room staff. This entry condition contains an EPI flag as this condition also impacts the implementation of the Emergency Plan. If entry is made under these conditions the Shift Manager SHALL be notified of this condition.

EPI

EPI Flag

Transition into SAGs is completed by the Shift Manager notification to the ERO that Primary Containment Flooding is Required. The ERO will notify the Control Room that SAGs are entered and to discontinue the actions of the EOPs. Until SAGs are entered by the ERO the actions the EOPs shall be continued in an effort to restore RPV level to provide Adequate core cooling as soon as possible. In the event the Emergency Response Organization cannot be activated the implication of SAGs would fall upon the Control room staff.

<u>END OF SECTION</u>

PRIMARY CONTAINMENT FLOODING

Effective Date: 9-11-12

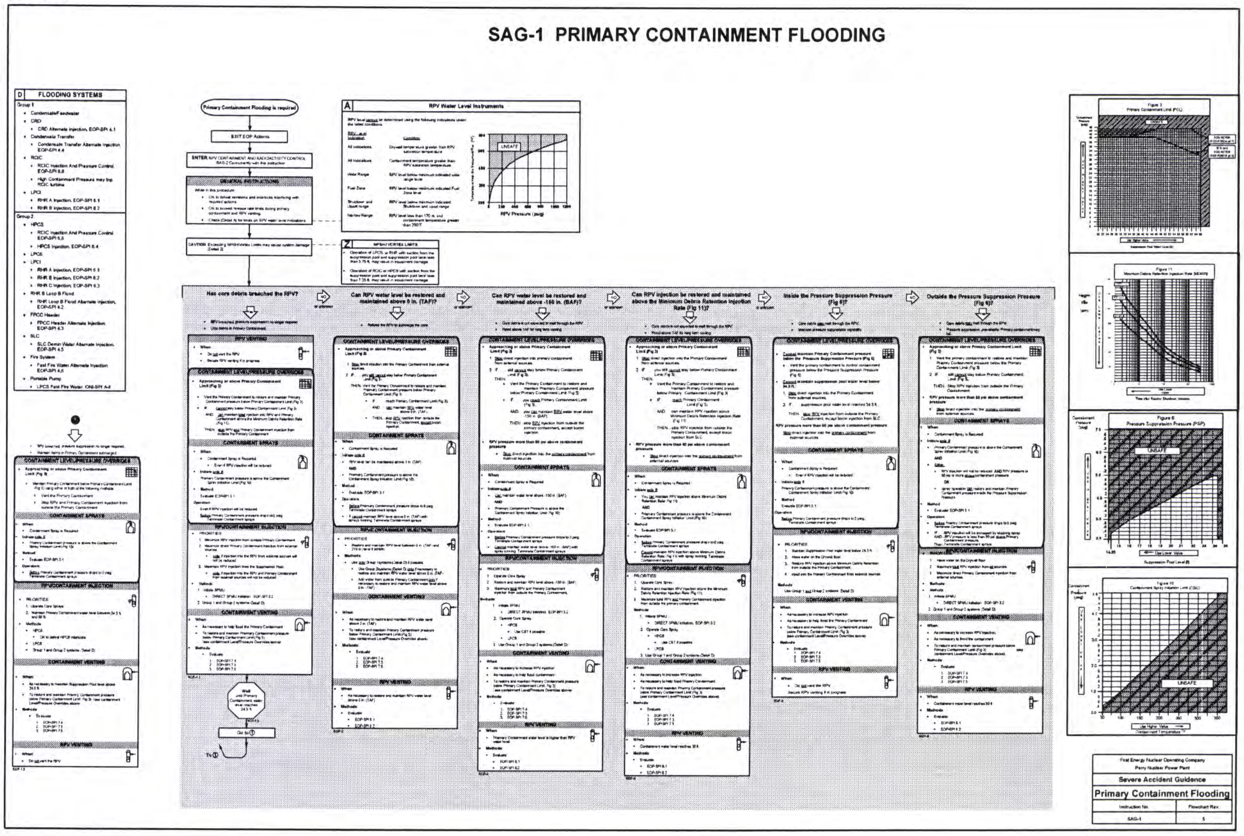
Preparer: Dan Roniger / 6-22-12
Date

Approver: David Duesing / 7-11-12
Date

5.1 RPV Control EOP-01

PSTG STEP: None

None



DISCUSSION

Primary Containment Flooding is required when:

- In the RPV Control flowchart, EOP-01, water level cannot be restored and maintained above - 25 inches (Minimum Steam Cooling RPV Water Level).
- In the RPV Control flowchart, EOP-01, water level cannot be restored and maintained above - 45 inches (elevation of the jet pump suction) with a core spray system injecting rated flow and at least one LPCI system is injecting inside the shroud.
- In RPV Flooding Flowchart, EOP-04-4, it is determined that core damage is occurring while RPV Flooding is in progress.
- In the Level Power Control flowchart, EOP-01A water level cannot be restored and maintained above – 25 inches (Minimum Steam Cooling RPV Water Level).

Under these conditions, the core cannot be adequately cooled using all available RPV injection sources and SAG entry is required. The Primary Containment is then flooded to submerge the core and preserve Primary Containment integrity. As Primary Containment water level rises, the RPV and Primary Containment are vented as necessary to facilitate flooding and to maintain Primary Containment pressure below applicable limits.

The requirements for Primary Containment flooding in the RPV Control, EOP-01, Level Power Control, EOP-01A, RPV Flooding, EOP-04-4, flowcharts are directed through concurrent transitions to the Primary Containment Flooding, SAG-1 and the RPV, Primary Containment, and Radioactivity Release Control, SAG-2 Flowcharts. Through these transitions, all parameter control paths transfer from the EOP flowcharts to the SAGs. Control of RPV water level to the Primary Containment Flooding SAG and RPV pressure, reactor power, and suppression pool water level, and all other Primary Containment, Secondary Containment, and Radioactivity Release Control functions to the RPV, Primary Containment and Radioactivity Release Control SAG.

The Primary Containment Flooding guideline defines an integrated strategy for flooding the RPV and Primary Containment.

The objectives of the integrated Primary Containment flooding strategy are to:

- Remove heat from the RPV
- Retain core debris in the RPV
- Maintain Primary Containment integrity
- Scrub fission products from Primary Containment atmosphere
- Prevent or minimize core-concrete interaction
- Submerge the core and core debris

These objectives are achieved through coordinated control of five functions RPV injection, Primary Containment injection, RPV venting, Primary Containment venting, and Primary Containment Spray. A matrix comparing the specific strategies employed in each step of Section RC/F is provided in Figure 1.

RPV and Primary Containment Injection

Even if the core cannot be adequately cooled and the core geometry changes, core debris will be retained in the RPV if the debris that contacts the RPV can be adequately cooled. If RPV water level can be maintained above the bottom of the active fuel, any debris relocated to the lower plenum will be submerged and adequately cooled. Injection into the RPV and Primary Containment from external sources is then maximized to flood Primary Containment to above the elevation of the top of the active fuel. Core spray systems are operated while flooding is in progress to provide some cooling to uncovered fuel remaining in the core region and to cool heated gases and internals in the RPV steam space.

The Primary Containment is flooded to fill the RPV, cool any fuel remaining in the core region, and, if an RPV breach occurs, to quench core debris discharged through the breach. A severe accident can be considered controlled only when all fuel and core debris is quenched and submerged. An accident in which the RPV is breached at an elevation below the top of the active fuel can be considered controlled only after Primary Containment has been flooded to above the top of the active fuel.

Flooding Primary Containment to above the elevation of core debris inside the RPV also provides some cooling to the debris, even if the water cannot enter the RPV, by heat transfer through the vessel wall. Perry's RPV skirt is not vented, however, and trapped noncondensibles can prevent water in the drywell from contacting parts of the lower head. The external cooling provided by flooding is therefore insufficient to ensure that core debris will be retained in the RPV, but may delay vessel failure by several hours and precludes in-core instrument thimble (instrument guide tube) failures.

RPV, CONTAINMENT, AND RADIOACTIVITY RELEASE CONTROL

Effective Date: 12-21-12

Preparer: Dan Roniger / 10-27-12
Date

Approver: David Duesing / 12-13-12
Date

STEP: Entry Conditions

START

GENERAL INSTRUCTIONS

- This procedure overrides all EOPs.
- Prepare Containment Vent Paths as soon as practical
  - Evaluate EOP-SPI 7.1

DISCUSSION

If Containment flooding is required, all parameter control paths transfer from the EOPs to the SAGs. All containment, secondary containment, RPV Pressure, Reactor Power, and radioactivity release control functions, transfer to the RPV, Containment and Radioactivity Release Control guideline, while RPV water level transfers to the Primary Containment Flooding guideline.

Primary Containment Flooding is required when:

- In the RPV Control flowchart, EOP-01, water level cannot be restored and maintained above - 25 inches (Minimum Steam Cooling RPV Water Level).
- In the RPV Control flowchart, EOP-01, water level cannot be restored and maintained above - 45 inches (elevation of the jet pump suction) with a core spray system injecting rated flow and at least one LPCI system is injecting inside the shroud.
- In RPV Flooding Flowchart, EOP-04-4, it is determined that core damage is occurring while RPV Flooding is in progress.

In the Level Power Control flowchart, EOP-01A water level cannot be restored and maintained above - 25 inches (Minimum Steam Cooling RPV Water Level).

Under these conditions, the core cannot be adequately cooled using all available RPV injection sources and SAG entry is required. The Primary Containment Flooding guideline and the RPV, Containment, and Radioactivity Release Control guideline are then entered and executed concurrently. Transferring all containment and radioactivity release control functions to a combined guideline in the SAGs provides the following advantages:

- Primary Containment Control functions can be properly coordinated with RPV and containment injection.
- Functions and conditions unique to severe accident conditions can be addressed without complicating the EOPs. Functions and conditions that are no longer of concern following emergency depressurization can be excluded.

LESSON PLAN	**Key Points, Aids, Questions/Answers**

E. S.T.A.R./BASIC

1. **S**top, **T**hink, **A**ct, **R**eview

   Use the Daily Plant Status if this is the first lesson of the day. Emphasize:

   1) Safety Message

   2) Human Performance Message

F. Student Objectives

1. State Terminal Objective.

2. Review Enabling Objectives.

II. PRESENTATION

A. Configuration, EOP-SAG Relationship, and Transition

1. Configuration:

   The SAGs consist of 3 guidelines: — *Guidelines*

SAG-1	Primary Containment Flooding
SAG-2	RPV, Containment and Radioactivity Release Control
TSGs	Technical Support Guidelines

2. Relationship

   a. The EOPs are entered and executed when a EOP entry condition is received. This is unchanged from revision 4 of the EPGs.

   b. The SAGs are entered and executed when primary containment flooding is required (essentially, when adequate core cooling cannot be restored and maintained) as defined in the applicable EOP guideline or contingency.

LESSON PLAN	Key Points, Aids, Questions/Answers
c. Once the SAGs are entered, the EOPs no longer apply and are exited.	
d. The EOPs will no longer apply once the SAGs have been entered because the configuration of the core may no longer be amenable to adequate cooling.	Cover CR 11-90255 as OE to illustrate that EOP flowcharts and their guidance do not apply while in SAGS even though SAGS make use of EOP SPIs.
3. Transition	Transition Points
a. SAG-1 and SAG-2 are entered and executed whenever primary containment flooding is required by the EOPs.	
b. Primary containment flooding is required when:	
1) In RPV Level Control (Non-ATWS), RPV water level cannot be restored and maintained above -25".	
2) In RPV Level Control (ATWS)	
a) Core Spray is below 6200 gpm	
b) No loop of RHR is injecting inside the shroud	
c) RPV level CANNOT be restored above -45in.	
3) In RPV Flooding, when core damage is occuring.	
B. Severe Accident Strategies	
1. Overview	Overview
a. Under the conditions requiring primary containment flooding, the core cannot be adequately cooled using all available RPV injection sources and SAG entry is required.	

LESSON PLAN	Key Points, Aids, Questions/Answers
b. The primary containment is then flooded to:	
1) Submerge the core.	
2) Preserve containment integrity.	
c. As primary containment water level rises due to the flooding, the RPV and primary containment are vented as necessary to:	
1) Facilitate flooding.	
2) Maintain primary containment pressure below applicable limits.	PSP/PCL
d. The coordination of RPV injection, primary containment flooding, and venting is known as the <u>integrated containment flooding strategy</u> (ICFS)..	
e. The objectives of the <u>integrated containment flooding strategy</u> are to:	ICFS Objectives
1) Remove heat from the RPV.	
2) Retain core debris in the RPV.	
3) Maintain primary containment integrity.	
4) Scrub fission products from the containment atmosphere.	
5) Prevent or minimize core-concrete interaction.	
6) Reestablish core submergence.	

ATTACHMENT #4

QUESTION SRO 8

The following conditions exist:

- The plant is operating at 98% power.
- Core flow is 103 Mlbs/Hr.
- OPRMs are INOPERABLE
- Alternate methods to detect and suppress thermal hydraulic instability oscillations have been initiated IAW 3.3.1.3 OPRM Instrumentation.

Reactor Recirculation Pump B then trips.

In accordance with ONI-C51, Unplanned Changes Reactor Power or Reactivity, the Unit Supervisor will direct ____.

Reference Provided:

A. inserting Cram Rods IAW FTI-B002, Control Rod Movements

B. restarting Recirc Pump B IAW SOI-B33, Reactor Recirculation

C. inserting a manual reactor scram IAW ONI-C71-1, Reactor Scram

D. shutting Recirc Pump B FCV IAW ONI-SPI G-2, Single Pump Operation

QUESTION SRO 8

	Level:	RO	SRO
	Tier #		1
	Group #		1
	K/A#	295001	2.1.23
	Importance Rating		4.4

1.0 K&A: ABILITY TO PERFORM SPECIFIC SYSTEM AND INTEGRATED PLANT PROCEDURES DURING ALL MODES OF PLANT OPERATION

Partial or Complete Loss of Forced Core Flow Circulation / 1 & 4

Explanation: **Answer A –** With the plant operating at 103 Mlbm/Hr/ a trip of 1 recirc pump would lower core flow to < 50% rated core flow, putting operation in the Controlled Entry Immediate Exit Region of the Backup Stability Protection (OPRM INOP) P/F map. The US would assess the situation and direct actions IAW the ONI-C51 Flow Chart - inserting Cram Rods to lower power to exit this region. (These are not Immediate Actions (from memory), these are actions directed by the Unit Supervisor.

B – Incorrect – Plausible since ONI-C51 indicates the preferred method to exit the CE/IE region is opposite the way it was entered.

C – Incorrect – Plausible since at lower core flows, a trip of 1 recirc pump will cause entry into the Manual Scram region.

D – Incorrect - Plausible since shutting the Recirc Loop Suction valve is required, but not the FCV.

Technical Reference(s): ONI-C51 Flow Chart Rev J and PDB-A6 rev 14	Reference Attached: ONI-C51 and PDB-A6 p 5

Proposed references to be provided to applicants during examination: **PDB-A06, Power Flow Map (modified)**

Learning Objective (As available): OT-COMBINED-B33-I

Question Source:	Bank # Modified Bank # New	 Peach Bottom 2008 # SRO-9
Question History:	Previous NRC Exam	Peach Bottom 2008
Question Cognitive Level:	Memory or Fundamental Knowledge Comprehension or Analysis	 x
10 CFR Part 55 Content:	55.41 55.43	 b.5

Comments: Level of Difficulty = x E. Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations. [10 CFR 55.43(b)(5)]

This 10 CFR 55.43 topic involves both 1) assessing plant conditions (normal, abnormal, or emergency) and then 2) selecting a procedure or section of a procedure to mitigate, recover, or with which to proceed.

Candidate Comments:

1. The question asks what actions are required for a trip of a Reactor Recirculation pump per ONI-C51. The stem of the question does not provide all the necessary information to answer the question. ONI-C51 FLOWCHART step C51-4 If While Executing step refers the operator to the Core Flow Caution. This step is applicable at all times the operator is using this procedure.
2. ONI-C51 FLOWCHART Core Flow Caution requires the operator to determine the actual core flow using core plate ΔP which is not provided.

   Core Flow — During single Recirculation pump operation the core flow instrument may not indicate properly. Actual core flow may be determined using:

   - Core plate ΔP and the curve in PDB-A0015.
   - A Core plate ΔP of 2.25 psid is approximately 42 Mblm/Hr core flow.
   - SPDS screen CFLOWV
3. The procedure directs that core flow be determined using core plate delta pressure which then outlines the correct course of action taken by the Unit Supervisor, core plate delta pressure was not provided.

Candidate Justifications for answers:

A. Incorrect answer –Actual core flow is indeterminate, inserting Cram Rods IAW FTI-B002, Control Rod is required when operating in the Controlled Entry / Immediate Exit region.

B. Incorrect answer - Actual core flow is indeterminate, restarting Recirc Pump B IAW SOI-B33, Reactor Recirculation may be a correct answer if core flow is known to not be in the Backup Stability Protection Regions – Two Loop Power – Flow Map Manual Scram Required region.

C. Incorrect answer -Actual core flow is indeterminate, inserting a manual reactor scram IAW ONI-C71-1, Reactor is a non-conservative action if core flow is >42 Mlbm/hr.

D. Incorrect answer - Actual core flow is indeterminate - shutting Recirc Pump B FCV IAW ONI-SPI G-2, Single Pump Operation is not directed by ONI-C51FLOWCHARTor by ONI-SPI G-2. ONI-SPI G-2 directs closure of the B Recirc Pump Suction valve

Recommendation: Delete question from exam, there is no correct answer.

References: ONI-C51, UNPLANNED CHANGE IN REACTOR POWER OR REACTIVITY, ONI-C51 Flowchart, See attached

Station Proposed Resolution:
Delete question from exam, there is no correct answer since this was "a question with an unclear stem that confused the applicants or did not provide all the necessary information."

