UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION III
2443 WARRENVILLE ROAD, SUITE 210
LISLE, IL 60532-4352

April 24, 2013

Mr. Vito Kaminskas
Site Vice President
FirstEnergy Nuclear Operating Company
Perry Nuclear Power Plant
P. O. Box 97, 10 Center Road, A-PY-A290
Perry, OH 44081-0097

SUBJECT: PERRY NUCLEAR POWER PLANT
NRC INITIAL LICENSE EXAMINATION REPORT 05000440/2013301

Dear Mr. Kaminskas:

On March 14, 2013, the U.S. Nuclear Regulatory Commission (NRC) completed the initial operator licensing examination process for license applicants employed at your Perry Nuclear Power Plant. The enclosed report documents the results of those examinations. Preliminary observations noted during the examination process were discussed on March 6, 2013, with you and other members of your staff. An exit meeting was conducted by telephone on April 4, 2013, between Mr. A. Mueller, Training Manager, of your staff, and Mr. M. Bielby, Chief Examiner, to review the proposed final grading of the written examination for the license applicants. During the telephone conversation, the NRC resolutions of the station's post-examination comments, initially received by the NRC on March 14, 2013, were discussed.

The NRC examiners administered an initial license examination operating test during the weeks of February 25 and March 4, 2013. The written examination was administered by Perry Nuclear Power Plant training department personnel on March 6, 2013. Nine Senior Reactor Operator and three Reactor Operator applicants were administered license examinations. The results of the examinations were finalized on April 5, 2013. Three applicants failed the senior operator portion of the administered written examination and were issued a proposed license denial letter. Nine applicants passed all sections of their respective examinations; five applicants were issued senior operator licenses; and three applicants were issued operator licenses. In accordance with NRC policy, the license for the remaining senior operator applicant is being withheld pending the outcome of any written examination appeal that may be initiated.

The written examination and other related written examination documentation will not be withheld from public disclosure. When an applicant receives a proposed license denial letter because of a written examination grade that is less than 80.0%, the applicant will be provided a copy of the written examination. For examination security purposes, your staff should consider that written examination uncontrolled and exposed to the public.

In accordance with Title 10 of the Code of Federal Regulations, Section 2.390 of the NRC's "Rules of Practice," a copy of this letter and its enclosures will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records System (PARS) component of NRC's Agencywide Documents Access and Management System (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/ By Bruce Palagi Acting For/

Hironori Peterson, Chief
Operations Branch
Division of Reactor Safety

Docket No. 50-440
License No. NPF-58

Enclosures:
1. Operator Licensing Examination Report 05000440/2013301
   w/Attachment: Supplemental Information
2. Simulation Facility Report
3. Written Examination Post-Examination Comment Resolution

cc w/encl: Distribution via ListServ™
A. Mueller, Training Manager, Perry Nuclear Power Plant

U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket No:	50-440
License No:	NPF-58
Report No:	05000440/2013301
Licensee:	FirstEnergy Nuclear Operating Company (FENOC)
Facility:	Perry Nuclear Power Plant
Location:	Perry, Ohio
Dates:	February 25 – March 14, 2013
Inspectors:	M. Bielby, Chief Examiner R. Baker, Examiner D. Oliver, Examiner
Approved by:	H. Peterson, Chief Operations Branch Division of Reactor Safety

SUMMARY OF FINDINGS

ER 05000440/2013301; 02/25/2013 – 03/14/2013; FirstEnergy Nuclear Operating Company, Perry Nuclear Power Plant; Initial License Examination Report.

The announced initial operator licensing examination was conducted by regional Nuclear Regulatory Commission (NRC) examiners in accordance with the guidance of NUREG-1021, "Operator Licensing Examination Standards for Power Reactors," Revision 9, Supplement 1.

Examination Summary

Nine of twelve applicants passed all sections of their respective examinations. Five applicants were issued senior operator licenses and three applicants were issued operator licenses. Three applicants failed the administered senior operator section of the written examination and were issued proposed license denials. The license for the remaining senior operator applicant is being held and may be issued pending the outcome of any written examination appeal(s). (Section 4OA5.1).

