August 22, 2003

Mr. Roy Anderson
Chief Nuclear Officer and President
PSEG Nuclear LLC - N09
Hope Creek Generating Station
P. O. Box 236
Hancocks Bridge, NJ 08038

SUBJECT: HOPE CREEK GENERATING STATION REACTOR OPERATOR AND SENIOR REACTOR OPERATOR INITIAL EXAMINATION REPORT NO. 05000354/2003302

Dear Mr. Anderson:

This report transmits the results of the reactor operator (RO) and senior reactor operator (SRO) licensing examinations conducted by the NRC during the period of June 16 - 25, 2003. This examination addressed areas important to public health and safety and was developed and administered using the guidelines of the “Examination Standards for Power Reactors” (NUREG-1021, Revision 8, Supplement 1).

Based on the results of the examination, two of three RO applicants and five of six SRO applicants passed all portions of the examination. One RO applicant failed the written exam and simulator scenario category of the operating test; one SRO applicant failed the administrative category of the operating test. On August 6, 2003, final examination results, including individual license numbers, were given during a telephone call between Mr. T. Fish and Mr. J. Reid and others of your staff.

There were a number of changes to the written exam after it was administered that exceeded certain thresholds of the Examination Standards. Your staff submitted comments on ten questions to be considered for changes (Attachment 2 of the enclosed report). NRC examiner staff evaluated those questions and, where appropriate, incorporated the comments (Attachment 3 of the enclosed report). Final revisions resulted in the following: 1) three questions (two common and one SRO-only) were deleted; 2) two common questions had two correct answers; and 3) four questions (three common and one SRO-only) required answer key changes or corrections. Overall, these revisions resulted in changes to 7% of the RO test and 9% of the SRO test and indicated a problem in the quality of review by your staff. We also noted similar quality-of-review problems for the 2002 Hope Creek Initial Examination.

Accordingly, as noted in NUREG-1021, Section ES-501, item C.2.c., we would like to meet with your training staff to discuss corrective actions related to the problem on exam quality. In a telephone conversation with Mr. Conicella of your staff, we agreed that the meeting would be on a mutually agreeable date in mid to late October 2003. At the meeting we request that your staff provide your organization’s perspective on the problem, including why so many changes were necessary, and what actions, if any, have been taken or will be taken to improve future initial licensing examinations. In addition, because of the apparent repetitive nature of the problem, we would appreciate your perspective on why previous actions to improve initial license exam quality were not effective.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter and its enclosure will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's document system (ADAMS). These records include the final examination and are available in ADAMS (SRO/RO Written-Accession Number ML032320112; SRO/RO Operating Section A-Accession Number ML032320127; SRO/RO Operating Section B-Accession Number ML032320162; and SRO/RO Operating Section C-Accession Number ML032320167). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/ADAMS.html (the Public Electronic Reading Room).

Should you have any questions regarding this examination, please contact me at (610) 337-5183, or by E-mail at RJC@NRC.GOV.

Sincerely,

/RA/

Richard J. Conte, Chief
Operational Safety Branch
Division of Reactor Safety

Docket No. 50-354
License No. NPF-57

Enclosure: Initial Examination Report No. 05000354/2003302 with Attachments

cc w/encl:
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DATE	08/19/03		08/19/03		08/19/03		08/20/03			

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REGION I

Docket No:	05000354
License No:	NPF-57
Report No:	2003-302
Licensee:	PSEG Nuclear LLC
Facility:	Hope Creek Generating Station
Dates:	June 16 - 25, 2003 (Operating Test Administration) June 23, 2003 (Written Examination Administration) July 2 - July 9, 2003 (Examination Grading and Evaluation of Facility Post Exam Comments)
Examiners:	T. Fish, Sr. Operations Engineer (Chief Examiner) H. Williams, Sr. Operations Engineer G. Johnson, Operations Engineer
Approved by:	Richard J. Conte, Chief Operational Safety Branch Division of Reactor Safety

Enclosure

SUMMARY OF FINDINGS

IR 05000354/2003-302; June 16 - 25, 2003; Hope Creek Generating Station; Initial Operator Licensing Examination Report.