UNPLANNED CHANGE IN REACTOR POWER OR REACTIVITY

Effective Date: 2-17-12

Preparer: Dan Roniger / 3-15-11
Date

Approver: Dave Duesing / 2-13-12
Date

TABLE OF CONTENTS

NOTE

OPRMs are operable for ONI-C51 until either Tech Spec 3.3.1.3 Condition A3 or B1 has been implemented. OPRMs are inoperable for ONI-C51 when Backup Stability Protection has been established as directed by Technical Specifications.

1.0 Operation in MANUAL SCRAM REQUIRED Region of the Power to Flow Map when the OPRMs are inoperable.

2.0 Operation in the CONTROLLED ENTRY/IMMEDIATE EXIT Region that is **NOT** in accordance with an approved instruction when the OPRMs are inoperable.

3.0 Operation in the IMMEDIATE EXIT Region of the Power to Flow Map when the OPRMs are Operable.

4.0 Unplanned changes in reactor power.

5.0 Flow control valve malfunction.

6.0 Recirculation Pump trip.

7.0 Unintentional Recirculation Pump downshift.

8.0 Recirculation loop jet pump flow mismatch exceeding Technical Specification limits.

9.0 Alarms (Not associated with planned power changes)

- SRM PERIOD SHORT
- ROD BLOCK SRM UPSC/INOP
- ROD BLOCK SRM DOWNSCALE
- ROD BLOCK IRM UPSCALE
- IRM A/E [(B/F), (C/G), (D/H)] UPSC TRIP/INOP
- ROD BLOCK IRM DOWNSCALE
- LPRM UPSCALE
- LPRM DOWNSCALE
- ROD BLOCK APRM UPSCALE
- APRM A/E [(B/F), (C/G), (D/H)] UPSC INOP/TRIP OPRM A/E TRIP
- ROD BLOCK APRM DOWNSCALE

- RCIRC A(B) MOTOR LOCKOUT
- RCIRC A(B) AUTO XFER INCOMPLETE
- RCIRC A(B) MG SET LOCKOUT
- MAST/FLUX CONT TROUBLE
- FCV A(B) MOTION INHIBITED
- FCV A(B) HPU INOP
- RECIRC A(B) FCV RUNBACK
- ROD BLOCK APRM RCIRC FLOW HI
- OPRM ALARM

10.0 Parameters

- Reactor power changes.
- Reactor level changes.
- MAIN GEN MWATTS changes.
- FEEDWATER FLOW & MAIN STEAM FLOW change.
- Recirculation loop flows mismatched.

1.0 Possible rod block or reactor scram due to high flux level.

2.0 Possible rod block due to low flux level.

3.0 Possible OPRM scram.

NOTE

The following are the Immediate Actions of ONI-C51 FLOWCHART:

- **IF** any of the following conditions exist,
  - More than one control rod has inadvertently scrammed
  - An unexplained power increase has occurred which CANNOT be quickly terminated. (Operator judgment)
  - **WHEN** OPRMs are inoperable,
    Entry into the manual scram region. (Normal feedwater temperature, loadline is
    >75% AND core flow is <42Mlbm/hour)
  - Mode switch in RUN **AND** NO Recirculation Pumps in operation
  - Power Oscillations are observed

  THEN SCRAM the Reactor

- **IF** OPRMs are operable,
  Entry into the Immediate Exit region. (Loadline is >75% AND core flow is <42Mlbm/hour),
  THEN REFER TO FTI-B0002 to insert cram rods to approximately 35% power.

- **IF** flow CAN NOT be controlled,
  THEN ARM AND DEPRESS the affected HPU SHUTDOWN switch.

NA 1.0

❑

Plant is in Mode 1	Plant is in Mode 2
THEN PERFORM the Immediate Actions of ONI-C51 FLOWCHART Revision J.	

NA	1.0	Plant is in Mode 1 or 2
❑		**THEN PERFORM** the remainder of actions of ONI-C51 FLOWCHART Revision J.

NA ❑ 1.1 **IF** a reactor scram occurs,
THEN EXIT this procedure.

NA 2.0 **IF** during Refueling an unplanned change in reactivity or flux level occurred,
THEN PERFORM the following:

❑ 2.1 **SUSPEND** core alterations.

❑ 2.2 **FULLY INSERT** all insertable control rods.

❑ 2.3 **EVACUATE** the Containment.

> NOTE
>
> Designated individuals for Containment Closure may be utilized to assist in closing Containment Airlock Doors per SOI-P53.

❑ 2.4 **EVALUATE** the need to:

- **ESTABLISH** Containment Integrity.
- **IMPLEMENT** the Containment Closure plan.

❑ 2.5 **DIRECT** Radiation Protection to initiate surveys.

NA 2.6 **IF** a reactor scram occurs,
THEN PERFORM the following:

❑ • **RESET** the scram.

❑ • **REMAIN** in this instruction.

❑ • **DO NOT PERFORM** ONI-C71-1 Reactor Scram.

Commitment - B00052
Commitment - B00078
Commitment - B00717
Commitment - B00916
Commitment - F01366
Commitment - F01368
Commitment - F01392
Commitment - F01393

The following records are completed/generated by this document:

<u>Quality Assurance Records</u>

None

<u>Non-Quality Assurance Records</u>

None

Rev. 25

1. Enhanced wording of decision step for 52 MLbm to clarify how to determine flow (600622985)(600637558).
2. Added action to refer to control rod movement sheet for actions needed for flow reduction.
3. Step wording for action at 894 MW corrected less than greater than (600663641).
4. ONI-C51 Limitation #1 added a note that it is from the ONI-SPI. (600547178).
5. Updated flowchart format to the writers guide for flowcharted procedure (600572035).

None

ONI-C51
Unplanned Change in Reactor Power or Reactivity

ENTRY CONDITIONS:
- As directed by ONI-C51
- Unplanned Operation in the Controlled Entry/ Immediate Exit Region of the power to flow map
- Uncontrolled change in reactor power
- Flow Control Valve malfunction
- Recirculation Pump trip
- Unintentional Reactor Recirculation Flow mismatch exceeding LCO that can not be corrected quickly
- Unintentional Recirculation Pump mismatch to slow speed
- Exceeding the MFOD Boundary Line

Immediate Actions

- IF any of the following conditions exist,
  - More than one control rod has inadvertently scrammed.
  - An unexplained power rise has occurred which CANNOT be quickly terminated. (Operator judgment)
  - WHEN OPRMs are inoperable, Entry into the manual scram region. (Normal feedwater temperature, loadline is >75% AND core flow is <42Mlbm/hour)
  - In Mode 1 OR 2 AND NO Recirculation Pumps in operation.
  - Power Oscillations are observed.

  THEN SCRAM the reactor
- IF flow CAN NOT be controlled, THEN ARM AND DEPRESS the affected HPU SHUTDOWN switch.

OPRMs

- IF OPRMs are operable, Entry into the Immediate Exit region (Loadline is >75% AND core flow is <42Mlbm/hour) THEN REFER TO FTI-B02 and INSERT cram rods to approximately 85% power.

C51-1 ☐ VERIFY Immediate actions are complete

C51-2 ☐ RECORD Entry Time

Time________________

C51-3 ☐

If While Executing the Following Steps:	
IF	THEN
NOT operating in Modes 1 or 2	EXIT ONI-C51 Flowchart actions
Power Oscillations are observed	1) SCRAM the reactor 2) EXIT ONI-C51 Flowchart actions

C51-4 ☐ **Oscillations**
MONITOR nuclear instruments for oscillations
SCRAM the reactor if oscillations are observed

EXECUTE CONCURRENTLY

C51-5 ☐

If While Executing the Following Steps:	
IF	THEN
Core Flow — Power is in either: Immediate Exit Region; Controlled entry / Immediate Exit Region	**Backup Stability** 1) IF both reactor Recirc pumps are running in Fast Speed AND region CAN be exited using Reactor Recirc Flow THEN RAISE Recirc Flow with flow control valves. **Current Step** **Rod Line** 2) REFER TO FTI-B02 and INSERT CRAM RODS. 3) IF CANNOT exit the affected Region, THEN SCRAM the Reactor
Thermal limits exceed administrative limits	**Rod Line** WITHIN 2 hours, RESTORE thermal limits within administrative limits per Reactor Engineering direction. PRIOR TO control rod withdraw, VERIFY SVI-C11-T1022, Rod Pattern Control System - Rod Withdrawal Limiter is current

2 hr
T.S. 2.1

C51-5 ☐ **MCPR**
DIRECT ONI SPI G-4, Power Verification

C51-6 ☐ Has an unplanned Reactor Power rise occurred? YES / NO

C51-6.1 ☐ **Current Step** / **Rod Line** / **Cram Rods** / **Oscillations**
REDUCE Power below initial power level
- LOWER Recirc Flow Refer to SOI-B33
- INSERT Control Rods
  - Cram Rods, Refer to FTI-B02
  - Reactor Engineering direction

C51-7 ☐ Is nuclear instrumentation failed? YES / NO

T.S. 3.3.1.1
T.S. 3.3.1.3
ORM 6.2.1
ORM 6.2.5

C51-7.1 ☐ BYPASS affected nuclear instrumentation
- REFER TO SOI-C51(APRM) and BYPASS failed APRM
- REFER TO SOI-C51(IRM) and BYPASS failed IRM
- REFER TO SOI-C51(SRM) and BYPASS failed SRM

C51-7.2 ☐ REFER TO SOI-C71 and RESET RPS

C51-7.3 ☐ VERIFY the following are reset:
- AFOL in control (H13-P680-4-E6)
- MAST/FLUX CONTROL TROUBLE (H13-P680-4-D6)

C51-7.4 ☐ RESTART HPUs shutdown due to Nuclear Instrument failure
REFER TO the following SOI-B33 sections as applicable:
- 7.20 HPU Emergency Shutdown
- 4.2 Hydraulic Power Unit (HPU) A Startup
- 4.3 Hydraulic Power Unit (HPU) B Startup

C51-8 ☐ EVALUATE entry into the following Off Normal Instructions
- ONI-C11-3 Control Rod Drop
- ONI-C34 Feedwater Flow Malfunction
- ONI-C85 Pressure Regulator Failure
- ONI-E12-1 Inadvertent Initiation Of ECCS/RCIC
- ONI-J11-1 GROSS FUEL CLADDING FAILURE
- ONI-N36 Loss Of Feedwater Heating
- ONI-N62 Loss of Main Condenser Vacuum

C51-9 ☐ REFER TO the following as applicable
- IOI-1, Cold Startup
- IOI-2, Hot Startup
- IOI-3, Power Changes

C51-10 ☐ WHEN All directed actions are complete THEN proceed

EXIT

C51-11 ☐

If While Executing the Following Steps:	
IF	THEN
Recirc pump A is tripped	GO TO Step C51-21
Recirc pump B is tripped	
Single Loop Operation is required	
Jet Pump failure is indicated	DIRECT SVI-B33-T1180, JET PUMP OPERABILITY
Jet Pumps are Inoperable	WITHIN 12 HOURS, VERIFY the plant in mode 3

12 hr
T.S. 3.4.3

C51-12 ☐ Have CRAM RODS been inserted? NO / YES

C51-13 ☐ **Thermal Limits**
REFER TO Control Rod Movement Cover Sheet
PERFORM applicable actions for lowering power.

C51-14 ☐ Is Thermal power greater than the MELLLA Boundary Line? YES / NO

C51-14.1 ☐ INSERT control rods to restore thermal power below the MELLLA Limit
REFER TO one or more of the following:
- Control Rod Movement Cover Sheet
- Cram Rods, Refer to FTI-B02
- Reactor Engineering Direction

T.S. 3.4.3

C51-15 ☐ CONFIRM Jet Pumps operation within established limits
A normal Plant shutdown is required if Jet Pumps are not operable
REFER TO one of the following:
- Applicable SPDS element for jet pump performance
  - JPDP
  - RFPERF
  - RSPERF
  - JPFLOW
- Jet Pump dp(s) on JET PUMP MONITORING PANEL (H13-P519)

C51-16 ☐ Is HPU Failure indicated? YES / NO

C51-16.1 ☐ REFER TO SOI-B33 and SHIFT HPU Subloops

C51-17 ☐ WHEN Recirc Flow mismatch exceeds Tech Spec limits THEN proceed

Owner:________________

C51-18 ☐ RECORD Loop parameters
DIRECT ONI-SPI G-2, Single Pump Operation

C51-19 ☐ BEFORE 2 hrs of operation with Recirc Flow mismatch exceeding Tech Spec limits
SINGLE LOOP OPERATION IS REQUIRED

2 hr **T.S 3.4.1**

C51-20 ☐ **1/2 Drive flow** / **Oscillations** / **Cram Rods**
WITHIN 2 hours PERFORM the following:
- REDUCE Load Line less than or equal to 100% per Reactor Engineering direction
- BALANCE Reactor Recirc flow when possible
- DECLARE Recirc loop with lower flow "Not in Operation"

Recirc Flow mismatch limits

Total Core Flow	Recirc Flow
> 72.8 Mlbm/hr total core flow	≤ 10.4 Mlbm/hr mismatch
≤ 72.8 Mlbm/hr total core flow	≤ 5.2 Mlbm/hr mismatch
1B33-R612A 1B33-R612B	A ___ Mlbm/hr - B ___ Mlbm/hr | C | ___ Mlbm/hr
Note: | C | Mlbm/hr is the absolute value of the Loop Flow Mismatch	

C51-21 ☐

If While Executing the Following Steps:	
IF	THEN
Both Reactor Recirc Pumps are in operation with Loop Flow Mismatch within limits	GO TO Step C51-12
Single Loop Operation is NOT desired	PERFORM a controlled plant shutdown to Mode 3 per IOI-3
Tripped Recirc Pump is to be restarted	RESTART Recirc Pumps, REFER TO SOI-B33
Feedwater is below 425.5°F curve of PDB-A011	WITHIN 6 hours REDUCE power to less than 23.8%, REFER TO IOI-3

6 hr

894 MWt
T.S. 3.2.2

C51-22 ☐ MONITOR feedwater temperature for being above the 425.5°F curve
REFER TO PDB-A011

C51-23 ☐ Is a Recirc pump tripped? YES / NO

1127 Mwt

C51-23.1 ☐ Is Thermal Power above 30%? NO / YES ________ %

1B33-R612A(B)

C51-23.2 ☐ Is running pump JP flow above 52 Mlbm/hr? NO / YES ________ %

T.S 3.4.1

C51-22.2.1 ☐ Within 15 Minutes of raising Recirc Flow OR Reactor Power, VERIFY reactor coolant temperature differentials within limits
DIRECT SVI-B33-T1170, Stratification Monitoring

C51-24 ☐ BALANCE Reactor Recirc flow when possible

C51-25 ☐ **Oscillations**
VERIFY Loop parameters within limits
DIRECT ONI-SPI G-2, Single Pump Operation

1 hr **T.S 3.4.1** **ORM 6.3.3**

C51-26 ☐ **1/2 Drive flow**
Within 1 hour, INSERT Control Rods to lower Power below 66.5% (2500 MWTh)

C51-27 ☐ RECORD the following in the Narrative Log
- Time Thermal Power is below 66.5% (2500 MWTh) the Narrative Log
- Data from ONI-SPI G-2

Time________________

EXECUTE CONCURRENTLY

C51-28 ☐ **Rod Line**
VERIFY operation within confines of the Single Loop Power To Flow Map
- Per Reactor Engineering direction
- PRIOR TO control rod withdraw, VERIFY SVI-C11-T1022, Rod Pattern Control System - Rod Withdrawal Limiter is current

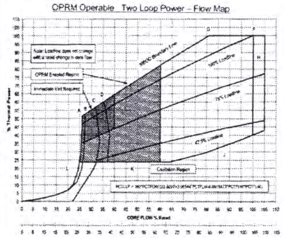
24 hr

C51-29 ☐ **Single Loop**
Within 24 hours, DIRECT SVI-B33-T1169, Establishing Limits For Single Loop Operation, to ESTABLISH single loop Setpoints

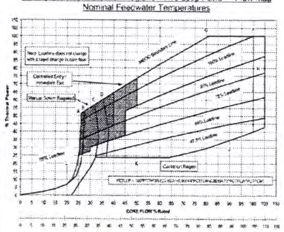
ONI-SPI G-2 Limitations

Parameter	Limit
Recirc Pump Amps	Less than or equal to 307 amps
Recirc Loop Flow	• Greater than 45%, when other pump is declared not in operation • Less than 103%
Recirc Pump Status Temperature	Less than or equal to 246 °F

OPRM Operable Two Loop Power – Flow Map



Backup Stability Protection Regions - Two Loop Power – Flow Map
Nominal Feedwater Temperatures



☐ **OPRMs** — OPRMs are operable for ONI-C51 until either Tech Spec 3.3.1.3 Condition A3 or B1 has been implemented. OPRMs are inoperable for ONI-C51 when Backup Stability Protection has been established as directed by Technical Specifications.

☐ **Oscillations** — The possibility of power oscillations in BWRs due to thermal-hydraulic instabilities and the consequences of such events is of concern to the NRC and industry. These oscillations have been observed to occur when operating in regions of high power and low flow. Since it is possible for some oscillations to grow to levels exceeding NRC safety limits in the order of a minute, immediate response to power oscillations is required.

The following are considered indications of core oscillation:
- Any APRM oscillates more than 10% from its lowest to its highest reading.
- Periodic LPRM UPSCALE or LPRM DOWNSCALE annunciators are received indicating out of phase oscillations which may NOT be observable on APRMs.
- SRM PERIOD meters indicate strong positive and negative swings (i.e., < 100 seconds positive to < 100 seconds negative). This indication of power oscillations is valid even with the SRMs fully withdrawn.