REPORT DETAILS

4OA5 Other Activities

.1 Initial Licensing Examinations

a. Examination Scope

The NRC examiners and members of the facility licensee's staff used the guidance prescribed in NUREG-1021, "Operator Licensing Examination Standards for Power Reactors," Revision 9-Supplement 1, to develop, validate, administer, and grade the written examination and operating test. Members of the facility licensee's staff prepared the outline and developed the written examination and operating test. The NRC examiners validated the proposed examination during the week of February 4, 2013, with the assistance of members of the facility licensee's staff. During the on-site validation week, the examiners audited three license applications for accuracy. The NRC examiners, with the assistance of members of the facility licensee's staff, administered the operating test, consisting of job performance measures and dynamic simulator scenarios, during the period of February 25 through March 5, 2013. The facility licensee administered the written examination on March 6, 2013.

b. Findings

(1) Written Examination

The NRC examiners determined that the written examination, as proposed by the licensee, was within the range of acceptability expected for a proposed examination. Less than 20% of the proposed examination questions were determined to be unsatisfactory and required modification or replacement. All changes made to the proposed written examination, were made in accordance with NUREG-1021, "Operator Licensing Examination Standards for Power Reactors," and documented on Form ES-401-9, "Written Examination Review Worksheet," which will be available electronically in the NRC Public Document Room or from the Publicly Available Records component of NRC's Agencywide Documents Access and Management System (ADAMS). On March 14, 2013, the licensee submitted documentation noting that there were seven post-examination comments for consideration by the NRC examiners when grading the written examination. Additional supporting information was submitted to the NRC examiners on April 3, 2013, based on discussions with the licensee. The post-examination comments and the NRC resolution for the post-examination comments are included in Enclosure 3 of this report. The final as-administered examination and answer key (ADAMS Accession Number ML13113A060), will be available electronically in the NRC Public Document Room or from the Publicly Available Records component of NRC's ADAMS.

The NRC examiners graded the written examination on April 5, 2013, and conducted a review of each missed question to determine the accuracy and validity of the examination questions.

(2) Operating Test

In general, the overall operating test submitted by the licensee met the NRC expectations; however, the proposed Job Performance Measure (JPM) test items did not

meet the NRC's expectations and additional attention in this area is warranted. Four JPMs were significantly modified or replaced. Two system JPMs were replaced during validation because they lacked significant evaluation steps to be performed by the applicant. One system JPM was not considered an alternate path as written, and was modified to make it alternate path. Another system JPM was changed from having the applicant verbally perform it in the control room to requiring the applicant to actually manipulate the controls in the simulator. Several other JPMs were modified to add more significance to the task required to be performed. One additional administrative JPM was developed, validated and administered to one applicant when it was discovered that the applicant had been released into a group of applicants that had completed one JPM not previously administered to the applicant. This was considered an examination security issue. Several modifications were made to the dynamic simulator scenarios to enhance evaluation of performance. Changes made to the operating test, documented in a document titled, Operating Test Comments, as well as the final as-administered dynamic simulator scenarios and JPMs, are available electronically in the NRC Public Document Room or from the Publicly Available Records component of NRC's ADAMS.

The NRC examiners completed operating test grading on April 5, 2013.

(3) Examination Results

Nine applicants at the Senior Reactor Operator (SRO) level and three applicants at the Reactor Operator (RO) level were administered written examinations and operating tests. Nine applicants passed all portions of their examinations and were issued their respective operating licenses. Three applicants failed the administered SRO portion of the written examination and were issued proposed license denials. One applicant passed all portions of the license examination, but received an SRO written test grade below 74 percent. In accordance with NRC policy, the applicant's license will be withheld until any written examination appeal possibilities by other applicants have been resolved. If the applicant's grade is still equal to or greater than 70 percent after any appeal resolution, the applicant will be issued a senior operator license. If the applicant's SRO written grade has declined below 70 percent, the applicant will be issued a proposed license denial letter and offered the opportunity to appeal any questions the applicant feels were graded incorrectly.