Cornerstone: Mitigating Systems

Five of six SRO and two of three RO applicants passed all portions of the examinations. The written examinations were administered by the facility and the operating tests were administered by three NRC examiners. Because of the relatively large number of post exam changes to the written exam (in excess of 5%), the quality of the initial submittal was considered problematic.

Report Details

1. REACTOR SAFETY

Mitigating Systems - Reactor Operator (RO) and Senior Reactor Operator (SRO) Initial License Examinations

a. Scope of Review

The Hope Creek examination team developed the written and operating initial examinations and together with NRC personnel, verified or ensured, as applicable, the following:

- The examination was prepared and developed in accordance with the guidelines of Revision 8, Supplement 1 of NUREG-1021, "Operator Licensing Examination Standards for Power Reactors" and it met the overall quality goals (range of acceptability) of these standards. The review was conducted both in the Region I office and at the Hope Creek power plant and training facility. Final resolution of comments and incorporation of test revisions was conducted during and following the onsite preparation week.

- Generally, simulation facility operation was proper. During administration of the operating test the week of June 16, 2003, the simulator experienced an unexpected loss of the control room information display system (CRIDS) due to a lightning strike on June 17th. The examiners resumed the operating test June 20, and completed its administration June 25.

- Facility licensee completed a test item analysis on the written examination for feedback into the systems approach to training program.

- Examination security requirements were met.

The NRC examiners administered the operating portion of the examination to all applicants from June 16 - 25, 2003. Hope Creek training staff administered the written examination on June 23, 2003.

b. Findings

Grading and Results

Seven of nine applicants (2 ROs and 5 instant SROs) passed all portions of the initial licensing examination.

Examination Preparation and Quality

During the pre-exam NRC review, the draft exam met the quality tolerances of the examiners standards. However, the licensee submitted ten post-examination comments for the written exams (Attachment 2). Where appropriate, these comments were incorporated (Attachment 3). The revisions resulted in the following: 1) three questions (two common and one SRO-only) were deleted; 2) two common questions had two correct answers; and 3) four questions (three common and one SRO-only) required answer key changes or corrections. Overall, these revisions resulted in changes to 7% of the RO test and 9% of the SRO test. The region verified no impact on the written test outline sampling plan.

Subsequently, Hope Creek wrote two notifications related to the initial exam process:

(1) Notification 20153937 - High attrition rate among 2003 HC initial operator class (about half of the applicants did not successfully complete the course); and

(2) Notification 20153950 - Written examination post exam changes above NRC threshold of 5% changes. A total of 9 of 126 exam items required changes.

<u>Examination Administration and Performance</u>

NRC examiners did not note generic performance errors by the applicants during examination administration.

4. OTHER ACTIVITIES

4OA6 <u>Meetings, including Exit</u>

On July 2, 2003, the licensee submitted post examination comments. On July 10, the NRC evaluation period for comments ended. On July 25, NRC resolution of facility comments was discussed with licensee representatives. On July 28, the licensee reported no challenges to NRC resolution of comments.

On August 6, the NRC provided observations and examination results to Hope Creek training personnel, via telephone. License numbers for the applicants who passed the exam were provided during this call. The NRC also expressed appreciation for the cooperation and assistance the licensee's training staff provided during the preparation and administration of the examination.

ATTACHMENT 1

KEY POINTS OF CONTACT

J. Reid	Superintendent, Operations Training
N. Conicella	Supervisor, HC Licensed Operator Training
A. Faulkner	Lead, Licensed Operator Training
D. Rein	Licensed Operator Training

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

None.

ATTACHMENT 2

FACILITY COMMENTS ON WRITTEN EXAM

Record 23
RO Question 18
SRO Question 20

Given the following conditions:

- An ATWS occurs from 100 percent power.
- As corrective actions are being taken, the MSIVs inadvertantly isolate from a spurious high steam tunnel temperature signal.
- Other MSIV closure interlocks are clear.
- Visual inspection of the steam tunnel show NO abnormalities.
- Main condenser vacuum is 3 InHgA.
- Reactor coolant activity levels are normal.