☐ **Core Flow** — During single Recirculation pump operation the core flow instrument may not indicate properly. Actual core flow may be determined using:
- Core plate dP and the curve in PDB-A0015.
- A Core plate dP of 2.25 psid is approximately 42 Mlbm/hr core flow.
- SPDS screen CFLOW

☐ **Backup Stability** — When using the Backup Stability Power to Flow Maps, the preferred method for exit is the opposite to the operation which caused entry (e.g., if a drop in Recirculation flow caused entry, exit by raising Recirculation flow). Exit from the CONTROLLED ENTRY / IMMEDIATE EXIT Region should not be delayed to start additional equipment (e.g., HPUs, RFPTs). Both raising Recirculation flow and control rod insertion may be used to exit in which case raising flow should precede control rod insertion.

☐ **Current Step** — The current step on the Control Rod Movement Sheet is the lowest numbered step of the approved sequence in which all the control rods in that step are not positioned at the withdrawal limit. If all the control rods in a step are positioned at the withdrawal limit, then the last step is the current step.

☐ **1/2 Drive flow** — Operation of one Recirculation loop with drive flow <1/2 of the drive flow in the other loop may result in the need to track fatigue usage on the Jet Pump. Drive Flow is read on RCIRC LOOP A FLOW, C51-R614 (blue pen) and RCIRC LOOP B FLOW, C51-R614 (red pen).

☐ **Single Loop** — While awaiting to enter Single loop operation the following apply:
- Tripped Reactor Recirculation Pump(s) should be started as soon as plant conditions permit to avoid thermal shock interlock conditions.
- The decision to operate in single loop should be made within 4 hours to allow time to implement single loop setpoints.
- Plant operations, within single loop limitations, may continue per IOI-3, Power Changes, while performing actions to establish Single Loop conditions or restoration of Reactor Recirculation Pumps.

☐ **Thermal Limits** — The Control Rod Movement Sheet contains actions to maintain thermal limits. If a plant transient causes a power reduction the guidance the of the Control Rod Movement Sheet still applies. If an action level has been exceeded (e.g. flow reduction below the requirement to insert control rods) the action needs to be completed as soon as practical.

☐ **Oscillations**
Lowering Core flow, feedwater temperature reduction, and Control Rod withdraws raise the likelihood of power oscillations.

☐ **MCPR**
Minimum Critical Power Ratio (MCPR) must be adjusted per the MCPR Limit With One Pressure Regulator Out of Service graph in IOI-3 when one Steam Bypass and Pressure Regulator (SB&PR) System pressure regulator is out of service.

☐ **Rod Line**
Control Rod insertion below the 80% load line may result in Reactor Recirculation Pumps Down shifting due to feedwater flow venturi inaccuracies.

☐ **Cram Rods**
Total core flow should not be raised without reactor engineering concurrence if cram rods were inserted.

FENOC	
Perry Nuclear Operating Plant	UNIT 1
OFF NORMAL INSTRUCTION	
UNPLANNED POWER CHANGES IN REACTOR POWER OR REACTIVITY	
Instruction No.	Flowchart Rev
ONI-C51	J

C51-4 ☐

If While Executing the Following Steps:

IF	THEN
Core Flow	**Backup Stability**
Power is in <u>either</u>: • Immediate Exit Region • Controlled entry / Immediate Exit Region	1) **IF** both reactor Recirc pumps are running in Fast Speed **AND** region **CAN** be exited using Reactor Recirc Flow, **THEN RAISE** Recirc Flow with flow control valves **Current Step** **Rod Line** 2) **REFER TO** FTI-B02 and **INSERT** CRAM RODS 3) **IF CANNOT** exit the affected Region, **THEN SCRAM** the Reactor
Thermal limits exceed administrative limits	**Rod Line** • **WITHIN** 2 hours, **RESTORE** thermal limits within administrative limits per Reactor Engineering direction • **PRIOR TO** control rod withdraw, **VERIFY** SVI-C11-T1022, Rod Pattern Control System - Rod Withdrawal Limiter is current

<u>2 hr</u>
<u>T.S. 2.1</u>

C51-5 ☐

MCPR

DIRECT ONI-SPI G-4, Power Verification

UNPLANNED CHANGE IN REACTOR POWER OR REACTIVITY

Effective Date: 2-17-12

Preparer: Dan Roniger / 3-15-11
Date

Approver: Dave Duesing / 2-13-12
Date

TABLE OF CONTENTS

1.0 ENTRY CONDITIONS

> NOTE
>
> OPRMs are operable for ONI-C51 until either Tech Spec 3.3.1.3 Condition A3 or B1 has been implemented. OPRMs are inoperable for ONI-C51 when Backup Stability Protection has been established as directed by Technical Specifications.

1.1 Operation in MANUAL SCRAM REQUIRED Region of the Power to Flow Map when the OPRMs are inoperable.

1.2 Operation in the CONTROLLED ENTRY/IMMEDIATE EXIT Region that is **NOT** in accordance with an approved instruction when the OPRMs are inoperable.

1.3 Operation in the IMMEDIATE EXIT Region of the Power to Flow Map when the OPRMs are Operable.

1.4 Unplanned changes in reactor power.

1.5 Flow control valve malfunction.

1.6 Recirculation Pump trip.

1.7 Unintentional Recirculation Pump downshift.

1.8 Recirculation loop jet pump flow mismatch exceeding Technical Specification limits.

1.9 Alarms (Not associated with planned power changes)

- SRM PERIOD SHORT
- ROD BLOCK SRM UPSC/INOP
- ROD BLOCK SRM DOWNSCALE
- ROD BLOCK IRM UPSCALE
- IRM A/E [(B/F), (C/G), (D/H)] UPSC TRIP/INOP
- ROD BLOCK IRM DOWNSCALE

- LPRM UPSCALE
- LPRM DOWNSCALE
- ROD BLOCK APRM UPSCALE
- APRM A/E [(B/F), (C/G), (D/H)] UPSC INOP/TRIP OPRM A/E TRIP
- ROD BLOCK APRM DOWNSCALE
- RCIRC A(B) MOTOR LOCKOUT
- RCIRC A(B) AUTO XFER INCOMPLETE
- RCIRC A(B) MG SET LOCKOUT
- MAST/FLUX CONT TROUBLE
- FCV A(B) MOTION INHIBITED
- FCV A(B) HPU INOP
- RECIRC A(B) FCV RUNBACK
- ROD BLOCK APRM RCIRC FLOW HI
- OPRM ALARM

1.10 Parameters

- Reactor power changes.
- Reactor level changes.
- MAIN GEN MWATTS changes.
- FEEDWATER FLOW & MAIN STEAM FLOW change.
- Recirculation loop flows mismatched.

2.0 AUTOMATIC ACTIONS

2.1 Possible rod block or reactor scram due to high flux level.

2.2 Possible rod block due to low flux level.

2.3 Possible OPRM scram.

3.0 IMMEDIATE ACTIONS

NOTE

The following are the Immediate Actions of ONI-C51 FLOWCHART:

- **IF** any of the following conditions exist,
  - More than one control rod has inadvertently scrammed
  - An unexplained power increase has occurred which CANNOT be quickly terminated. (Operator judgment)
  - **WHEN** OPRMs are inoperable,
    Entry into the manual scram region. (Normal feedwater temperature, loadline is >75% AND core flow is <42Mlbm/hour)
  - Mode switch in RUN **AND** NO Recirculation Pumps in operation
  - Power Oscillations are observed

  THEN SCRAM the Reactor

- **IF** OPRMs are operable,
  Entry into the Immediate Exit region. (Loadline is >75% AND core flow is <42Mlbm/hour),
  THEN REFER TO FTI-B0002 to insert cram rods to approximately 35% power.

- **IF** flow CAN NOT be controlled,
  THEN ARM AND DEPRESS the affected HPU SHUTDOWN switch.

NA 3.1

❑

Plant is in Mode 1	Plant is in Mode 2
THEN PERFORM the Immediate Actions of ONI-C51 FLOWCHART Revision J.	

4.0 SUPPLEMENTAL ACTIONS

NA 4.1 ❑

Plant is in Mode 1 or 2
THEN PERFORM the remainder of actions of ONI-C51 FLOWCHART Revision J.

NA 4.1.1 ❑ **IF** a reactor scram occurs,
THEN EXIT this procedure.

NA 4.2 **IF** during Refueling an unplanned change in reactivity or flux level occurred,
THEN PERFORM the following:

❑ 4.2.1 **SUSPEND** core alterations.

❑ 4.2.2 **FULLY INSERT** all insertable control rods.

❑ 4.2.3 **EVACUATE** the Containment.

NOTE
Designated individuals for Containment Closure may be utilized to assist in closing Containment Airlock Doors per SOI-P53.

❑ 4.2.4 **EVALUATE** the need to:

- **ESTABLISH** Containment Integrity.
- **IMPLEMENT** the Containment Closure plan.

❑ 4.2.5 **DIRECT** Radiation Protection to initiate surveys.

NA 4.2.6 **IF** a reactor scram occurs,
THEN PERFORM the following:

- ☐ • **RESET** the scram.
- ☐ • **REMAIN** in this instruction.
- ☐ • **DO NOT PERFORM** ONI-C71-1 Reactor Scram.

5.0 REFERENCES

Commitment - B00052

Commitment - B00078

Commitment - B00717

Commitment - B00916

Commitment - F01366

Commitment - F01368

Commitment - F01392

Commitment - F01393

6.0 RECORDS

The following records are completed/generated by this document:

Quality Assurance Records

None

Non-Quality Assurance Records

None

7.0 SCOPE OF REVISION

Rev. 25

1. Enhanced wording of decision step for 52 MLbm to clarify how to determine flow (600622985)(600637558).
2. Added action to refer to control rod movement sheet for actions needed for flow reduction.
3. Step wording for action at 894 MW corrected less than greater than (600663641).
4. ONI-C51 Limitation #1 added a note that it is from the ONI-SPI. (600547178).
5. Updated flowchart format to the writers guide for flowcharted procedure (600572035).

8.0 ATTACHMENTS

None

ATTACHMENT #5

QUESTION SRO 14

The plant was operating at rated power.
A manual Rx scram was inserted due to a loss of main condenser vacuum.
The transient resulted in the following conditions:

- An ATWS is in progress
- MSIV's are shut
- RPV water level is 65 inches and stable
- RPV pressure is 960 psig and stable
- 2 SRV's are open
- Suppression Pool Temperature is 116°F and rising slowly
- Suppression Pool Level 18.1' and lowering due to a leak
- The margin to exceeding HCL is 3°F

What is the action that the Unit Supervisor would order first?

A. Open an additional SRV to lower RPV pressure

B. Open MSIV's IAW EOP-SPI 9.2, Opening MSIV's

C. Transition to EOP 4-2, Emergency Depressurization

D. Anticipate Emergency Depressurization IAW EOP-02, Containment Control

QUESTION SRO 14

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #		1
	Group #		1
	K/A#	295030	EA2.02
	Importance Rating		3.9

K&A: Ability to determine and/or interpret the following as they apply to Low Suppression Pool Water Level: Suppression pool temperature

Low Suppression Pool Wtr Lvl / 5

Explanation: **Answer A** – With SP temp rising and SP level lowering, the margin to HCL is shrinking. IAW EOP-01A, the SRO would direct the RO to lower RPV pressure by opening an additional SRV to maintain margin to HCL.

B – Incorrect – With the main condenser not available, opening the MSIV's would not be appropriate.

C – Incorrect – ED would only be performed if SP temperature could not be restored and maintained below HCL.

D – Incorrect – Anticipating ED is not appropriate during an ATWS.

Technical Reference(s): EOP-1A Chart Rev D	Reference Attached: EOP-1A Chart (partial)

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OT-3402-04B-D.1

Question Source:	Bank # Modified Bank # New	 x
Question History:	Previous NRC Exam	
Question Cognitive Level:	Memory or Fundamental Knowledge Comprehension or Analysis	 x
10 CFR Part 55 Content:	55.41 55.43	 b.5

Comments: Level of Difficulty = x E. Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations. [10 CFR 55.43(b)(5)]

This 10 CFR 55.43 topic involves both 1) assessing plant conditions (normal, abnormal, or emergency) and then 2) selecting a procedure or section of a procedure to mitigate, recover, or with which to proceed

Candidate Comments:

1. The answer key states answer A. Open additional SRV to lower RPV pressure. This answer is correct in accordance with EOP-1A, Step LPC/P-
   Answer C is also correct.
   - Though the EOP-SPI Supplement Figure 4 was not given to the students, if one were to plot the given conditions on Figure 4, HCL would already be in the UNSAFE region requiring immediate Emergency Depressurization (ED) per EOP Bases, EOP-1A, EOP-2 and Hardcards (OAI-1703).
   - The stem of the question notes a rapidly deteriorating HCL challenge. HCL is normally attempted to be maintained between 5-10 F if ordered. This information was not in the stem. The EOP flowcharts were also not available to the students which are not required to be known by memory. The stem states a 3 F margin to HCL which is very narrow as seen in simulator scenarios. Because this band is not being maintained, the deteriorating challenge can be immediately deduced.
   - The deteriorating challenge also includes a lowering Suppression Pool Level due to a pipe break of unknown size or unknown level change rate. Figure 4, Heat Capacity Limit, shows that as Suppression Pool Level lowers, HCL is further challenged as there is less water to absorb heat causing heatup rate to raise more.
   - Due to the lowering level in the suppression pool, another challenge is immediately deduced that also leads to Emergency Depressurization per EOP-02, Step SPL-6. OAI-1703 Hardcard states "ED required at <u>OR</u> before 14.25 feet".
   - Transient Strategies and Mitigating Actions (PYBP-POS-0030) state that suppression pool heatup rate is approximately 3F/minute/SRV. With 2 SRV's open, this is a 6F/minute heatup rate, accelerated by 3. above, that would cause HCL to be violated in less than 30 seconds (if not already violated per 1. above). A decision that HCL cannot be maintained and that Emergency Depressurization is required can be made at that point also and allow proceeding into Emergency Depressurization from below the hold box of step STC-5 or within LPC/P-2 (EOP-1A Pressure Control).

Candidate Justifications for answers:

A. Correct answer - With SP temp rising and SP level lowering, the margin to HCL is shrinking. IAW EOP-01A, the SRO would direct the RO to lower RPV pressure by opening an additional SRV to maintain margin to HCL.
B. Incorrect answer – With the main condenser not available, opening the MSIV's would not be appropriate.
C. Correct answer – The deteriorating challenge can be immediately deduced. Due to the lowering level in the suppression pool, another challenge is immediately deduced that also leads to Emergency Depressurization per EOP-02.
D. Incorrect answer – Anticipating ED is not appropriate during an ATWS.

Recommendation: Modify the answer key to show two correct answers.

References: Required portions of EOP-SPI Supplement, EOP Bases, PYBP-POS-0030, and OAI-1703 are attached.

Station Proposed Resolution:
The station staff does not support the candidate's basis and this question does, in fact, have only one correct answer.

Step LPC/P-2 states that if HCL cannot be maintained below the limits of Figure 4 then the operator is directed to maintain RPV pressure below the HCL limit. As stated in the candidate's response the normal pressure band directed is 5 to 10 °F below HCL. The information given in the stem indicates that there is currently only 3 °F margin to HCL therefore the first appropriate action directed by the Unit Supervisor is to open additional SRVs to restore within the assigned pressure band. It is true that a lowering suppression pool level due to a leak adds another layer of complexity however Emergency Depressurization due to lowering suppression pool level is required before 14.25 feet and the stem provides information that this level is not currently challenged.

In addition EOP Step STC-5 requires Emergency Depressurization when HCL cannot be restored and maintained below the limits of Figure 4. EOP Bases defines Restore and Maintain as taking actions using available systems to restore a parameter to within a desired band or condition and maintain it there which includes actions to bring on additional equipment. Definition includes a note that states a parameter can be considered to be restored and maintained even if the parameter exceeds the limit multiple times as long as the majority of the time is spent within the limit.

With the conditions stated in the stem there are additional actions that can be taken to restore and maintain RPV pressure within the limits of HCL, Figure 4 and should be directed first by the Unit Supervisor which requires opening an additional SRV to lower RPV pressure.