.2 Examination Security

a. Scope

The NRC examiners reviewed and observed the licensee's implementation of examination security requirements during the examination validation and administration to assure compliance with Title10 of the Code of Federal Regulations, Section 55.49, Integrity of Examinations and Tests. The examiners used the guidelines provided in NUREG -1021, "Operator Licensing Examination Standards for Power Reactors," to determine acceptability of the licensee's examination security activities.

b. Findings

One examination security concern was identified during the process of administering the JPM examination. The licensee's security policy was not to mix applicants that had performed JPMs with those that had not performed the same JPMs. One applicant that had completed all but one of his JPMs for the day was released from the simulator into a

sequestered classroom and was inadvertently allowed to mingle with other applicants that had completed all of their JPMs for the day. The Chief Examiner determined that there was no evidence of examination compromise based on the short time period until discovery, and interviews with the applicants that were directly involved in the incident who indicated no JPMs had been discussed. However, to insure there was no unfair advantage to the applicant, another administrative JPM was developed and validated with concurrence of the regional Branch Chief, and administered to the applicant to replace the JPM in question. This was considered to be an examination security issue.

4OA6 Management Meetings

.1 Debrief

The chief examiner presented the examination team's preliminary observations and findings on March 6, 2013, to Mr. V. Kaminskas, Site Vice President, and other members of the Perry Operations and Training Department staff.

.2 Exit Meeting

The chief examiner conducted an exit meeting on April 4, 2013, with Mr. T. Mueller, Training Manager, by telephone. The NRC's final disposition of the station's post-examination comments were disclosed and discussed with Mr. T. Mueller during the telephone discussion. The examiners asked the licensee whether any of the material used to develop or administer the examination should be considered proprietary. No proprietary or sensitive information was identified during the examination or debrief/exit meetings.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee

V. Kaminskas, Site Vice President-Nuclear
R. Brooks, Lead-Fleet Exam Development Team
H. Hanson Jr., Director-Performance Improvement
J. Kelly, Lead-Initial Licensed Operator Training
T. Mueller, Manager-Training
D. O'Donnell, Shift Manager-Facility Reviewer
J. Pelcic, Engineer-Regulatory Compliance
R. Strohl, Superintendent-Operator Training
R. Torres, Fleet Training Exam Author
J. Tufts, Manager-Operations

NRC

M. Marshfield, Senior Resident Inspector
J. Nance, Resident Inspector
M. Bielby, Chief Examiner
R. Baker, Examiner
D. Oliver, Examiner

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened/Closed

None

LIST OF ACRONYMS USED

ADAMS	Agencywide Document Access and Management System
ATWS	Anticipated Transient Without Scram
CFR	Code of Federal Regulations
DRS	Division of Reactor Safety
DG	Diesel Generator
ECCS	Emergency Core Cooling Systems
EOF	Emergency Operating Facility
EOP	Emergency Operating Procedures
ER	Examination Report
ERO	Emergency Response Organization
HCL	Heat Capacity Limit
HPCS	High Pressure Core Spray
IAW	in accordance with
JPM	Job Performance Measures
LOCA	Loss of Coolant Accident
LOOP	Loss-of-Offsite-Power
LPCI	Low Pressure Coolant Injection
LPCS	Low Pressure Core Spray
MSIV	Main Steam Isolation Valve
NRC	U.S. Nuclear Regulatory Commission
OPRM	Oscillation Power Range Monitor
PARS	Publicly Available Records System
RHR	Residual Heat Removal
RO	Reactor Operator
RPV	Reactor Pressure Vessel
SAG	Severe Accident Guideline
SDC	Shutdown Cooling
SRO	Senior Reactor Operator
SRV	Safety Relief Valve

SIMULATION FACILITY REPORT

Facility Licensee: Perry Nuclear Power Plant

Facility Docket No: 50-440

Operating Tests Administered: Weeks of February 25 and March 4, 2013

The following documents observations made by the NRC examination team during the initial operator license examination. These observations do not constitute audit or inspection findings and are not, without further verification and review, indicative of non-compliance with 10 CFR 55.45(b). These observations do not affect NRC certification or approval of the simulation facility other than to provide information which may be used in future evaluations. No licensee action is required in response to these observations.