Based on these conditions, which one of the following will allow the MSIVs to be re-opened?

a. Reactor power is 10 percent.

b. Suppression pool temperature is rising towards HCTL.

c. RPV Level is less than -129 inches.

d. Emergency depressurization is anticipated.

Answer: a

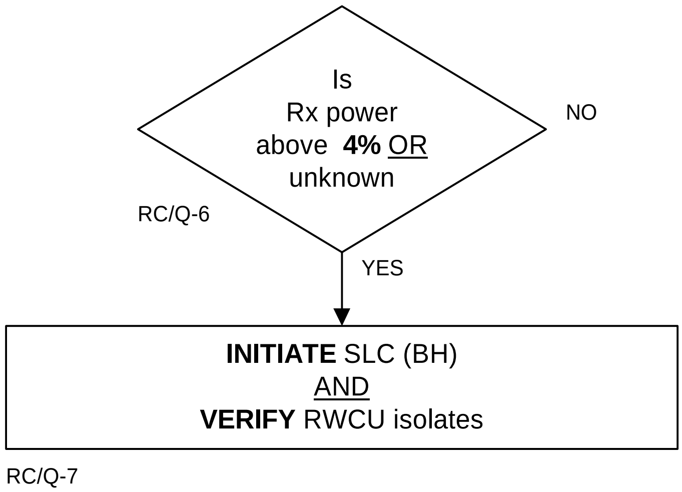
The premise of the questions asks basically, 'what allows reopening of the MSIVs during an ATWS?" The answer is directed by the Retainment Override step in EOP-0101A RC/P-17.

IF while executing the following steps:	THEN:
Boron injection is required AND Main condenser is available AND There is **NO** indication of gross fuel failure OR steam line break	**OPEN** MSIVs to reestablish the main condenser as a heat sink, **BYPASS** low RPV water level isolation interlocks using, if necessary • MSIVs using OP-EO.ZZ-301 • PCIG using OP-EO.ZZ-311 • Instrument Air using OP-EO.ZZ-319
SRVs are being used to stabilize pressure AND PCIG is OR becomes unavailable	**PLACE** the control switch for each SRV in the CLOSE OR AUTO position.

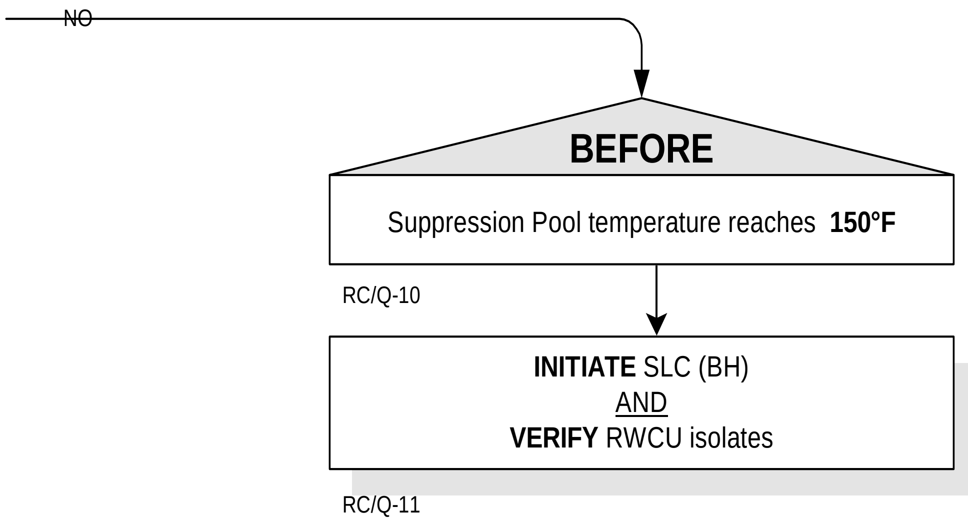
RC/P-17

The stem of the question satisfies the Main condenser availability, status of the Fuel and integrity of the steam lines. The candidates had to determine when Boron Injection was required.