<table>
<tr><td rowspan="1">PERRY NUCLEAR POWER PLANT</td><td colspan="2">Instruction Number:
EOP-SPI SUPPLEMENT</td></tr>
<tr><td rowspan="2">Title:
EOP-SPI SUPPLEMENT</td><td colspan="2">Use Category:
In-Field Reference</td></tr>
<tr><td>Revision:
3</td><td>Page:
1 of 14</td></tr>
</table>

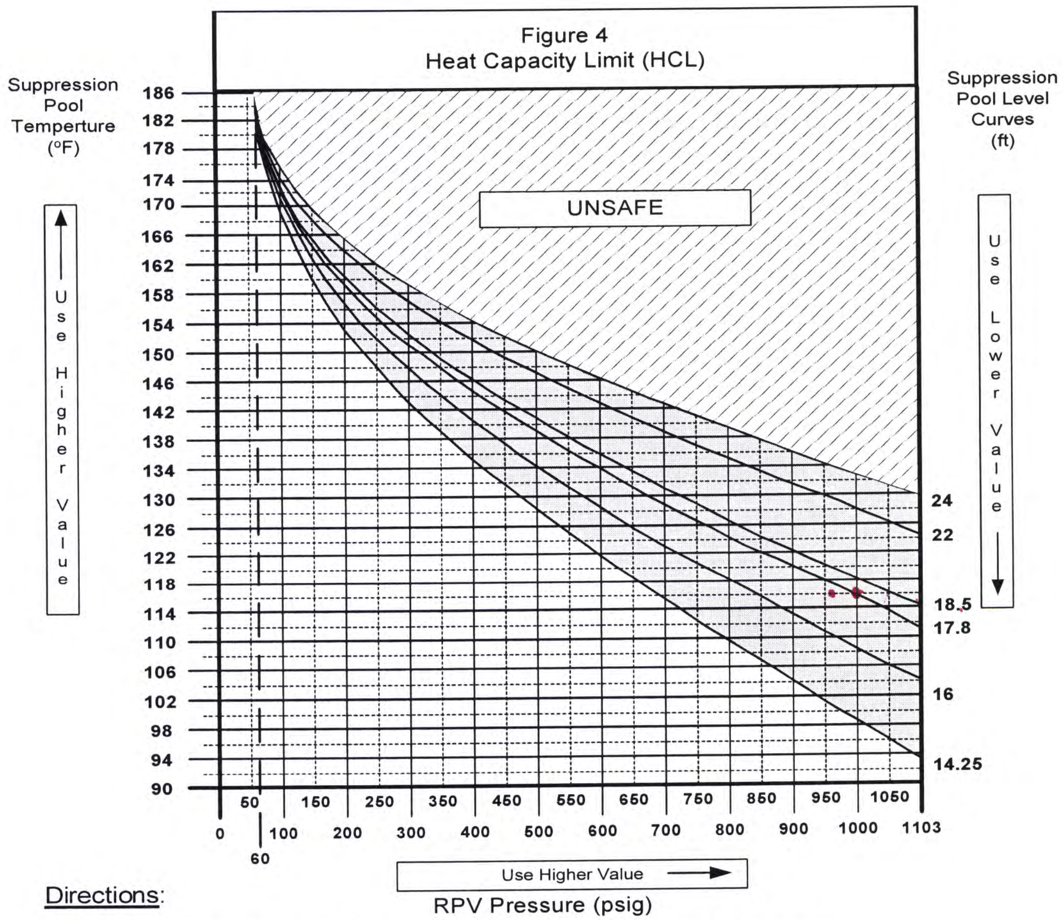
EOP-SPI SUPPLEMENT

Effective Date: 7/28/2012

Preparer: Dan Roniger / 7-28-12
Date

Approver: Dan Stanley. / 7-28-12.
Date

6.0 FIGURE #4 HEAT CAPACITY LIMIT



Directions:

1.0 **IDENTIFY** RPV Pressure on the horizontal axis of the figure.

2.0 **IF** the value falls between marked lines on the figure,
THEN USE then <u>higher</u> value.

3.0 **IDENTIFY** Suppression Pool Temperature on the vertical axis of the figure.

4.0 **IF** the value falls between marked lines on the figure,
THEN USE then <u>higher</u> value.

5.0 **SELECT** the Suppression Pool Level Curve that corresponds to current Suppression Pool Level.

6.0 **IF** Suppression Pool Level falls between the marked curves,
THEN USE the next lower curve.

7.0 **IDENTIFY** the point formed by the intersection of the two values with respect to the Suppression Pool Level Curve selected.

8.0 **IF** the selected value is above the Suppression Pool level Curve selected,
THEN HCL is exceeded.

PERRY NUCLEAR POWER PLANT	Instruction Number: EOP-01A	
Title: LEVEL POWER CONTROL	Use Category: In-Field Reference	
	Revision: 4	Page: 1 of 85

LEVEL POWER CONTROL

Effective Date: 9-21-12

Preparer: Dan Roniger / 8-28-12
Date

Approver: David Duesing / 8-29-12
Date

5.5.2 Step LPC/P-2

PSTG STEP: RC/P 1st IWE 2nd IWE 3rd IWE

If while executing the following steps:

- A high drywell pressure ECCS initiation signal **1.68 psig** (drywell pressure which initiates ECCS) exists, prevent injection from those LPCS and LPCI pumps not required to assure adequate core cooling prior to depressurizing below their maximum injection pressures.
- EMERGENCY Depressurization is anticipated and it can be determined that the reactor will remain shut down, rapidly depressurize the RPV with the main turbine bypass valves irrespective of the resulting cool down rate.
- EMERGENCY Depressurization is or has been required; enter **Emergency Depressurization**.
- RPV water level cannot be determined, enter **RPV Flooding**.

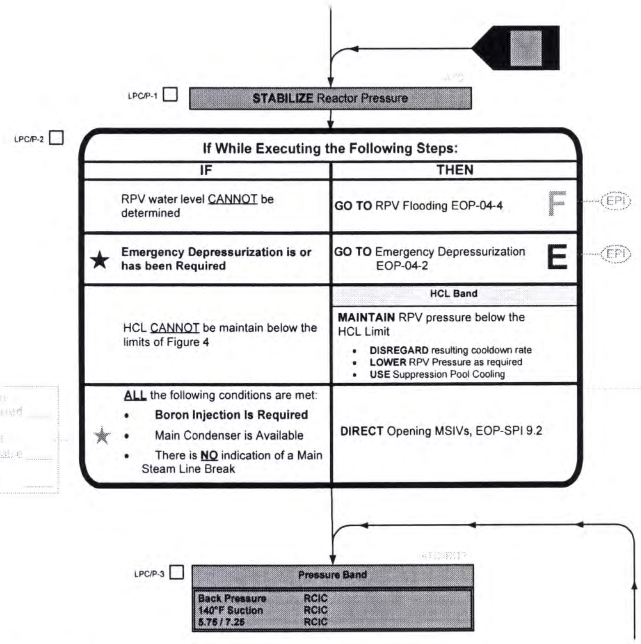
If while executing the following steps:

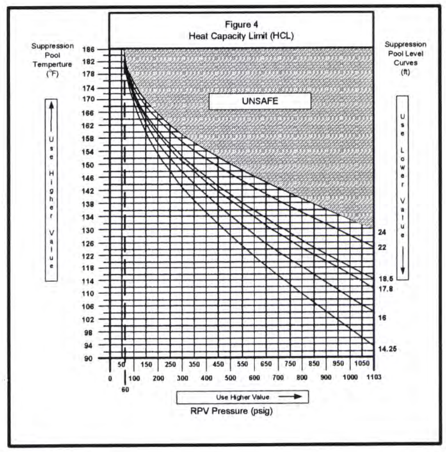
- Suppression pool temperature cannot be maintained below the **Heat Capacity Limit, Figure 4**, maintain RPV pressure below the Limit, exceeding **100°F/hr** (RPV cooldown rate LCO) cooldown rate if necessary.
- Steam Cooling is required, enter Steam Cooling.

If while executing the following steps:

- Boron Injection is required, and
- The main condenser is available, and
- There has been no indication of a steam line break,

open MSIVs, bypassing low RPV water level interlocks if necessary, to re-establish the main condenser as a heat sink.





DISCUSSION

These If While Executing steps are applicable throughout the performance of the remainder of RPV Pressure Control.

RPV water level cannot be determined

If RPV water level cannot be determined, adequate core cooling by submergence cannot be verified. The RPV is therefore flooded according to RPV Flooding, EOP-04-4 to assure that adequate core cooling is established and maintained. Since RPV flooding requires the control of RPV pressure in a manner which would conflict with the direction provided in the Reactor Pressure Control, RPV pressure control also transfers to RPV Flooding, EOP-04-4. Actions specified in RPV Flooding, EOP-04-4 will depressurize the RPV.

Offscale RPV water level indications do not necessarily require entry of RPV Flooding, EOP-04-4 provided the indications are believed to be valid.

This transition to RPV Flooding, EOP-04-4 contains an EPI flag as this condition also impacts the implementation of the Emergency Plan. If the transition to RPV Flooding, EOP-04-4 is made the Shift Manager SHALL be notified of this condition.

EPI

EPI Flag

Emergency Depressurization is or has been Required

Emergency Depressurization, EOP-04-2 provides appropriate guidance for rapidly depressurizing the RPV and preventing RPV repressurization. Once Emergency Depressurization is required, Emergency Depressurization, EOP-04-2 remains in effect until the EOPs are exited. If RPV Control must be re-entered during this time, the wording of this override ("Emergency Depressurization is or has been required") returns control of RPV pressure to Emergency Depressurization, EOP-04-2.

Placement of the Emergency Depressurization, EOP-04-2 flowchart over the pressure leg prohibits the operator form using the guidance of RPV Control, EOP-01 Pressure control leg.

This transition to Emergency Depressurization, EOP-04-2 contains an EPI flag as this condition also impacts the implementation of the Emergency Plan. If the transition to Emergency Depressurization, EOP-04-2 is made the Shift Manager SHALL be notified of this condition.

EPI

EPI Flag

Emergency Depressurization is Anticipated

This is a conditional action based upon the Reactor Being shut down under all conditions and does not apply to LEVEL POWER CONTROL, EOP-01A.

HCL cannot be maintain below the limits of Figure 4

The Heat Capacity Limit (Figure 4) is a combination of the EPG Heat Capacity Temperature Limit and Heat Capacity Level Limit Curves. The curves are combined to simplify their use.

The Heat Capacity Temperature Limit (HCTL) is defined to be the highest suppression pool temperature at which initiation of RPV depressurization will not result in exceeding 185°F before the rate of energy transfer from the RPV to the containment is within the capacity of the containment vent. This temperature is a function of RPV pressure, and the limit is used to prevent failure of the containment or equipment necessary for the safe shut down of the plant.

The Heat Capacity Level Limit (HCLL) is defined to be two feet above the elevation of the horizontal vents.

This water level is a function of the margin to the HCTL. The HCLL is used in conjunction with the HCTL to prevent failure of the Primary Containment or equipment necessary for the safe shut down of the plant and to prevent loss of the pressure suppression function of the Primary Containment.

Control of suppression pool temperature is addressed in Primary Containment Control, EOP-02. If the actions being taken to limit suppression pool temperature increase are inadequate or not effective, RPV pressure must be reduced in order to remain below the HCL. Therefore, actions in the Reactor Pressure Control must accommodate these requirements. Failure to do so may lead to failure of Containment or loss of equipment necessary for the safe shut down of the plant.

The normal cooldown rate limit may be exceeded to the extent necessary to maintain RPV pressure below the HCL. If RPV pressure cannot be maintained below the HCL, emergency RPV depressurization will be required, possibly resulting in an even more rapid cooldown.

Guideline **HCL Band** states:

When maintaining below the Heat Capacity Limit (HCL); Suppression Pool Temperature is used as the reference for the required RPV pressure. Maintaining a Suppression Pool Temperature 5°F to 10°F below the HCL Limit allows for control without excessive depressurization of the RPV.

Suppression Pool Temperature is used based upon the linearity of the temperature rise when the suppression pool is heated.

Steam Cooling is Required

This is a conditional action based upon the Reactor Being shut down under all conditions and does not apply to LEVEL POWER CONTROL, EOP-01A.

PRIMARY CONTAINMENT CONTROL

Effective Date: 6-13-12

Preparer: Dan Roniger / 2-1-12
Date

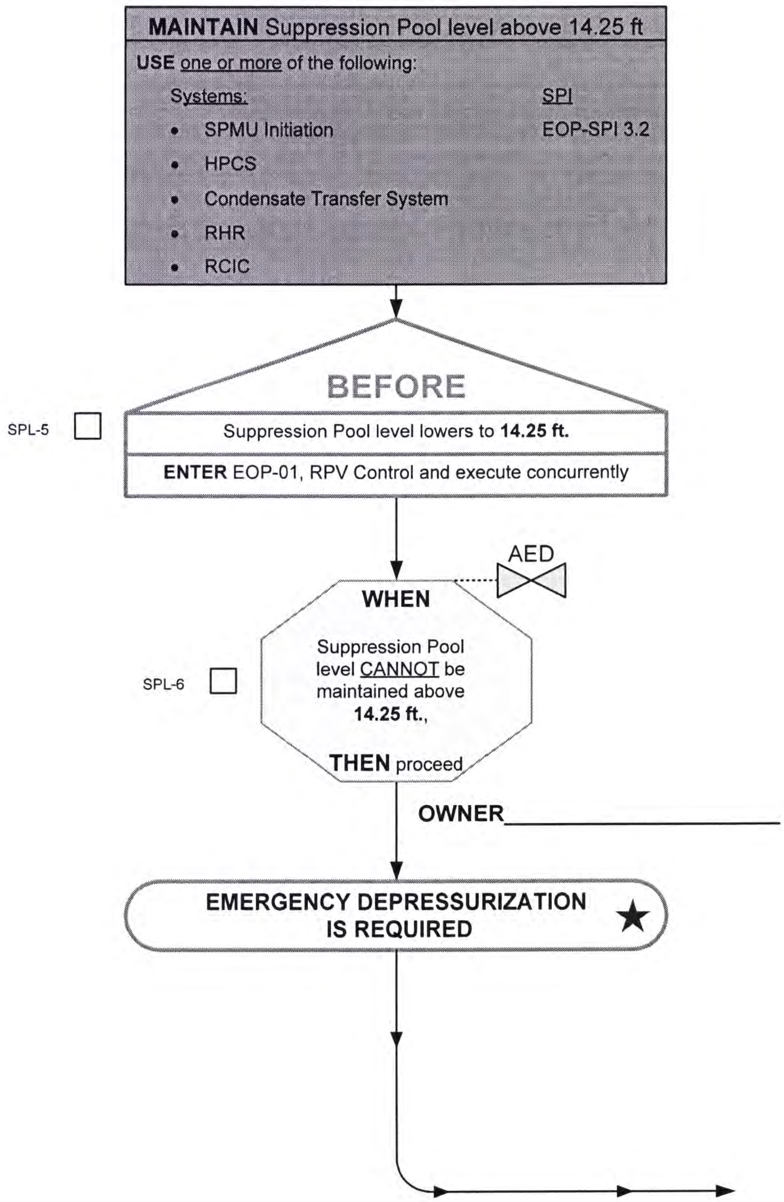
Approver: David Duesing / 6-6-12
Date

5.8.4 Steps SPL-5 and SPL-6

PSTG STEP: SP/L-2.1 (cont)

(1) Before Suppression Pool water level drops to **14.25 ft.** (2 ft. above the elevation of the top of the horizontal vents), enter RPV Control at Step RC 1 and execute it concurrently with this procedure.

(2) If Suppression Pool water level cannot be maintained above **14.25 ft.** (2 ft. above the elevation of the top of the horizontal vents), EMERGENCY DEPRESSURIZATION IS REQUIRED.



HARDCARDS

Operations Administrative Instruction

Effective Date: 10-8-12

Preparer: Dan Roniger / 9-28-12
Date

Approver: David Duesing / 10-4-12
Date

ATTACHMENT 5: LEVEL POWER CONTROL EOP-01A HARDCARD

Page 1 of 1

OAI-1703 *Hardcard Revision 1*

EOP TRANSIENT OVERSIGHT CHECKLIST

Level Power Control EOP-01A

POWER CONTROL

- Power control actions in progress (B33, SLC, Rods)
- Initiate SLC > 4% power OR Suppression Pool Temperature > 110°F

REACTOR LEVEL CONTROL

- Inhibit ADS
- If <4% power, Terminate and Prevent ECCS Injection (E12, E21, E22) (Level Band -25 in. to 219 in.)
- If >4% power, Terminate and Prevent FW and ECCS Injection (E12, E21, E22 and FW) until <100 in. (Level Band -25 in. to 100 in.)
- If >4% power, and >110°F in the Suppression Pool, Terminate and Prevent FW and ECCS Injection (E12, E21, E22 and FW) until either <4% power, < 16.5 in. or containment challenge terminated, (Level Band -25 in. to the level to which level was lowered (100 in. max level)
- Monitor and control Hydrogen
  - Start H2 Ignitors
  - Start H2 Analyzers
- ED required when RPV CANNOT be restored above < -25 in.

PRESSURE CONTROL

- Stabilize Pressure (Reference Pressure Control Hardcard / IOI-5, as necessary)
  - As last resort, Close MSIVs if uncontrolled pressure drop raises power or cooldown limit is threatened
  - Maintain RPV pressure less than HCL, exceed cool down rate limits if necessary
- Cooldown when reactor is shutdown and no boron has been injected
  - Within 30 Min of an MSIV closure (Maximize Cool down)
  - Entry of TLAC condition. Maximize Cool down while keeping RCIC operation.
  - When Conditions permit a controlled cool down
- If ED is required: Evaluate SLC initiation due to suppression pool heatup, Terminate and Prevent injection, Open 8 SRVs, Recommence injection slowly when <MSCP to level band that existed prior to ED

ATTACHMENT 7: PRIMARY CONTAINMENT CONTROL EOP-02 HARDCARD
Page 1 of 1

OAI-1703 Hardcard Revision 1

EOP TRANSIENT OVERSIGHT CHECKLIST

Primary Containment Control EOP-02

Drywell Temperature

- Restore Drywell cooling
- Enter EOP-01 before 330ºF
- ED at OR before 330ºF

Suppression Pool Temperature

- If not required for adequate core cooling, initiate Suppression Pool Cooling (10 Min MSIV Closure, 30 Min DBA LOCA)
- Prior to 110ºF, Enter EOP-01, If the Reactor is critical initiate SLC
- Initiate SPMU prior to 185 ºF
- ED when HCL cannot be restored and maintained below the limit.

Suppression Pool Level

NOTE: Top of weir wall is 24.2 ft.