During the conduct of the simulator portion of the operating tests, the following items were observed:

ITEM	DESCRIPTION
Simulator Lockup, CR 2013-02777	During the examination administration of the second set of Scenario 1 on February 25, 2013, a simulator lockup (IO quit responding) occurred after Event 2, Raise Reactor Power With Flow to 100%. Simulator was re-booted and reset to just before the lockup point and the scenario resumed. There were no further lockup occurrences during the examination administration.

POST-EXAMINATION COMMENTS WITH NRC RESOLUTION

RO Question 40:

The plant was operating at rated power.

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

Only 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. not automatically restart

b. automatically restart immediately

c. automatically restart in 10 seconds

d. automatically restart in 15 seconds

Answer: a.

(The following candidate comment and station proposed resolution are directly copied from the post-examination comment submission.)

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 Justification:

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

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.

POST-EXAMINATION COMMENTS WITH NRC RESOLUTION

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

References: Plant drawings 208-0060-00004, Revision AA, and 208-0060-00008, Revision DD.

Station Proposed Resolution:

The station agrees with the above candidate Comments, Justification and Recommendation.

NRC Resolution:

The NRC agrees with the applicant's comment. The question basically asks how the LPCS pump will start when power restored by the Diesel Generators after the LPCS pump was placed in STOP followed by a LOOP/LOCA occurrence. Reference to the Power Monitor and LPCS Pump 1E21-C001, Control portion of plant drawing, 208-0060-00004, and plant drawing 208-0060-00008, support the following relay actuations and contact development which indicate an automatic LPCS pump start without time delay:

INITIAL CONDITIONS

- 1R22-27X1A and S13 closed energizing K1 and K1B.
- Pump control switch (S6) in AUTO; K13 is deenergized (Manual Override is NOT active).
- K1 contact M1-T1 is closed; K10 contact M2-T2 is open (no LOCA); K13 contact M1-R1 is closed (M1-T1 is open) resulting in relays K12 and K12A being deenergized.
- K1B contact (3-5) is open (opened 19 seconds after bus was energized); K12 contact (1-5) is open; K12A contact M3-T3 is open.

INADVERTENT LOCA (seals in)

- K10 contact M2-T2 closes energizing relays K12 and K12A; K12A contact M3-T3 closes immediately but does not start pump since K1B contact 3-5 is open; K12 contact 1-5 closes after a 15 second time delay, starting the LPCS pump.
- Per ONI immediate actions pump is stopped (Manual Override) by momentarily placing pump control switch (S6) in STOP tripping the pump breaker and removing the auto start signal by energizing relay K13; K13 contact M1-T1 closes (seals in Manual Override) and K13 contact M1-R1 opens deenergizing relays K12 and K12A opening relay K12 contact 1-5 and K12A contact 3-5.

SUBSEQUENT LOOP (LOCA signal still sealed in)

- 1R22-27X1A opens de-energizing relays K1 and K1B; relay K1 contact M1-T1 opens de-energizing relay K13; K13 contact M1-T1 opens (breaking the seal-in on for the Manual Override) and K13 contact M1-R1 closes.

OFFSITE POWER RESTORED

- 1R22-27X1A and S13 closed energizing K1 and K1B.
- Pump control switch (S6) in AUTO; K13 is deenergized (Manual Override is NOT active).
- K1 contact M1-T1 is closed; K10 contact M2-T2 is closed (LOCA is sealed-in); K13 contact M1-R1 is closed (M1-T1 is open) Relays K12 and K12A energize;

K12 contact is open (closes after 15 seconds), but K12A contact M3-T3 closes; the LPCS pump starts without any delay.

Additionally, the NRC reviewed the questions asked by the applicants concerning this question and found that there were none asked. The answer key was amended to accept distracter b. as the only correct answer.