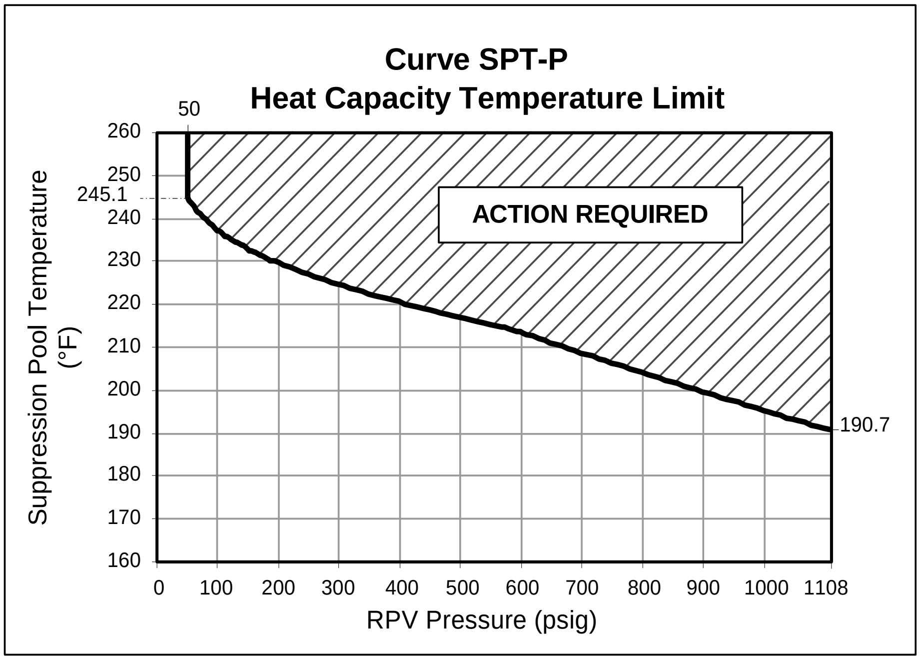
Answer “a” - “Reactor Power is 10 percent” is a correct answer because boron injection is required. (**See RC/Q-6 below)**



The **NO** answer to RC/Q-6 requires Boron Injection if the Suppression Pool temperature is approaching the Boron Injection Initiation Temperature (**see RC/Q-10 below)**



The choice “b” - ‘Suppression Pool temperature is rising towards HCTL’ was intended to be a viable distractor, but if we view the HCTL curve below,



we see that to be on the curve approaching the limit, we are well above the 150 °F required for Boron Injection Initiation temperature requirement of step RQ/Q-10. Therefore Choice “b” is also correct.

Recommendation: Change answer key to both “a” and “b” as correct answers.

References: HC.OP-EO.ZZ-0101A

Record #35
SRO Question #31

Given the following conditions:

- The plant is several hours into a LOCA.
- HPCI automatically initiated and then subsequently tripped on low oil pressure.
- A & B RHR loops are NOT available.
- All other available ECCS are injecting.
- Drywell pressure is 64.4 psig and rising.
- HPCI Pump suction pressure is 73 psig.
- SP level indication is failed.
- SP temperature is 175 F.

What is containment water level and based on that level, which of the following actions are required?

a. 21.2 ft; Vent the Suppression Pool.

b. 22.0 ft; Vent the Suppression Pool.

c. 21.2.ft; Vent the Drywell.

d. 22.0 ft; Vent the Drywell.

Answer: b

The question required the students to calculate Primary Containment Water level based on plant conditions. The stem of the question states all Suppression Pool Water Level indication is lost. Determination of Suppression Pool Water level, directs the Venting of the Primary Containment using the Suppression Pool path or the Drywell path. The EOPs use Suppression Pool indicated level up to 180 inches, which is the top of the instrument's range. If unable to determine if level is below 180" the EOPs direct a less desirable vent path of the Drywell (**see DW/P-12 below)**

Is
Supp Pool
Level below
180 in.