- Before reaching 14.25 ft. OR 24,5 ft Enter RPV Control, EOP-01
- ED required at OR before 14.25 ft. OR 24.5 ft. high level

Containment Pressure

- If both of the following conditions are met, initiate spray prior to PSP limit
  - Containment Pressure above the CSIL, and
  - RHR is NOT required for adequate core cooling by continuous injection
- Terminate Spray prior to 0 psig
- ED required at OR before PSP Limit
- At 15 psig, prepare to vent and evaluate vent criteria

Containment Temperature

- Restore Containment Cooling
- If both of the following conditions are met, initiate spray prior to 185ºF
  - Containment Pressure above the CSIL, and
  - RHR is NOT required for adequate core cooling by continuous injection
- ED required at OR before 185ºF

OPERATOR MEMORY ITEMS

Effective Date: 9-17-12

Approved: Steve Benedict / 9-6-12
Supt., Nuclear Operations Date

ATTACHMENT 1: Licensed Operator Knowledge List
Page 3 of 16

4. Emergency Depressurize based on HCL Curve per EOP-01A (7%)

   This operator action models the proceduralized direction to emergency depressurize once the Heat Capacity Limits can not be met. This operator action presents a conservative modeling of this action for containment preservation actions as directed per the EOP flowcharts. Sequences applicable for making this a risk significant operator action typically involve applications where a loss of offsite power has occurred, RCIC is in operation and HPCS is unavailable. Failure to depressurize would not allow low pressure injection or alternate injection sources to be utilized.

5. Recover form a Loss of M23/24 HVAC (7%)

   This operator action represents recovery from a loss of safety related HVAC following an event. With the loss of HVAC to the safety related electrical switchgear rooms, eventually heating of these areas could result in excessive heating in the local panels that could result in the tripping of thermal overloads associated with the 480 V rotating equipment, eventually resulting in support system unavailability.

6. Emergency Depressurize when no High Pressure Injection Available per EOP-01 (5%)

   This operator action models the action to emergency depressurize the RPV during an event in which all high pressure injection capability is not available or has been lost for injection. Without a depressurized reactor, low pressure injection and alternate injection capability would be unable to provide an injection source.

7. Initiate Suppression Pool Cooling per EOP-02 (4%)

   This operator action models the action of the operator to place Suppression Pool Cooling in operation following an event in which heat is being directly added to the pool. In the events where RCIC in operation and the main condenser is not available, failure to initiate suppression pool cooling could lead to the procedural loss of RCIC due to Heat Capacity Limits. Longer term failure could result in pressurization of the containment and the eventual need for venting.

TRANSIENT STRATEGIES AND MITIGATING ACTIONS

Effective Date: 9-17-12

Approved: Steve Benedict / 9-6-12
Date

ATTACHMENT 2: THUMBRULES

Page 1 of 4

NOTE
The below thumb rules are approximations only. Additionally, as RPV pressure is lowered, the below thumb rules become much less useful to the operator (i.e. - there is a large delta between SRV flow rate at 1000 psig in the reactor versus SRV flow rate at 200 psig in the reactor.

Thumb Rules (at rated conditions):

1.0 SRV and Bypass Valve Impact

- Suppression Pool heatup rate is approximately 3°F per minute per SRV.
- 1 Mlbm /Hr is approximately equal to ~2000 gpm = about 5% power.

  (Thought process: at 100% power feedwater flow is ~38000 gpm and steam flow is ~ 16.5 Mlbm / hour, basically 2000 gpm for each 1 Mlbm / hour).
- 1 SRV open flow rate is approximately equal to 1 Mlbm / hr (Actual value is about 830 Klbm/hr).
- 1 SRV open is approximately equal to ~ 5% power.
- 1 bypass valve is approximately equal to 4% power
  (Bypass valve capacity = ~28% power for all bypass valves,
  therefore 28/7 = 4% power each = 1600 gpm feedflow needed).
- In Summary:
  - 1 SRV open = 5% power = 2000 gpm Feedflow
  - 1 bypass valve open = 4% power = 1600 gpm Feedflow

EOP 1A

Pressure Band
800# - 1000# Nominal
5°F - 10°F below HCL
65# - 200# RCIC (TLAC)

Pressure	Time

LPC/P-3 ☐

Pressure Band

Back Pressure	**RCIC**
140°F Suction	**RCIC**
5.75 / 7.25	**RCIC**

STABILIZE RPV pressure below 1065 psig

USE one or more of the following systems:

1) **MAIN TURBINE / BYPASS VALVES**

2) Alternate pressure control systems — SPI

- **SRVs**
  - **EVALUATE** Bypass of Instrument Air Isolation — EOP-SPI 2.8
  - **OPEN** in sequence listed on P601
  - **PLACE** control switch in OFF for SRVs with **NO** air available
  - **USE** Suppression Pool Cooling
- **RCIC** Injection and Pressure Control — EOP-SPI 6.6
- **STEAM JET AIR EJECTORS**
- **RFPTs**
- **MAIN STEAM LINE DRAINS**
- **RWCU**
  - **DIRECT** Bypass of RWCU Isolations — EOP-SPI 2.6
    - RWCU—Recirculation Mode

LPC/P-4 ☐

WHEN ANY of the following conditions exist:

- Plant conditions permit a controlled depressurization
- MSIVs are closed
- Operating in a TLAC,

ATTACHMENT #6

QUESTION SRO 15

The plant was operating at full power, when the following occurred:

- Both Feedwater Pump Turbines tripped.
- The Motor Feed Pump failed to start.
- The reactor automatically scrammed.
- One Control Rod is at position 48.
- All other Control Rods are fully inserted.
- HPCS initiation raised RPV Water Level from 110 inches.
- HPCS was manually overridden OFF as RPV Water Level reached 210 inches.

Current Plant conditions are:

- Reactor pressure 700 psig, rising at 10 psig per minute.
- MSIVs are open.
- The operating CRD Pump tripped.

Over the next ten minutes RPV Water Level will __(1)__.
The procedure used to control RPV Water Level is __(2)__.

	__(1)__	__(2)__
A.	rise due to swell	EOP-1, RPV Control
B.	rise due to swell	EOP-1A, Level Power Control
C.	lower due to shrink	EOP-1, RPV Control
D.	lower due to shrink	EOP-1A, Level Power Control

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #		1
	Group #		2
	K/A#	295008	AA2.05
	Importance Rating		3.1

K&A: Ability to determine and/or interpret the following as they apply to High Reactor Water Level: Swell

High Reactor Water Level / 2

Explanation: **Answer A** – HPCS injected100 inches x 250 gal per inch= 25,000 gallons of cold CST water. As this water is heated, SWELL occurs. Shrink cannot occur because heatup and pressurization of saturated system is in progress with NO steam voids. Since the nominal assigned pressure band is 800-1000 psig, all SRVs and BPVs are shut for the next ten minutes because Reactor Pressure will be below 800 psig. With only 1 control rod out, the reactor is shutdown and EOP-1 is the appropriate procedure.

B – Incorrect – Plausible since EOP-1A is for ATWS and 1 rod is withdrawn. However, SHUTDOWN is defined as all rods in except for 1.

C – Incorrect - RPV level will rise due to swell.

D – Incorrect - RPV level will rise due to swell. And EOP-1A is for ATWS and 1 rod is withdrawn. However, SHUTDOWN is defined as all rods in except for 1.

Technical Reference(s): GFE Chap. 7 Rev 4, EOP Bases Rev 3, EOP-1A Chart Rev D & EOP-1 Chart Rev D	Reference Attached: GFE Chap. 7 p 28, EOP Bases pp 46-47, EOP-1A Chart (partial) & EOP-1 Chart (partial)

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OT-3303-07 & OT-3302-04A

Question Source:	Bank #	
	Modified Bank #	Fermi 2008
	New	
Question History:	Previous NRC Exam	Fermi 2008 #83
Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	x
10 CFR Part 55 Content:	55.41	
	55.43	b.5

Comments: Level of Difficulty = x Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations. [10 CFR 55.43(b)(5)]

This 10 CFR 55.43 topic involves both 1) assessing plant conditions (normal, abnormal, or emergency) and then 2) selecting a procedure or section of a procedure to mitigate, recover, or with which to proceed

Candidate Comments:

1. The question describes a situation in which most sources of injection to the RPV are not operating. A significant volume of relatively cold water has been injected to the RPV, and is heating up as indicated by the rising pressure. No information is provided about the status of RCIC, but the stem indicates that RPV water level did fall below the automatic initiation setpoint for RCIC. The stem of the question also does not indicate that any operator action was taken with respect to the multiple drain lines that would normally be open under the stated conditions. Therefore, there are multiple competing effects on RPV water level.

2. Within the industry, the terms shrink and swell are commonly used to describe water level effects associated with changes in steam flow. Since this is not occurring, the terms as commonly used do not apply. Additionally, these terms are not specifically defined in the BWR General Fundamentals reference material. Therefore, the terms shrink and swell must be interpreted according to their generic definitions.

3. A rise in level can be anticipated based on the heating of the water injected by HPCS. A drop in level can also be anticipated based on RPV inventory loss through the open drain lines. A simulator scenario utilizing the same initial conditions outlined in the stem of the question was run to validate the plant response. Over the 10 minute time frame specified in the question, RPV water level displayed both rising and lowering trends.

Candidate Justifications for answers:

A. Correct – RPV water level does exhibit a rising trend within the 10 minute time frame due the heat-up of the colder water, as well as the continued injection from RCIC. Since the reactor is shutdown under all conditions, EOP-1, RPV control is the correct procedure.
B. Incorrect answer – The stem of the question states that only 1 control rod is not fully inserted, which meets the criteria for the reactor being shutdown under all conditions. Therefore, EOP-1A, Level Power Control, is not the correct procedure.
C. Correct – RPV water level does exhibit a lowering trend within the 10 minute time frame due to the rising pressure, as well as the inventory loss through the drain lines. Since the reactor is shutdown under all conditions, EOP-1, RPV control is the correct procedure.
D. Incorrect answer – The stem of the question states that only 1 control rod is not fully inserted, which meets the criteria for the reactor being shutdown under all conditions. Therefore, EOP-1A, Level Power Control, is not the correct procedure.

Recommendation: Allow two correct answers, A and C correct

References: None provided by candidate

Station Proposed Resolution:
The station staff does not support the candidate's basis and this question does, in fact, have only one correct answer.

While it is true over the next ten minutes RPV Water level both rises and lowers, the causes indicate there is only one correct answer. The injection of cold water from HPCS causes level to rise due to swell as the cold water injected begins to heat up and expand making answer A the correct answer. RPV Water Level begins to lower later in the ten minute period however the cause is a loss of inventory due to open main steam line drains which makes C an incorrect answer.

ATTACHMENT #7

QUESTION SRO 18

The plant is in shutdown with the following conditions:

- Average Reactor Coolant temperature is 190°F.
- RHR A loop placed in Shutdown Cooling (SDC) Mode of operation IAW SOI-E12

Based on this information, the LPCI mode of RHR A system is ____.

A. NOT affected, since it is NOT required to be OPERABLE with the current plant conditions

B. INOPERABLE, since the RHR Minimum Flow Valve is deenergized closed for SDC Operations

C. INOPERABLE, since the system must be manually realigned when required

D. OPERABLE, provided the system can be manually realigned when required

QUESTION SRO 18

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #		2
	Group #		1
	K/A#	203000	2.2.25
	Importance Rating		4.2

K&A: Knowledge of the bases in Technical Specifications for limiting conditions for operations and safety limits.

RHR/LPCI: Injection Mode

Explanation: **Answer D** – Per TS 3.5.2 Bases, one LPCI subsystem may be considered operable during alignment or operation for decay heat removal in Mode 4 or 5, if capable of being manually realigned to the LPCI mode.

A – Incorrect – LPCI Mode is required to be Operable in this mode.

B – Incorrect - LPCI subsystem is considered operable under this condition.

C – Incorrect - LPCI subsystem is considered operable under this condition.

Technical Reference(s): TS 3.5.2, TS 3.5.2 Bases Rev 7,	Reference Attached: TS 3.5.2, TS 3.5.2 Bases p B 3.5-15

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OT-3037-09-B

Question Source:	Bank # Modified Bank # New	 LaSalle 2003
Question History:	Previous NRC Exam	LaSalle 2003 #102
Question Cognitive Level:	Memory or Fundamental Knowledge Comprehension or Analysis	 x
10 CFR Part 55 Content:	55.41 55.43	 b.2

Comments: Level of Difficulty = x Facility operating limitations in the TS and their bases. [10 CFR 55.43(b)(2)]

Knowledge of TS bases that are required to analyze TS required actions and terminology.

HO – need to determine RHR cut in permissive pressure and mode of operation.

Candidate Comments:

1. The question asks how the LPCI mode of operation is impacted when a train of the Residual Heat Removal System is placed in the Shutdown Cooling Mode of operation.
2. In Mode 4, only 2 ECCS injection/spray subsystems shall be OPERABLE (TS 3.5.2).
3. Per TS 3.5.2 Bases, one LPCI subsystem may be considered operable during alignment or operation for decay heat removal in Mode 4 or 5, if capable of being manually realigned to the LPCI mode.
4. Candidates are instructed prior to the test (NUREG 1021 Appendix E) not to make assumptions regarding conditions that were not specified in the question unless they occurred as a consequence of other conditions that were stated in the question.
5. Stem of the question does not identify any other systems as being INOPERABLE. ALL ECCS systems are OPERABLE therefore LPCI A is NOT required to be OPERABLE.

Candidate Justifications for answers:

A. **Correct Answer** – LPCI A is NOT required to be OPERABLE with the current plant conditions. In Mode 4, only 2 ECCS injection/spray subsystems shall be OPERABLE (TS 3.5.2). Stem does not identify any other systems as being INOPERABLE

B. Incorrect Answer - INOPERABLE, the RHR Minimum Flow Valve deenergized closed for SDC Operations does not make the system INOPERABLE. TS 3.5.2 Bases, one LPCI subsystem may be considered operable during alignment or operation for decay heat removal in Mode 4 or 5, if capable of being manually realigned to the LPCI mode

C. Incorrect Answer - INOPERABLE, since the system must be manually realigned when required. TS 3.5.2 Bases, one LPCI subsystem may be considered operable during alignment or operation for decay heat removal in Mode 4 or 5, if capable of being manually realigned to the LPCI mode

D. **Correct Answer** - OPERABLE, provided the system can be manually realigned when required. TS 3.5.2 Bases, one LPCI subsystem may be considered operable during alignment or operation for decay heat removal in Mode 4 or 5, if capable of being manually realigned to the LPCI mode

Recommendation: Amend the Answer Key to list 'Answer A' as a correct answer, in addition to 'Answer D'

References: TS 3.5.2, TS 3.5.2 bases

Station Proposed Resolution:

Station agrees with the above candidate Comments, Justifications and Recommendation since we have provided "newly discovered technical information that supports a change in the answer key."

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS) AND REACTOR CORE ISOLATION COOLING (RCIC) SYSTEM

3.5.2 ECCS—Shutdown

LCO 3.5.2 Two ECCS injection/spray subsystems shall be OPERABLE.

APPLICABILITY: MODE 4,
MODE 5 except with the reactor vessel head and steam dryer storage/reactor well gate removed and water level ≥ 22 ft 9 inches over the top of the reactor pressure vessel flange.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One required ECCS injection/spray subsystem inoperable.	A.1	Restore required ECCS injection/spray subsystem to OPERABLE status.	4 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1	Initiate action to suspend operations with a potential for draining the reactor vessel (OPDRVs).	Immediately
C. Two required ECCS injection/spray subsystems inoperable.	C.1	Initiate action to suspend OPDRVs.	Immediately
	<u>AND</u>		
	C.2	Restore one ECCS injection/spray subsystem to OPERABLE status.	4 hours

(continued)

B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS) AND REACTOR CORE ISOLATION COOLING (RCIC) SYSTEM

B 3.5.2 ECCS-Shutdown

BASES

BACKGROUND

A description of the High Pressure Core Spray (HPCS) System, Low Pressure Core Spray (LPCS) System, and low pressure coolant injection (LPCI) mode of the Residual Heat Removal (RHR) System is provided in the Bases for LCO 3.5.1, "ECCS-Operating."

APPLICABLE SAFETY ANALYSES

ECCS performance is evaluated for the entire spectrum of break sizes for a postulated loss of coolant accident (LOCA). The long term cooling analysis following a design basis LOCA (Ref. 1) demonstrates that only one ECCS injection/spray subsystem is required, post LOCA, to maintain the peak cladding temperature below the allowable limit. It is reasonable to assume, based on engineering judgement, that while in MODES 4 and 5, one ECCS subsystem can maintain adequate reactor vessel water level. To provide redundancy, a minimum of two ECCS subsystems are required to be OPERABLE in MODES 4 and 5.

The ECCS satisfy Criterion 3 of the NRC Final Policy Statement on Technical Specification Improvements (58 FR 39132). 09-047

LCO

Two ECCS injection/spray subsystems are required to be OPERABLE. The ECCS injection/spray subsystems are defined as the three LPCI subsystems, the LPCS System, and the HPCS System. The ECCS injection/spray subsystems are further divided into the following groups:

a) The low pressure ECCS injection/spray subsystems are the LPCS System and the three LPCI subsystems;
b) The ECCS injection subsystems are the three LPCI subsystems; and
c) The ECCS spray subsystems are the HPCS System and the LPCS System.

One LPCI subsystem (A or B) may be considered OPERABLE during alignment and operation for decay heat removal in MODE 4 or 5, if capable of being manually realigned (remote or local) to the LPCI mode and is not otherwise inoperable. Because of low pressure and low temperature conditions in MODES 4 and 5, sufficient time will be available to manually align and initiate LPCI subsystem operation to provide core cooling prior to postulated uncovering of the fuel. 09-047

(continued)

Attachment 5
L-13-117

Written Examination Seating Chart
(1 page)

Perry NRC Exam Seating Chart
March 6, 2013

Proctor's Table

RO Chiarelli	SRO-I Vriezen	SRO-I T. Smith
SRO-I D. Smith	RO Wiley	SRO-I Waggel
SRO-I Kuntz	RO Kozlowski	SRO-I Vaughn
SRO-I Douglass	SRO-U Jones	SRO-U Hallmark

Attachment 6
L-13-117

Form ES-403-1, Written Examination Grading Quality Checklist
(1 page)

Attachment 7
L-13-117

Scenario #2 forms ES-D-1 and ES-D-2
(28 pages)

Facility: Perry | Scenario No.: 2- 100% | Op-Test No.: 2013-01

Examiners: ____________ Operators: ____________ SRO
____________ ____________ ATC
____________ ____________ BOP

Initial Conditions: Reactor is at full power. RHR B was tagged out of service yesterday for valve work. RHR C breaker was racked out late last shift due to observation of nicked control power wire in breaker cubicle. TS 3.5.1 Condition C was entered 2 hours ago. Efforts are in progress to restore RHR B or C to Operable status. M29 Boiler is out of service for repair. Service Water B pump is OOS due to high vibrations – awaiting new motor. IOI-3 Section 4.5 is complete, rods at Step 69. Unusually low ambient temperatures are predicted for today. Very low load on grid today. PSA Risk is Green. Grid is Normal Annunciators locked in: H13-P970 F6 (MH #11 B/U Pump Start) & H13-870-7A E2 (SSE Drain Tank Low Level)

Turnover: Shift TBCC pumps to equalize run time. When concurrence granted from WCC, commence Rx Power reduction per IOI-3 for low grid load.