POST-EXAMINATION COMMENTS WITH NRC RESOLUTION

RO Question 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.

	(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

Answer: c.

The following candidate comment and station proposed resolution are directly copied from the post-examination comment submission.

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 100 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 Justification:

		(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°F > 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°F > 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.

Station Proposed Resolution:

Station agrees with the above candidate Comments, Justification and Recommendation.

NRC Resolution:

The NRC agrees with the candidate's comment and station proposed resolution. After reviewing the calculations it was discovered that the cooldown rate will exceed the Technical Specification limit at the time specified by the candidate comment. Additionally, the NRC reviewed the questions asked by the applicants concerning this question and found that none were asked. The answer key was amended to accept distracter a. as the only correct answer.

POST-EXAMINATION COMMENTS WITH NRC RESOLUTION

SRO Question 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

Answer: b.

The following candidate comment and station proposed resolution are directly copied from the post-examination comment submission.

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 Justification:

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

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

		(1)	(2)
a.	Incorrect answer	SAG-1, Primary Containment Flooding	Continued
		Correct – Primary Containment (SG-1) and Radioactivity Release Control guideline (SAG-2) are entered and executed concurrently	Incorrect – EOP Actions are discontinued when SAGs are entered.
b.	Correct answer	SAG-1 Primary Containment Flooding	Exited
		Correct – Primary Containment (SG-1) and Radioactivity Release Control guideline (SAG-2) are entered and executed concurrently	Correct – EOP Actions are discontinued when SAGs are entered.
c.	Incorrect answer	SAG-2, RPV, Containment, and Radioactivity Release Control	Continued
		Correct – Primary Containment (SAG-1) and Radioactivity Release Control guideline (SAG-2) are entered and executed concurrently	Incorrect – EOP Actions are discontinued when SAGs are entered.
d.	Correct answer	SAG-2, RPV, Containment, and Radioactivity Release Control	Exited
		Correct – Primary Containment (SAG-1) and Radioactivity Release Control guideline (SAG-2) are entered and executed concurrently	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, EOP-1 pg 61, SAG bases, SAG-1 bases pg 6, and SAG-2 pg 8.

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.

NRC Resolution:

The NRC disagrees with the candidate's comment/justification and station proposed resolution that both answers b. and d. are correct. The question asks for the required Shift Manager actions after the determination that adequate core cooling no longer exists and the required status of the Emergency Operating Procedures (EOPs) when transitioning to the Severe Accident Guidelines (SAGs). The EOP-1, RPV Control; EOP-1A, Level Power Control; and EOP 04-4, RPV Flooding, flowcharts do not direct concurrent entry into SAG-1, Primary Containment Flooding, and SAG-2, Primary Containment and Radioactivity Release Control, as stated by the candidate and supported by the station proposed resolution. The EOP-1, EOP-1A and EOP 04-4 flowcharts direct transition to Primary Containment Flooding which is SAG-1. Once the SAG-1 flowchart is entered, a subsequent step directs entry into SAG-2 and both SAGs are performed concurrently. Once transition to the SAGs is performed, the EOPs are exited.

Additionally, the NRC reviewed the questions asked by the applicants concerning this question and found that none were asked. Based upon the above discussion, the NRC did not modify the answer key and retained choice b. as the only correct answer to the question.

SRO Question 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 _____.

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

Answer: a.

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(sic)/Hr core flow.
   - SPDS screen CFLOWV.
3. The procedure direct 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 Justification:

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-C51 FLOWCHART or by ONI-SPI G-2. ONI- SPI G-2 directs closure of the B Recirc Pump Suction valve.

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

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."

NRC Resolution:

The NRC disagrees with both 1) the candidate's comment/justification, to delete the question because an actual core flow value following the Reactor Recirculation Pump (RRP), which is not provided, is required in order to determine and perform any actions in accordance with (IAW) ONI-C51; and 2) the facility's proposed resolution to delete the question because this was a question with an unclear stem that confused the applicants or did not provide all the necessary information.