Event No.	Malf. No.	Event Type*	Event Description
1		N (BOP/ SRO)	Shift TBCC pumps
2		R (ATC/ SRO)	Lower Rx power with flow
3	CP03_0P4 3C0001B	C (BOP/ SRO)	NCC B Pump degradation / trip. Enter ONI-P43
4	Cb01_1n21 c0002b	C (ATC/ SRO)	Condensate booster pump B trip
5	RP01A	C (BOP) C (SRO) C(BOP) C(SRO)	Loss of RPS Bus A. Enter ONI-C71-2 Enter ORM 6.3.1 Testing Requirement 5 RWCU Valve G33-F004 fails on isolation Evaluate TS 3.6.1.3
6	cb04_1e21 c0001	M (ALL) C (BOP) C (SRO) C (BOP)	Steam Leak in Steam Tunnel. Enter EOP-3, Plant scram on MSIV Closure due to high tunnel temperature with MSL A failure to isolate. ATWS – enter EOP-1 and EOP-1A ~~Enter T.S. 3.6.1.3~~ LPCS fails to auto start on T&P
7	TH25 MS01A & MS01E	C (BOP/ SRO)	SRV failure to open during manual operations while controlling Rx pressure with SRVs.
8			Emergency Depressurization on lowering Rx level – EOP-4-2

* (N)ormal, (R)eactivity, (I)nstrument, (C)omponent, (M)ajor

Op-Test No.: 2013-01	Scenario No.: 2 – 100%	Page 1 of 27
Event Description: N/A - Driver Instructions		
Cue: None		

Time	Position	Applicant's Actions or Behavior
Driver	**Driver**	Simulator Setup: Reset Simulator to IC 91 Load Schedule File: NRC 2013 Scen 2.sch Verify Schedule Files Loaded: Loss_of_FW.sch NRC-13_info.sch Verify Event File Loaded: NRC 2013 Scen 2.evt Verify temporary Recirc placard is removed from P680.
Driver	**Driver**	**Verify** Initial Conditions: Reactor Power 100%. BOL Pull Sheets, Rods @ Step 69. IOI-3 Step 4.5 is complete. M29 Boiler is out of service for repairs. Place yellow switch cap on RHR B, SWP B and SWP B discharge valve. PSA Risk: Yellow due to RHR B and C being Inoperable. Verify Traffic Light changed to Yellow.
Driver	**Driver**	Turnover: Reactor is at full power. RHR B was tagged out of service yesterday for valve work. RHR C breaker was racked out late last shift due to observation of nicked control power wire in breaker cubicle. TS 3.5.1 Condition C was entered 2 hours ago. Efforts are in progress to restore RHR B or C to Operable status. M29 Boiler is out of service for repair. Service Water B pump is OOS due to high vibrations – awaiting new motor. IOI-3 Section 4.5 is complete, rods at Step 69. Unusually low ambient temperatures are predicted for today. Very low load on grid today. PSA Risk is Green. Grid is Normal Annunciators locked in: H13-P970 F6 (MH #11 B/U Pump Start) & H13-870-7A E2 (SSE Drain Tank Low Level) Shift TBCC pumps from A to B to equalize run times. When concurrence granted from WCC, commence Rx Power reduction per IOI-3 for low grid load.

Op-Test No.: 2013-01	Scenario No.: 2 – 100%	Page 2 of 27
Event Description: 1 - Shift TBCC pumps		
Cue: From Turnover/SRO direction		

Time	Position	Applicant's Actions or Behavior
00:00	SRO	Directs BOP to shift TBCC pumps IAW SOI-P44 Section 7.2
	SRO	Ensures plant operations are conducted IAW Operations Expectations and Standards.
	BOP	Reviews SOI-P44 and performs TBCC pump shift
		7.2.1 TAKE the oncoming TBCC PUMP to START. 1P44-C001B 7.2.2 TAKE the offgoing TBCC PUMP to STOP. 1P44-C001A
		Observes 'B' TBCC pump discharge pressure rise prior to stopping 'A' TBCC.
Driver	**Driver**	Role play as NLO, communicate with BOP during pump shift
	BOP	Inform SRO that TBCC pump shift is complete
Driver	**Driver**	When BOP is ~95% complete with TBCC pump shift, or when directed by the Lead Examiner, continue on to the next Event.

Op-Test No.: 2013-01	Scenario No.: 2 – 100%	Page 3 of 27
Event Description: 2 - Lower Reactor power with flow		
Cue: From Turnover/SRO direction		

Time	Position	Applicant's Actions or Behavior
Driver	**Driver**	When directed by Lead Examiner, inform the Unit Supervisor that WCC has authorized a power reduction to 95% due to low loading on the grid.
	SRO	Directs ATC to lower power to 95% using flow IAW Reactivity Plan, IOI-3, and SOI-B33
	ATC	Commences lowering reactor power to 95%
		SOI-B33 Sect 7.7 Rcirc Flow Control in Loop Manual
		7.7.1 CONFIRM RCIRC LOOP FLOW CONTROL is in MAN. 1B33-K603A 7.7.2 CONFIRM RCIRC LOOP FLOW CONTROL is in MAN. 1B33-K603B 7.7.3 PERFORM the following concurrently as required for the desired Recirc Flow: ADJUST RCIRC LOOP FLOW CONTROL with the slide switch. 1B33-K603A ADJUST RCIRC LOOP FLOW CONTROL with the slide switch. 1B33-K603B
Driver	**Driver**	Role play as necessary as Shift Manager, Chemistry, RP, etc.
	Evaluator	Note: It takes almost 2 minutes from the time the next Event is triggered until the first alarm comes in.
Driver	**Driver**	When Power is lowered to 95% or Evaluator has determined sufficient power decrease has been achieved, proceed to next Event.

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Event Description: 3 - NCC pump degradation / trip. Enter ONI-P43		
Cue: Annunciators H13-P680-8A-B4 & H13-P970-B1		

Time	Position	Applicant's Actions or Behavior
Driver	**Driver**	When directed by Lead Examiner, initiate Event 3.
	ATC	Announces Unexpected Alarm, "COM LONG RESPONSE P970". (H13-P680-8A-B4)
	BOP	Responds to H13-P970 annunciators and NCC header discharge pressure and determines that NCC pump B is degrading. Informs crew of NCC Pump B degradation.
		Refers to ARI and announces entry condition for ONI-P43. 4.1 REFER TO ONI-P43, Loss of Nuclear Closed Cooling. 4.2 VERIFY the NCC Pump Suct valve for the operating NCC Pumps are open.
		Directs NLO to verify NCC B Pump Suct valve is open
Driver	**Driver**	Role play as NLO. If sent to investigate NCC pump B, report abnormal noise and vibration
	SRO	Enters ONI-P43, Loss of Nuclear Closed Cooling
		Directs BOP to perform Supplemental Actions of ONI-P43
	Crew	ONI-P43 4.1.1 IF only ONE NCC pump is running AND a standby NCC pump is available, THEN REFER TO SOI-P43 and START the standby NCC pump.
	BOP	Directs NLO to support shifting of pumps.
	BOP	Refer to SOI-P43 and performs sections 7.1, 4.2 and 6.1 of the operating instruction. 7.1 <u>Shifting NCC Pumps</u> 7.1.1 REFER TO Additional NCC Pump Startup and START the standby pump. 7.1.2 REFER TO NCC Pump Shutdown and STOP one of the running pump. 4.2 <u>Additional NCC Pump Startup</u> 4.2.1 THROTTLE the oncoming NCC Pump Disch 10% open. P43-F513C 4.2.2 TAKE the oncoming NCC PUMP control switch on Common Long Response Control Panel H13-P970 to START. P43-C001C 4.2.3 OPEN the oncoming NCC Pump Disch. P43-F513C 4.2.4 VERIFY NCC HDR PRESSURE on P970 stabilizes between 94 – 123 psig. P43-R221

Op-Test No.: 2013-01	Scenario No.: 2 – 100%	Page 5 of 27
Event Description: 3 - NCC pump degradation / trip. Enter ONI-P43		
Cue: Annunciators H13-P680-8A-B4 & H13-P970-B1		

Time	Position	Applicant's Actions or Behavior
	BOP	6.1 NCC Pump Shutdown CAUTION Operation of the NCC Pump with its discharge valve < 10% open should be minimized to prevent pump damage. 6.1.1 SLOWLY CLOSE the desired NCC Pump Disch. P43-F513B 6.1.2 IMMEDIATELY TAKE the offgoing NCC PUMP control switch on H13-P970 to STOP. P43-C001B 6.1.3 OPEN the offgoing NCC Pump Disch. P43-F513B 6.1.4 VERIFY proper discharge check valve operation by confirming no indication of reverse pump rotation.
Driver	**Driver**	Role play as NLO to support shifting NCC pumps. Use Remote Function SW016 to throttle/open P43-F513C. Use Remote Function SW015 to close/open P43-F513B.
Driver	**Driver**	If requested report that there is no indication of reverse pump rotation on the NCC B pump.
Driver	**Driver**	**When the pump shift is almost complete and the BOP operator is still at P970**, or when directed by the Lead Examiner, initiate Event 4.

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Event Description: 4 - Condensate Booster pump B trip		
Cue: Annunciator H13-P680-2A-B6		

Time	Position	Applicant's Actions or Behavior
Driver	**Driver**	When directed by Lead Examiner, initiate Event 4.
	Evaluator	ATC has approximately 3.5 minutes before HST level lowers to 60". Starting the 2nd Condensate Booster pump will stop level lowering and restore HST level.
	ATC	Announce H13-P680 unexpected annunciators. Observe alarms, CBP breaker status lights, and discharge pressure meters on P680 and determines CBP "B" has tripped. Informs crew of CBP B trip.
		Refers to ARI and starts standby CBP IAW Subsequent Actions: 4.1 IF the turbine has NOT tripped, THEN start the standby CBP, 1N21-C002C 4.2 MONITOR HOT SURGE TANK LEVEL & CNDS TO HTR 4 FLOW. 1N21-R323 4.3 IF required, THEN REDUCE reactor power to stabilize Hot Surge Tank level. 1N21-R323 4.4 MAINTAIN motor current <353 amps. (1N21-C001A & 1N21-C001C)
		Informs US of CBP "C" start – completion of ARI steps 4.1, 4.2 and 4.4
Driver	**Driver**	If requested respond as NLO to breaker H1205. Report that Overcurrent Relays for the breaker are tripped.
Driver	**Driver**	If requested to respond as NLO to the pumps – report that nothing appears abnormal for B pump, and that start-up of the C pump appears normal.
Driver	**Driver**	If requested to respond as NLO to Condensate Filter System – High Differential Pressure alarm – reset on Acknowledgement (Use Extreme View to acknowledge local alarms)
Driver	**Driver**	When ATC is complete with CBP pump shift, or when directed by the Lead Examiner, continue on to the next Event.

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Event Description: 5 - Loss of RPS Bus A. Enter ONI-C71-2. Enter ORM 6.3.1 Testing Requirement 5 RWCU Valve G33-F004 fails on isolation. Evaluate TS 3.6.1.3		
Cue: RPS CH SCRAM SOL VALVES indicating lights not lit for GP 1A, 2A, 3A and 4A		

Time	Position	Applicant's Actions or Behavior
Driver	**Driver**	When directed by Lead Examiner, insert Event 5.
	ATC	Announces multiple unexpected alarms. Responds to multiple Annunciators and the RPS CH SCRAM SOL VALVES indicating lights not lit for GP 1A, 2A, 3A and 4A. Reports ½ scram RPS A bus
Driver	**Driver**	Role play as NLO. If requested to investigate RPS electrical power, report that 1C71-S003A breaker has green indicating light on, and 1C71-S001A has a red indicating light on and everything else looks normal.
	SRO	Enter ONI-C71-2, Loss of One RPS Bus.
	SRO	Direct BOP to re-energize RPS Bus A per ONI-C71-2, Supplemental Actions
	BOP	Co-ordinate with ATC and re-energize RPS Bus A per ONI-C71-2. MG SET TRANSFER switch is in NORM RPS Bus A GEN ALT AVAIL light on THEN PLACE the MG SET TRANSFER switch in RPS Bus A Alternate Source on P640. (1C71-S1)
	SRO	Direct ATC to reset ½ Scram per SOI-C71. Direct BOP to reset/restore isolations per IOI-18.
		4.1.8 Refer to Technical Specifications - NOTE: Refer to ONI-C71-2 for multiple applicable Tech Spec/ORMs. Enter ORM 6.3.1 Testing Requirement 5
	ATC	Coordinate with BOP to Reset RPS per SOI-C71 Sect 7.4
	ATC	7.4.1 VERIFY the following: • The conditions which caused the full or half scram have cleared. • There is reasonable assurance that another scram signal will NOT be generated.
		7.4.4 MOMENTARILY DEPRESS the appropriate RPS division pushbuttons on P680: RPS A • SCRAM RESET CH A. 1C71A-S5A • SCRAM RESET CH B. 1C71A-S5C

Op-Test No.: 2013-01	Scenario No.: 2 – 100%	Page 8 of 27
Event Description: 5 - Loss of RPS Bus A. Enter ONI-C71-2. Enter ORM 6.3.1 Testing Requirement 5 RWCU Valve G33-F004 fails on isolation. Evaluate TS 3.6.1.3		
Cue: RPS CH SCRAM SOL VALVES indicating lights not lit for GP 1A, 2A, 3A and 4A		

Time	Position	Applicant's Actions or Behavior
	ATC	7.4.7 VERIFY the following SCRAM DISCH VOL DRAIN VALVE lights are on at 1H13-P680: • INSTR VOLUME VENT VLV OPEN. • INSTR VOLUME DRAIN VLV OPEN.
		Inform US that ½ scram RPS A has been reset.
	BOP	Perform ONI-C71-2 Supplemental Actions
	BOP	ONI-C71-2 Supplemental Actions
		4.1.10 VERIFY all SRV control switches on P601 are in AUTO 4.1.11 VERIFY all SRV control switches on P631 are in AUTO 4.1.12 VERIFY GROSS/FAIL TRIP/LATCH lights are reset at the following panels: • 1H13-P692, • 1H13-P693, • 1H13-P691, • 1H13-P694
	Driver / Evaluator	If asked, the above gross fail light are reset.
	BOP	Inform SRO above actions are complete.
	BOP	4.1.13 REFER TO IOI-18 and RESTORE the following isolations as appropriate. • BALANCE OF THE PLANT ISOLATION (L2 /1.68#) RESTORATION
		4.1.16 If required then OPEN the MSL DRM & MSIV BYP OTBD ISOL B21F019.
	RO	Identifies G33-F004 failed to isolate. Inform US.
	SRO	Evaluates TS 3.6.1.3 and enters Condition A . Directs G33-F001 to be closed within 4 hours.
	BOP	Closes G33-F001 as directed.

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Event Description: 5 - Loss of RPS Bus A. Enter ONI-C71-2. Enter ORM 6.3.1 Testing Requirement 5 RWCU Valve G33-F004 fails on isolation. Evaluate TS 3.6.1.3		
Cue: RPS CH SCRAM SOL VALVES indicating lights not lit for GP 1A, 2A, 3A and 4A		

Time	Position	Applicant's Actions or Behavior
	Evaluator	**Note**: Restoration of isolations is not necessary and does not impact the remainder of this scenario.
	BOP	IOI-18 Actions
		Perform Attachment 33 - BALANCE OF THE PLANT ISOLATION (LEVEL 2 / 1.68#)
		1.0 CONFIRM the following alarms reset: • BOP ISOL DW PRESS HIGH H13-P601-19A-A6 • BOP ISOL RX LEVEL LO L2 H13-P601-19A-B6
		2.0 MOMENTARILY DEPRESS the following: • MSL & NS4 OTBD ISOL SEAL IN RESET. B21H-S32 • MSL & NS4 INBD ISOL SEAL IN RESET. B21H-S33
		4.0 IF restoring an outboard isolation (Division 1), THEN PERFORM the following: 4.1 VERIFY the following open: • SA SUPPLY HDR CNTMT ISOL. P51-F150 • CTS SUPPLY HDR CNTMT ISOL. P11-F060
		4.2 AT 1H13-P881, VERIFY the following are open: • PERS AL EL 603 OTBD ALRM ISOL P53-F070 • PERS AL EL 692 OTBD ALRM ISOL P53-F075 • PERS AL EL 692 SUPP AIR OTBD ISOL P52-F170 • PERS AL EL 603 SUPP AIR OTBD ISOL P52-F160 • DW EQUIP DRAIN OTBD DW ISOL G61-F035 • DW FLOOR DRAIN OTBD DW ISOL G61-F155 • CNTMT EQUIP DRAIN OTBD ISOL G61-F080 • CNTMT FLOOR DRAIN OTBD ISOL G61-F170 • RWCU BACKWASH OUT OTBD ISOL G50-F277 • MIXED BED WTR CNTMT SUPPLY ISOL P22-F010 • DW CO2 SUPPLY OTBD ISOL P54-F395
		4.3 VERIFY the valves closed: • PERS AL EL 603 INNER DR AEGTS ISOL P53-F035 • PERS AL EL 692 INNER DR AEGTS ISOL P53-F045
		4.4 IF the Containment Airborne Radiation Monitor was in service, THEN VERIFY the following valves open: • CNTMT RAD MON OTBD SUCT ISOL D17-F081A • CNTMT RAD MON OTBD DISCH ISOL D17-F089A

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Event Description: 5 - Loss of RPS Bus A. Enter ONI-C71-2. Enter ORM 6.3.1 Testing Requirement 5 RWCU Valve G33-F004 fails on isolation. Evaluate TS 3.6.1.3		
Cue: RPS CH SCRAM SOL VALVES indicating lights not lit for GP 1A, 2A, 3A and 4A		

Time	Position	Applicant's Actions or Behavior
	BOP	4.5 IF the Drywell Airborne Radiation Monitor was in service, THEN VERIFY the following valves open: • DW RAD MON OTBD SUCT ISOL D17-F071A • DW RAD MON OTBD DISCH ISOL D17-F079A
		5.0 IF Containment Vessel Chilled Water was in service, PERFORM the following: 5.1 AT H13-P800, VERIFY the following valves open: • CVCW OTBD SUPP ISOL VALVE P50-F060 • CVCW OTBD RETURN MOV ISOL VALVE P50-F150 • CVCW INBD RETURN MOV ISOL VALVE P50-F140
		5.2 REFER to SOI-P50 and SHIFT chillers.
		Direct an NLO to start a P50 chiller per SOI-P50 and SHIFT chillers.
		6.0 TAKE the following to closed at 1H13-P800: • DW VAL RLF MOV ISOL VALVE M16-F010A • DW VAL RLF MOV ISOL VALVE M16-F010B
		7.0 VERIFY the following are open at 1H13-P800: • CNTMT VAC RLF MOV ISOL VALVE M17-F015 • CNTMT VAC RLF MOV ISOL VALVE M17-F025 • CNTMT VAC RLF MOV ISOL VALVE M17-F035 • CNTMT VAC RLF MOV ISOL VALVE M17-F045
Driver	**Driver**	When ½ isolation is reset, or when directed by the Lead Examiner, initiate Event 6

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Event Description: 6 - Steam Leak in Steam Tunnel. Enter EOP-3, Plant scram on MSIV Closure due to high tunnel temperature with MSL A failure to isolate. ATWS – enter EOP-1 and EOP-1A, ~~Enter T.S. 3.6.1.3~~

Cue: H13-P680-7A-D6 for Steam Break, 1H13-P601-21A-B2 for EOP-3 entry,1H13-P601-19A-A1 for MSIV Closure

Time	Position	Applicant's Actions or Behavior
Driver	**Driver**	When directed by Lead Examiner, initiate Event 6.
	ATC	Announce unexpected H13-P680 alarm. (ARI-H13-P680-7A-D6) Direct BOP to investigate alarm on P800.
	BOP	Report rising temperatures in Steam Tunnel Areas, including STEAM TUNNEL ZONE 3. May investigate NUMAC (E31-N700) to determine if rise in temperature are consistent with P800.
	SRO	Enters ONI-N11, Pipe Break Outside Containment as directed by ARI-H13-P680-7A-D6 Supplemental Action 4.3.
	Driver / Evaluator	Role play as Shift Manager and PES Manager if requested.
Driver	**Driver**	As NLO, call control room and report hearing sound of steam leak in turbine building..
	ATC	Announce unexpected alarm H13-P601-21A-B2, and potential entry condition for EOP-03
	SRO	Enter EOP-03 Secondary Containment Control
		Directs BOP to monitor Area Temperatures and Area Radiation levels. Directs ATC to monitor Area Water levels.

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Event Description: 6 - Steam Leak in Steam Tunnel. Enter EOP-3, Plant scram on MSIV Closure due to high tunnel temperature with MSL A failure to isolate. ATWS – enter EOP-1 and EOP-1A, ~~Enter T.S. 3.6.1.3~~		
Cue: H13-P680-7A-D6 for Steam Break, 1H13-P601-21A-B2 for EOP-3 entry,1H13-P601-19A-A1 for MSIV Closure		

Time	Position	Applicant's Actions or Behavior
	SRO	Works way down EOP-03 chart to HOLD box 'WHEN any area temperature entry condition is currently met THEN'.
	Evaluator	Annunciator H13-P601-21A-B2 is Entry Condition for EOP-03.
	BOP	Monitors area temperatures on NUMAC E31-N700A A6-2(T) using EOP-03 Conditions Monitoring Hardcard. Reports area above Entry Condition for steam tunnel.
	Evaluator	The transient will progress such that the MSIV closure event will occur before the crew is able to complete many of the following actions.
	Evaluator	When steam tunnel temperature reaches MSIV isolation setpoint, 3 of 4 MSL's will isolate and Rx will scram.
	SRO	Directs crew to Isolate all systems discharging into affected area except for systems required for the following: Shutdown the Reactor, Assure adequate core cooling, Damage control. (**Critical Task #1**) Determine if a Primary system is discharging into affected area. Before any area is above its Max SAFE condition, enter EOP-01, RPV Control and execute concurrently.
	ATC	Responds to new alarms on P680 and 601. Recognize reactor scram.
		Recognizes failure to scram and initiates RPS and ARI. (**Critical Task #2**)
	CREW	Recognize MSIV isolation & failure of MSL 'A' to isolate.
	RO	Isolate MSL A and reports to SRO (**Critical Task #1**)
	~~SRO~~	~~Evaluate TS 3.6.1.3, PCIV – Cond A & Cond B~~
	~~**Evaluator**~~	~~Due to pace of scenario, TS evaluation should be done following scenario termination.~~

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Event Description: 6 - Steam Leak in Steam Tunnel. Enter EOP-3, Plant scram on MSIV Closure due to high tunnel temperature with MSL A failure to isolate. ATWS – enter EOP-1 and EOP-1A, ~~Enter T.S. 3.6.1.3~~

Cue: H13-P680-7A-D6 for Steam Break, 1H13-P601-21A-B2 for EOP-3 entry,1H13-P601-19A-A1 for MSIV Closure

Time	Position	Applicant's Actions or Behavior
	ATC	Performs SCRAM Hardcard actions: Verify the following actions are complete: • Mode Switch Locked in Shutdown • RPS Initiated if all control rods are not fully inserted. • ARI Initiated if RPS failed to Scram the reactor.
		If Reactor Recirc Pumps are running in fast speed: Then simultaneously take the following to XFER: • RECIRC PUMP A BRKR 5A • RECIRC PUMP B BRKR 5B
		IF Reactor power is above 4%, THEN START SLC A and SLC B pumps.
		Perform crew update with the following information: • "The Mode Switch is locked in Shutdown" (Report any failures) • If RPS was initiated, the RPS is initiated (Report any failures) • If ARI was initiated, then ARI is initiated (Report any failures) • "All Control Rods (are/are not) inserted" • Reactor Power is _______________ % ↑↔↓ • Reactor Pressure is _______________ psig ↑↔↓ • Reactor Level is _______________ inches ↑↔↓ • Reactor Recirc Pumps (Running in Slow Speed/Tripped) • Standby Liquid Control System Initiated (only if manually initiated) • EOP-01 Entry (only if conditions met): L2, Rx Press Hi, RPS Failure • If MSIVs are closed, then a Time Critical Operator Action for Suppression Pool Cooling is applicable.
	Evaluator	ATC should report EOP-1 Entry on RPS failure.

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Event Description: 6 - Steam Leak in Steam Tunnel. Enter EOP-3, Plant scram on MSIV Closure due to high tunnel temperature with MSL A failure to isolate. ATWS – enter EOP-1 and EOP-1A, ~~Enter T.S. 3.6.1.3~~		
Cue: H13-P680-7A-D6 for Steam Break, 1H13-P601-21A-B2 for EOP-3 entry,1H13-P601-19A-A1 for MSIV Closure		

Time	Position	Applicant's Actions or Behavior
	ATC	When generator load less than 90 MWe, then perform the following: a) TRIP the main turbine by depressing the TURBINE TRIP push-button. b) VERIFY the following have occurred: • MAIN STOP VALVEs, CONTROL VALVEs and COMBINED INTERMEDIATE VALVEs are shut. • GEN BRKRs S-610-PY-TIE and S-611-PY-TIE open • GEN FIELD BREAKER
	ATC	Insert Nuclear Instruments, • SRMs • IRMs • Place recorders in IRM (leave A or E in APRM)
		Verify HST Lvl CV Manual Control, N21-S19, in OFF
		STABILIZE reactor water level. a) Feedwater (REFER TO FEEDWATER HARDCARD) b) RCIC c) RPV
		STABILIZE reactor pressure: a) Turbine/Turbine Bypass valves (REFER TO PRESSURE CONTROL HARDCARD) b) SRVs • Evacuate Containment • REFER TO PRESSURE CONTROL HARDCARD • Evaluate placing RCIC in Pressure Control Mode

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Event Description: 6 - Steam Leak in Steam Tunnel. Enter EOP-3, Plant scram on MSIV Closure due to high tunnel temperature with MSL A failure to isolate. ATWS – enter EOP-1 and EOP-1A, ~~Enter T.S. 3.6.1.3~~		
Cue: H13-P680-7A-D6 for Steam Break, 1H13-P601-21A-B2 for EOP-3 entry,1H13-P601-19A-A1 for MSIV Closure		

Time	Position	Applicant's Actions or Behavior
	SRO	Enters EOP-1 RPV Control on Reactor Scram required Reactor Power above 4% and transitions into EOP-1A, Level Power Control (RPS Failure)
		Announces entry into EOP-1 and EOP-1A
		Directs ATC to: • Monitor and Control Reactor Power • Stabilize Reactor Water Level • Stabilize Reactor Pressure
	SRO	Works down Power Control leg of EOP-1A
		Answers YES to 'Are APRM's downscale?' Directs ATC to INSERT control rods IAW EOP-SPIs 1.1-1.7 If APRM's come off Downscale, then Directs SLC initiation
	Evaluator	If SLC is directed, it will not inject due to leaking piping inside containment. The CRD B pump will trip.
	SRO	Works down Level Control leg of EOP-1A
		• Directs BOP to verify Actuations and Isolations. • Directs ATC to inhibit ADS. • Directs BOP to perform EOP-SPI 2.3, Bypass MSIV and ECCS interlocks.
	ATC	Inhibits ADS
	BOP	Commences verifying Isolations and Actuations IAW Hardcard.
	Evaluator	If not previously discovered on Rx scram, will find MSL A failed to isolate while performing verification of Isolations and Actuations
	ATC	Perform EOP-SPI 1.3, Manual Rod Insertion. 2.0 VERIFY CRD HYDRAULICS FLOW CONTROL is in MANUAL. C11-R600 3.0 ADJUST CRD HYDRAULICS FLOW CONTROL output to 100. C11-R600 4.0 CLOSE CRD DRIVE PRESS CONTROL VALVE. C11-F003 5.0 WHEN any CRD Pump is running, THEN PERFORM the following to Insert all control rods to position 00 concurrently with the remainder of this procedure follows: 5.1 DEPRESS AND HOLD the IN TIMER SKIP pushbutton. 5.2 SELECT Control Rods not fully inserted.

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Event Description: 6 - Steam Leak in Steam Tunnel. Enter EOP-3, Plant scram on MSIV Closure due to high tunnel temperature with MSL A failure to isolate. ATWS – enter EOP-1 and EOP-1A, ~~Enter T.S. 3.6.1.3~~

Cue: H13-P680-7A-D6 for Steam Break, 1H13-P601-21A-B2 for EOP-3 entry,1H13-P601-19A-A1 for MSIV Closure

Time	Position	Applicant's Actions or Behavior
	BOP	6.0 VERIFY the following keylock switches in BYPASS: • AT H13-P629, LO POWER SET PT DIV 1 BYPASS C11A-S4 • AT H13-P618, LO POWER SET PT DIV 2 BYPASS C11A-S3
	Evaluator	Step 6.0 is also performed in EOP-SPI 2.3.
	BOP	Informs ATC when Lo Power Setpoint is bypassed.
	RO	7.0 PLACE BUS XH11 LOCA BYPASS keylock switch in BYPASS. 8.0 PLACE BUS XH12 LOCA BYPASS keylock switch in BYPASS. 9.0 IF Bus EH11 is energized, THEN TAKE ISOLATING BRKR to CLOSED. EH1116 10.0 IF Bus EH12 is energized, THEN TAKE ISOLATING BRKR to CLOSED. EH1214
	BOP	11.0 AT H13-P970, VERIFY only one of the following is running: • NCC PUMP A P43-C001A • NCC PUMP B P43-C001B • NCC PUMP C P43-C001C
	Evaluator	The remainder of EOP-SPI 1.3 is low priority and can be done later.
	ATC	Announces control rods are going in. (**Critical Task #2**)
	Evaluator	ATC may determine MSIV failure to close while stabilizing pressure if not previously identified.
	BOP	Performs EOP-SPI 2.3, Bypass MSIV's and ECCS Interlocks.
		1.0 DEFEAT MSIV low RPV level isolation as follows: • AT H13-P694, PLACE MSIV ISOL LO LEVEL BYPASS CH D keylock switch in BYP B21H-S76D • AT H13-P691, PLACE MSIV ISOL LO LEVEL BYPASS CH A keylock switch in BYP B21H-S76A • AT H13-P692, PLACE MSIV ISOL LO LEVEL BYPASS CH B keylock switch in BYP B21H-S76B • AT H13-P693, PLACE MSIV ISOL LO LEVEL BYPASS CH C keylock switch in BYP B21H-S76C

Op-Test No.: 2013-01	Scenario No.: 2 – 100%	Page 17 of 27
Event Description: 6 - Steam Leak in Steam Tunnel. Enter EOP-3, Plant scram on MSIV Closure due to high tunnel temperature with MSL A failure to isolate. ATWS – enter EOP-1 and EOP-1A, ~~Enter T.S. 3.6.1.3~~		
Cue: H13-P680-7A-D6 for Steam Break, 1H13-P601-21A-B2 for EOP-3 entry,1H13-P601-19A-A1 for MSIV Closure		

Time	Position	Applicant's Actions or Behavior

	BOP	2.0 DEFEAT ECCS interlocks as follows:
		• AT H13-P625, PLACE HPCS LOGIC BYPASS E22-F023 keylock switch in BYPASS. E22AS25
		• AT H13-P618, PLACE the following Keylock switches in BYPASS:
		  • RHR ISOL BYPASS E12-F053B keylock switch E12AS73
		  • LPCI C LOGIC BYP E12-F021 keylock switch E12AS77
		  • LO POWER SET PT DIV 2 BYPASS C11A-S3
		• AT H13-P629, PLACE the following Keylock switches in BYPASS:
		  • RHR ISOL BYPASS E12-F053A keylock switch E12AS74
		  • LPCS LOGIC BYPASS E21-F012 keylock switch E21S16
		  • LO POWER SET PT DIV 1 BYPASS C11A-S4

		3.0 Instrument Air is isolated to the Drywell NO known air leak is present in Containment NO known air leak is present in Drywell THEN RESTORE Instrument Air to Containment and Drywell as follows: 3.1 VERIFY INST AIR DRYWELL ISOL valve is OPEN. P52-F646 3.3 VERIFY INST AIR CNTMT ISOL valve is OPEN. P52-F200
	BOP	4.0 CONFIRM instrument air is available as follows: 4.1 VERIFY BUS XH11 LOCA BYPASS keylock switch in BYPASS. 4.2 VERIFY BUS XH12 LOCA BYPASS keylock switch in BYPASS. 4.3 IF Bus EH11 is energized, THEN TAKE ISOLATING BRKR to CLOSED. EH1116 4.4 IF Bus EH12 is energized, THEN TAKE ISOLATING BRKR to CLOSED. EH1214 4.5 AT H13-P970, VERIFY only one of the following is running: • NCC PUMP A P43-C001A • NCC PUMP B P43-C001B • NCC PUMP C P43-C001C 5.0 IF RHR C pump is available, THEN PERFORM the following: 5.1 VERIFY LPCI C Injection Valve is CLOSED. 1E12-F042C

Op-Test No.: 2013-01	Scenario No.: 2 – 100%	Page 18 of 27
Event Description: 6 - Steam Leak in Steam Tunnel. Enter EOP-3, Plant scram on MSIV Closure due to high tunnel temperature with MSL A failure to isolate. ATWS – enter EOP-1 and EOP-1A, ~~Enter T.S. 3.6.1.3~~		
Cue: H13-P680-7A-D6 for Steam Break, 1H13-P601-21A-B2 for EOP-3 entry,1H13-P601-19A-A1 for MSIV Closure		

Time	Position	Applicant's Actions or Behavior
	SRO	Directs RO's to terminate ECCS for level control IAW Hardcard **(Critical Task 3)** Directs Level Band of 150 to 219 inches using Feedwater.
	BOP	Terminate and prevent ECCS injection per Hardcard for level control. **(Critical Task 3)**
	BOP	Observes LPCS pump fails to auto start on T&P and starts LPCS Pump manually.
	SRO	Directs startup of Hydrogen Igniters and Hydrogen Analyzers.
		Works way down to LPC/L10 HOLD box.
	BOP	Performs startup of Hydrogen Igniters and Hydrogen Analyzers per Hardcard.
	Evaluator	Crew may initially determine that Bypass valves are controlling pressure and continue to use Bypass Valves until it is determined that the MSIVs should have isolated, and the 'A' MSL valves are manually taken to close. Then Pressure Control will shift to SRVs.
	ATC	As part of Stabilizing Pressure, should determine that MSIVs are isolated with exception of MSIV line A. With A line open, pressure control will be on Bypass Valves,
	ATC	Once MSIV line A is isolated by at least one valve, should report that Pressure Control is on SRVs.
	SRO	Works down Pressure Control leg of EOP-1A
		Direct Pressure Band of 800-1000 psig.
	Evaluator	When taking first SRV (1B21-F0051D) to OPEN for pressure control, it will not open manually, commencing Event 7.
	RO	Controls Rx pressure in directed band using SRV's (or BPV's if MSIV still open).
	SRO	Works down to LPC/P4 HOLD box. Proceeds through HOLD box when MSIV's close.
		Works down to LPC/P6 HOLD box and waits until Rx is shutdown with boron