The question asked which of the four options listed would be directed by the Unit Supervisor, given the plant conditions stated in the stem and the direction provided by ONI-C51. Since no additional failures beyond the trip of the 'B' RRP were presented, and the plant is expected to respond as anticipated for a loss of one RRP, entry into ONI-C51 is required due to the RRP trip. ONI-C51 directs performance of the immediate and supplemental actions IAW ONI-C51 flowchart revision J. Once immediate actions have been verified complete, steps C51-2, C51-3, and C51-4 are completed in series. These actions may be performed without an actual core flow value being known since core flow will be greater than 42 Mlbm/hr based upon the conditions provided in the stem. The flowchart then directs the supplemental actions of steps C51-4 and C51-12 be executed concurrently. Step C51-12 directs the operator to **GO TO** Step C51-21 if either RRP has tripped and single loop operation is required, which will subsequently direct the operator to verify loop parameters within limits for single RRP operation and, within 1 hour insert control rods to lower reactor power below 66.5% (2500 MWth). None of these required supplemental actions necessitate knowing the actual core flow prior to performance.

As discussed above, the stated assumption made by the applicant that an actual core flow value is required before any actions may be performed IAW ONI-C51 is incorrect. Therefore, the NRC does not agree with deleting the question because no action IAW ONI-C51 is possible without being given a value for actual core flow. Also, even though having the core plate ΔP value following the RRP trip, along with the curve, PDB-A0015, would allow an actual core flow

value to be determined, this would simply be additional information to confirm the expected position on the Power-to-Flow map. Therefore, the NRC does not agree with the licensee's resolution to delete the question because the question did not provide all necessary information in the stem and the stem was unclear and confused the applicants. The only related question asked by any of the applicants during the examination dealt with a clarification on completion of the required Technical Specification actions for the inoperable Oscillating Power Range Monitors (OPRMs); no questions were raised concerning actual core flow values.

Although the NRC believes that adequate information to answer the question was provided in the question stem, the NRC has now identified an issue stemming from the validation of the question when the original examination was submitted and approved. Based upon the station's training, the applicants were expected to recognize that the trip of the 'B' RRP would lower core flow to less than 50% rated core flow, based upon the initial condition given that core flow was 103 Mlbm/hr. The applicant would have to assess the situation, using the provided PDB-A06, Power-to-Flow Map (modified), realize this would result in operation in the Controlled Entry/Immediate Exit Region of the Backup Stability Protection (OPRM INOP) Power-to-Flow map, and direct actions IAW the ONI-C51 Flow Chart – Insert cram rods IAW FTI-B002, Control Rod – to lower power to exit this region. Per the NRC's request, the licensee provided additional reference material that indicates the expected actual plant core flow value will be in excess of 50%, and operation will be outside and to the right of the Controlled Entry/Immediate Exit Region of the Backup Stability Protection (OPRM INOP) Power-to-Flow map. The correct action to be directed by the operator IAW ONI-C51 is to verify loop parameters within limits for single RRP operation and, within 1 hour insert control rods as necessary to lower reactor power below 66.5% (2500 MWth). However, this action was not listed as one of the distracters for this question.

Based upon the newly discovered technical information, this question does not have a correct answer, and the NRC has changed the answer key to delete this question.

SRO Question 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.
- MSIVs are shut.
- RPV water level is 65 inches and stable.
- RPV pressure is 960 psig and stable.
- 2 SRVs 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 MSIVs IAW EOP-SPI 9.2, Opening MSIVs

c. Transition to EOP 4-2, Emergency Depressurization

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

Answer: a.

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-5. OAI-1703, Hardcard states "ED required at **OR** 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 SRVs 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 Justification:

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 MSIVs 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 no appropriate during an ATWS.

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

Station Proposed Resolution:

The station 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 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.

NRC Resolution:

The NRC agrees with the licensee proposed resolution that there is only one correct answer to this question. The candidate incorrectly states that plotting the given conditions would place HCL in the UNSAFE region of the HCL curve. As stated by the station proposed resolution 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 by the Unit Supervisor. The first of these requires opening an additional SRV to lower RPV pressure to restore RPV pressure within the pressure band and below the HCL limit. As stated by the station, the question stem indicates that the lowering suppression pool level is well above the required Emergency Depressurization level of 14.25 feet.

As stated by the station, Emergency Depressurization is required 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.

Additionally, the NRC reviewed the questions asked by the applicants concerning this question and found that none were asked. The answer key was not modified and choice a. is the only correct answer to the question.

SRO Question 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

Answer: a.

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 effect 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 Justification:

a. Correct – RPV water level does exhibit a rising trend within the 10 minute time frame due (SIC) 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.

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 10 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.

NRC Resolution:

The NRC agrees that choice a. is the only correct answer to this question. The question asks for two distinct pieces of information, (1) what will be the effect on RPV level over the next 10 minutes due to the effects of shrink and swell; and (2) which is the correct procedure used to control RPV level given the plant conditions stated in the stem. Distracters b. and d. are incorrect because they specify an incorrect procedure (EOP-1A, Level Power Control) for controlling RPV level. The RPV level will change due to two discrete effects. One is a change

in the inventory of fluid in the RPV, i.e., steam being removed by the MSL drains or water injected by high pressure sources. The other is a change in RPV level due to a change in the RPV water's density from changes in the water's temperature. As the colder water injected by HPCS is heated, and RPV pressure and temperature rise, RPV level will continue to rise. Since the question only asks for the change in RPV level due to the effects of shrink and swell, any change in RPV level due to a change in inventory does not factor in to the answer.

Additionally, the NRC reviewed the questions asked by the applicants concerning this question and found that one applicant asked if he was predicting the correct answer to the question based on the terms "shrink" and "swell." The other applicant asked if RCIC had initiated. The proctor directed the applicant in each case to attempt to answer the question based on the information provided in the question stem.

Based upon the above discussion, the NRC did not modify the answer key and retained choice a. as the only correct answer to the question.

SRO Question 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 de-energized 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.

Answer: d.

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 no 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 Justification:

a. **Correct answer** – LPCI A is NOT required to 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 de-energized 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'

Station Proposed Resolution:

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

NRC Resolution:

The NRC disagrees with both 1) the candidate's comment/justification, that since only 2 ECCS injection/spray subsystems are required to be OPERABLE given the conditions stated in the stem, the LPCI mode of RHR A system is NOT affected; and 2) the station's proposed resolution to accept two correct answers based upon newly discovered technical information that supports a change in the answer key.

The question asks whether, based upon the conditions given in the stem, the LPCI mode of RHR A system is considered OPERABLE per Technical Specifications. Just because a subsystem is not required to be OPERABLE for a given plant condition does not mean the "operability" of the subsystem is NOT affected by the given plant condition. Therefore, the candidate's comment/justification is incorrect. The licensee submitted the associated portions of the facility's Technical Specifications, including the bases, for ECCS systems – Shutdown, which does not provide any additional nor newly discovered technical information to support a change to the answer key. Additionally, the NRC reviewed the questions asked by the applicants concerning this question and found that none were asked.

Based upon the above discussion, the NRC did not modify the answer key and retained choice d. as the only correct answer to the question.

In accordance with Title 10, Code of Federal Regulations (CFR), Part 50, Section 2.390 of the NRC's "Rules of Practice," a copy of this letter, its enclosure, and your response (if any), will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records System (PARS) component of NRC's Agencywide Documents Access and Management System (ADAMS), accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/ By Bruce Palagi Acting For/

Hironori Peterson, Chief
Operations Branch
Division of Reactor Safety

Docket No. 50-440
License No. NPF-58

Enclosures:
1. Operator Licensing Examination Report 05000440/2013301 w/Attachment: Supplemental Information
2. Simulation Facility Report
3. Written Examination Post-Examination Comment Resolution

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Letter to Mr. Vito Kaminskas from Mr. Hironori Peterson dated April 24, 2013.

SUBJECT: PERRY NUCLEAR POWER PLANT
NRC INITIAL LICENSE EXAMINATION REPORT 05000440/2013301

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