Op-Test No.: 2013-01	Scenario No.: 2 – 100%	Page 19 of 27
Event Description: 6 - Steam Leak in Steam Tunnel. Enter EOP-3, Plant scram on MSIV Closure due to high tunnel temperature with MSL A failure to isolate. ATWS – enter EOP-1 and EOP-1A, ~~Enter T.S. 3.6.1.3~~		
Cue: H13-P680-7A-D6 for Steam Break, 1H13-P601-21A-B2 for EOP-3 entry,1H13-P601-19A-A1 for MSIV Closure		

Time	Position	Applicant's Actions or Behavior
	SRO	When all control rods are inserted, goes to override step LPC/L-1 and directs RO to terminate Boron Injection.
	RO	Terminates Boron injection by taking SLC pumps to OFF
	SRO	Transitions to EOP-01
Driver	**Driver**	If previously directed to investigate RCIC turbine trip, insert **Event 17** (to remove trip units) after Transition to EOP-01 and <u>report</u> that the RCIC mechanical trip mechanism is reset.
	RO	Use RCIC to restore RPV level to directed band.
		7.8 <u>Recovery to Operating Status from Automatic Turbine Trip</u> 7.8.3 VERIFY the following are closed: • RCIC TRIP THROTTLE VLV POSITION • RCIC PUMP MIN FLOW VALVE 1E51-F019 • RCIC INJECTION VLV 1E51-F013 • RCIC INJ CHECK VLV 1E51-F066 7.8.4 IF RPV level is less than level 2, THEN REMOVE the following trip units from 1H13-P629: • 1B21-N692A • 1B21-N692E 7.8.5 MOMENTARILY DEPRESS the RCIC INIT – MN & FDW TB TRIP - SEAL IN RESET pushbutton. • 1E51A-S18 7.8.6 CONFIRM the following: • The initiation signal has reset. • The white seal in light goes off.
		7.8.7 TAKE the RCIC STEAM SHUTOFF to CLOSE. 1E51-F045 7.8.8 PLACE the RCIC PUMP FLOW CONTROL in MANUAL. 1E51-R600 7.8.9 ADJUST the RCIC PUMP FLOW CONTROL to minimum (0 gpm). 7.8.10 HOLD the RCIC TURBINE TRIP THRT V LATCH in CLOSE UNTIL closed to reset the trip latch. 1E51-F510 7.8.11 IF the mechanical overspeed trip has actuated AND RCIC Turbine speed is below 3,000 RPM, THEN PULL the mechanical trip rod to the reset position locally to reset the trip device.

Op-Test No.: 2013-01	Scenario No.: 2 – 100%	Page 20 of 27
Event Description: 6 - Steam Leak in Steam Tunnel. Enter EOP-3, Plant scram on MSIV Closure due to high tunnel temperature with MSL A failure to isolate. ATWS – enter EOP-1 and EOP-1A, ~~Enter T.S. 3.6.1.3~~		
Cue: H13-P680-7A-D6 for Steam Break, 1H13-P601-21A-B2 for EOP-3 entry,1H13-P601-19A-A1 for MSIV Closure		

Time	Position	Applicant's Actions or Behavior
	RO	7.8.13 START the RCIC Turbine as follows: 7.8.13.b HOLD the RCIC TURBINE TRIP THRT V LATCH in OPEN. 7.8.13.c TAKE the RCIC STEAM SHUTOFF to OPEN to roll the turbine. 7.8.13.d ADJUST the RCIC PMP FLOW CONTROL to raise RCIC Turbine speed to above 2000 RPM on RCIC TURB RPM.
		7.8.13.e IF pump discharge pressure is > 125 psig. THEN VERIFY the RCIC PUMP MIN FLOW VALVE opens WHEN pump flow is < 120 gpm. 7.8.14 RECORD the appropriate Maintenance Rule status in the Plant Narrative Log. 7.8.15 IF injecting to the reactor vessel, THEN TAKE the RCIC INJECTION VLV to OPEN. 1E51-F013 7.8.17 REFER TO RPV Level and Pressure Control, and ADJUST the flow rate.
	Evaluator	It is anticipated that all control rods <u>will</u> be inserted prior to reaching -25 inches. If the crew fails to insert all rods prior to reaching -25 inches, transition to EOP 4-2 Emergency Depressurization would be appropriate. ED would then be a Critical Task.
	SRO	If control rods are out and RPV level lowers to -25 inches, transitions to EOP 4-2 Emergency Depressurization. **(Critical Task 4)**

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Event Description: 7 - SRV failure to open during manual operations while controlling Rx pressure with SRVs.

Cue: Red indicating light for SRV discharge pressure high not illuminated, SOLENOID A STATUS light for SRV not illuminated on P601, no change in Rx Pressure after attempting to open

Time	Position	Applicant's Actions or Behavior
	BOP	When opening SRV's in sequence listed on P601 for pressure control, SRV 1B21-F0051D fails to open. Opens next SRV to control pressure.
		Announces SRV failure to open to crew. Uses next SRV in sequence to control pressure and updates crew.
Driver	**Driver**	Role play as NLO/Maintenance/I&C if requested to respond to determine why valve failed to open

Op-Test No.: 2013-01	Scenario No.: 2 – 100%	Page 22 of 27
Event Description: 8 - Emergency Depressurization on lowering Rx level – EOP-4-2		
Cue: Contingency if level lowers to < -25"		

Time	Position	Applicant's Actions or Behavior
	SRO	Transitions to EOP 4-2 Emergency Depressurization based on control rods out and RPV level < -25 inches.
		Directs ATC to Terminate and Prevent Feedwater per Hardcard for ED
	ATC	Terminates Feedwater IAW Hardcard for ED
	SRO	Directs RO to open 8 ADS SRV's **(Critical Task 4)**
	RO	Opens 8 ADS SRV's **(Critical Task 4)**
	SRO	Directs RO's to inject to maintain 150 to 219 inch level band using outside the shroud systems when RPV pressure lowers to 140 psig.
	RO's	Inject using RHR A or feedwater booster pumps to restore and maintain RPV level in 150 to 219 inch level band.

Op-Test No.: 2013-01	Scenario No.: 2 – 100%	Page 23 of 27
Event Description: Termination Criteria		
Cue:		

Time	Position	Applicant's Actions or Behavior
		1. Control rods are being inserted with Rx < range 3 on IRM's.
		2. Maintaining level and pressure in assigned bands.
		3*. RPV depressurized if RPV level lowered to < -25 inches with control rods not inserted
		* Contingency

Op-Test No.: 2013-01	Scenario No.: 2 – 100%	Page 24 of 27
Event Description: Critical Task #1		
Cue:		

Time	Position	Applicant's Actions or Behavior
		With the failure of a MSIV automatic isolation, take action to manually isolate the Main Steam Lines. 1. Safety Significance: • Take action to prevent degradation of a barrier to fission product release. 2. Cues: • Procedural compliance. • MSL "x" MSIV position indication shows valves OPEN. 3. Measured by: • The RO places B21-F022x Control Switch in CLOSE. 4. Feedback: • Main Steam Line Tunnel temperature trend MSIV valve position indications.

Op-Test No.: 2013-01	Scenario No.: 2 – 100%	Page 25 of 27
Event Description: Critical Task #2		
Cue:		

Time	Position	Applicant's Actions or Behavior
		With a reactor scram required and the reactor not shutdown, take action to reduce power by initiating ARI to cause control rod insertion /manually inserting control rods. 1. Safety Significance: • Shutting down reactor can preclude failure of containment or equipment necessary for the safe shutdown of the plant. • Correct reactivity control. 2. Cues: • Procedural compliance. • Reactor power indication. 3. Measured by: • Observation - ARI pushbuttons armed and depressed to cause control rod insertion. 4. Feedback: • Reactor power trend. • Rod status indication.

Op-Test No.: 2013-01	Scenario No.: 2 – 100%	Page 26 of 27
Event Description: Critical Task #3		
Cue:		

Time	Position	Applicant's Actions or Behavior
		During an ATWS, when conditions are met to deliberately lower RPV level; Terminate and Prevent injection into the RPV from ECCS and Feedwater until conditions are met to reestablish injection. 1. Safety Significance: • Precludes loss of primary containment integrity and uncontrolled release of radioactivity into the environment. 2. Cues: • Procedural compliance. 3. Measured by: • Observation - With Emergency Depressurization not required and the deliberate lowering level override met (>4% power, and > 110°F Suppression Pool temperature, and >16.5" RPV level, and > 1.68# Drywell pressure or SRV open) injection systems are terminated and prevented until <4% power, or 16.5" RPV level, or SRV's closed with <1.68# Drywell pressure. 4. Feedback: • Injection system flow rates into RPV.

Op-Test No.: 2013-01	Scenario No.: 2 – 100%	Page 27 of 27
Event Description: Critical Task #4		
Cue:		

Time	Position	Applicant's Actions or Behavior
		After RPV water level drops to 0 inches, when RPV level cannot be restored and maintained above MSCRWL (-25"), RO initiates Emergency Depressurization as directed by US. 1. Safety Significance: • Maintaining adequate core cooling. 2. Cues: • Procedural compliance. • RPV level indication. 3. Measured by: • At least 5 SRV's are opened when RPV level cannot be restored and maintained above -25". 4. Feedback: • RPV pressure trend. • Suppression Pool temperature trend. • SRV open status indication.

Attachment 8
L-13-117

Form ES-301-1 (SRO)
(1 page)

Facility: Perry	Date of Examination: Feb 2013
Examination Level: SRO	Operating Test Number: 2013-01

Administrative Topic (see Note)	Type Code*	Describe activity to be performed
Conduct of Operations OT-3701-ADM_320_SRO	R, D, P	2.1.18 Ability to make accurate, clear, and concise logs, records, status boards, and reports. Importance SRO 3.8 Make a 4 hour NRC notification.
Conduct of Operations OT-3701-ADM_017SRO	R, D	2.1.36 Knowledge of procedures and limitations involved in core alterations. Importance: SRO 4.1 Determine actions for Refuel Bridge PLC failure
Equipment Control OT-3701-ADM_026_SRO	R, D	2.2.41 Ability to obtain and interpret station electrical and mechanical drawings. Importance: SRO 3.9 Determine how to isolate a leaking fire protection valve. And evaluate the impact of leaking fire protection valve.
Radiation Control OT-3701-ADM_018 OT-3701-ADM_027	R, N D, C	2.3.4 Knowledge of radiation exposure limits under normal or emergency conditions. Importance: SRO 3.7 Emergency Dose Authorization. 2.3.12 – Knowledge of radiological safety principles pertaining to licensed operator duties, such as containment entry requirements, fuel handling responsibilities, access to locked high-radiation areas, aligning filters, etc Importance: SRO 3.7 Perform the Daily Inventory of all HRS Keys Assigned to the Control Room
Emergency Plan OT-3701-ADM_315SRO	R/S, N	2.4.38 Ability to take actions called for in the facility emergency plan, including supporting or acting as emergency coordinator if required. Importance: SRO 4.4 Determine Emergency Coordinator duties during a general emergency.

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)

Attachment 9
L-13-117

JPM OT-3701-ADM-027_SRO
(6 pages)

JOB PERFORMANCE MEASURE SETUP SHEET

System:	Administrative
Time Critical:	No
Alternate Path:	No
Applicability:	SRO
K/A:	2.3.12 – Knowledge of radiological safety principles pertaining to licensed operator duties, such as containment entry requirements, fuel handling responsibilities, access to locked high-radiation areas, aligning filters, etc. Importance: SRO 3.7
Safety Function:	Radiation Control
Validated Time:	15 Minutes
References:	NOP-OP-4101 Rev 9
Required Material:	NOP-OP-4101, Access Controls For Radiologically Controlled Areas
Task:	341-519-01-03 Perform the Daily Inventory of all HRS Keys Assigned to the Control Room HRS Key Locker
Task Standard:	Perform the Daily Inventory of all HRS Keys Assigned to the Control Room and perform actions for any missing keys.

1. Setup Instructions: None

2. Location / Method: Control Room / Administrative Performance

3. Initial Condition: Plant operating at near rated power due to coast down. Preparation for the next refuel outage is in progress.

4. Initiating Cue: As the Shift Manager, conduct the daily inventory of the HRS Keys assigned to the Control Room HRS Key Locker, in accordance with NOP-OP-4101, Access Controls For Radiologically Controlled Areas.

Start: ________ **Stop:** ________

Candidate: ________________

JPM BODY SHEET

Standard: Performer obtains or simulates obtaining all materials, procedures, tools, keys, radios, etc... before performing task.

Standard: Performer follows management expectations with regards to safety and communication standards.

Step 1

NOP-OP-4101, Access Controls For Radiologically Controlled Areas

4.12 Control and Inventory of LHRA/VHRA Key

4.12.8 LHRA keys located in the Control Room, shall be limited to emergency entries and shall be maintained by the Operations Shift Manager.

1. This provision allows for the emergency access to LHRA areas. In the case of emergency entry being required and at the direction of the Operations Shift Manager, Operations personnel may utilize the emergency key to gain access to the LHRA.
2. IF utilized, THEN RPS and RPM shall be notified that the key was used AND a CR will be initiated.
3. The emergency entry LHRA key issued to the Operations Shift Manager shall be inventoried daily by the Operation Shift Manager or designee utilizing form NOP-OP-4101-03.

Critical Step: Using a copy of the HRS Barricade List conduct an inventory of the HRS Keys and Locks assigned to the Control Room HRS Key Locker.

Instructor Cue: While performing steps below, inform candidate that key 7-9" is not in the locker

Notes:
1. Obtains key for HRS Key Locker from the Control Room Key Locker (located inside the horseshoe).
2. Obtains copy of HRS Barricade List. (located next to HRS Key Locker)
3. Checks keys against the HRS Barricade List.

SAT ___ **UNSAT** ___

Comment(s):______________________________

Step 2

4.12.8 An inventory of the ready for issue keys will be performed together by oncoming and off going RP Technicians and documented on NOP-OP-4101-03.

1. IF any key is not accounted for, THEN make a RP Log entry (i.e., SOMS) AND contact RPS Supervision.
2. RPS will make a determination as to whether LHRA door(s)/padlock(s) will be re-keyed and new key(s) issued.
3. Initiate a Condition Report.

Critical Step:	Candidate contacts RP Supervision.
Instructor Cue:	Acknowledge notification that key 7-9" is not accounted for.
Notes:	The SM would follow the same steps for missing keys as RP Techs.
SAT ___	**UNSAT ___**
Comment(s):	

Step 3

Document the results of the HRS Key inventory on the HRS Key Locker Inventory (PNPP No. 8861, Attachment 4).

Standard:	Candidate completes HRS Key Locker Inventory form.
Instructor Cue:	None
Notes:	None
SAT ___	**UNSAT ___**
Comment(s):	

Terminating Cue: Candidate inventories HRS keys and notifies RP Supervision upon discovery of missing key.

Evaluation Results: **SAT_____** **UNSAT_____**

End Time _________

HIGH RADIATION SERIES BARRICADE LIST

PNPP No. 8860 Rev. 9/7/07 HPI-D4

Location: **CONTROL ROOM LHRA LOCKBOX**

Tag #	ROOM NUMBER	DOOR NUMBER	KEY #	ROOM DESCRIPTION
M-05	Various	Various	4-05	LHRA Master Key
58	Various	Various	58	HRA Master Key
V-01	X630 IFTS	Shield Door	7-01	IFTS Valve Room (key 7-02 issued via RPS) (UNLK)
V-08	FHB IFTS	Floor Plug	7-08	FHB IFTS Plug (key 7-10 issued via RPS)
V-09	DW A/L	Shield Door	7-09	Drywell Shield (key 7-07 issued via RPS)
V-12	X-ANNU IFTS	Floor Plug	7-12	X Annulus IFTS Plug room (key 7-11 issued via RPS) (UNLK)

(**UNLK**) = indicates Barricade is unlocked

Updated 05/03/2008 Page 1 of 1

HIGH RADIATION SERIES KEY LOCKER INVENTORY

NOP-OP-4101-03 Rev. 03

Locker# / Location : Shift Managers Office

DATE	TIME	PRINTED NAME	SIGNATURE	ALL KEYS ACCOUNTED FOR
				☐ YES ☐ NO
				☐ YES ☐ NO
				☐ YES ☐ NO
				☐ YES ☐ NO
				☐ YES ☐ NO
				☐ YES ☐ NO
				☐ YES ☐ NO
				☐ YES ☐ NO
				☐ YES ☐ NO
				☐ YES ☐ NO
				☐ YES ☐ NO
				☐ YES ☐ NO
				☐ YES ☐ NO
				☐ YES ☐ NO

Remarks:

REVIEWED BY: (RP Supervision or Lead Technician)	DATE:

* NOTE: This form shall be reviewed within a week of completion.

JPM CUE SHEET

INITIAL CONDITIONS:	Plant operating at near rated power due to coast down. Preparation for the next refuel outage is in progress.
INITIATING CUE:	As the Shift Manager, conduct the daily inventory of the HRS Keys assigned to the Control Room HRS Key Locker, in accordance with NOP-OP-4101, Access Controls For Radiologically Controlled Areas.