NO

DW/P-12

YES

BEFORE

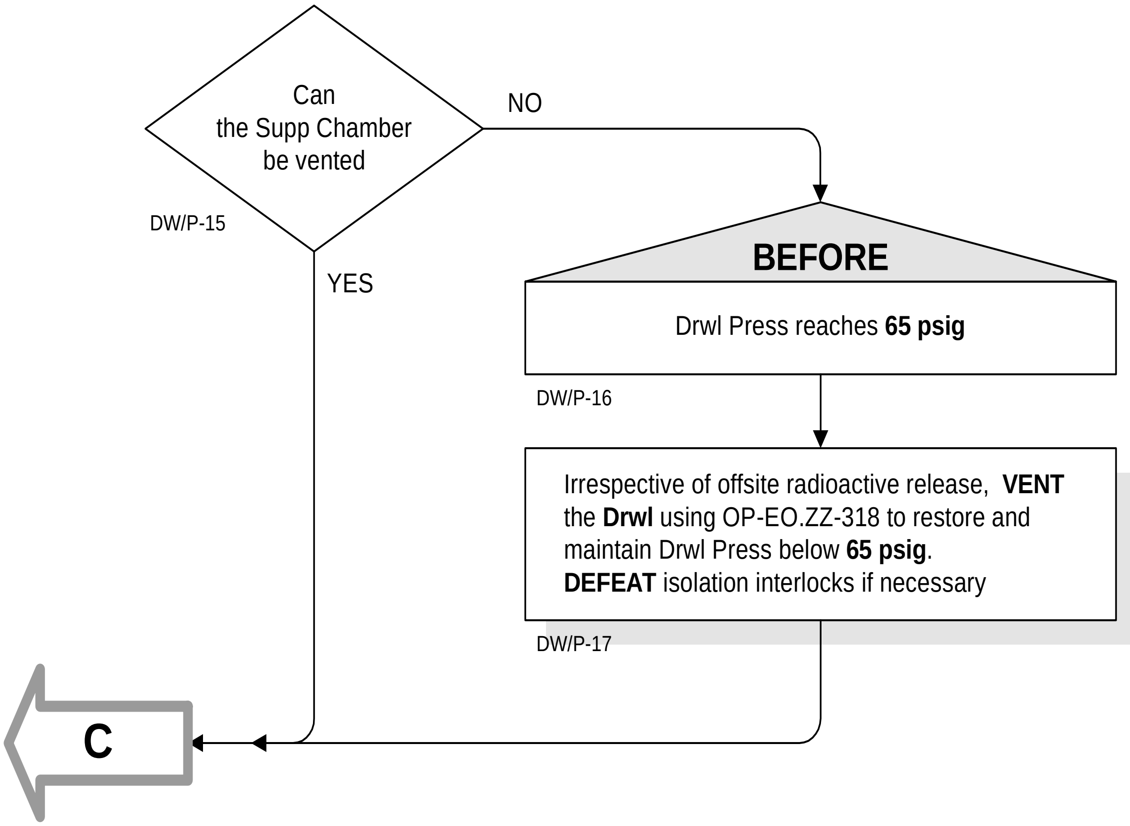
Drwl Press reaches **65 psig**

DW/P-13

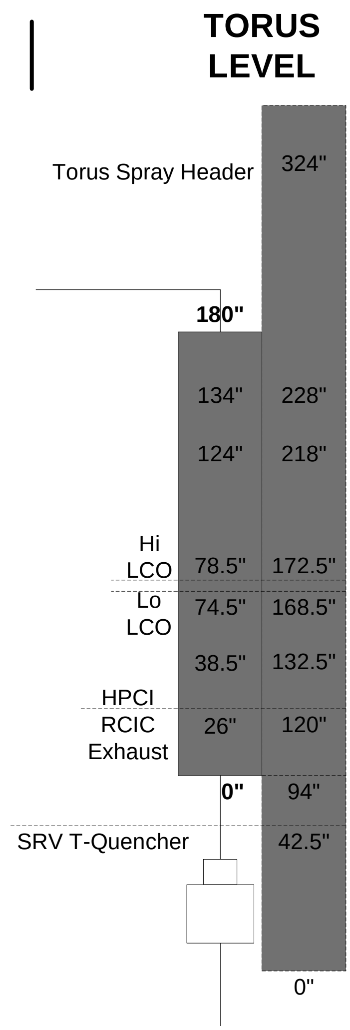
Irrespective of offsite radioactive release, **VENT** the **Supp Chamber** using OP-EO.ZZ-318 to restore and maintain Drwl Press below **65 psig**
DEFEAT isolation interlocks if necessary

DW/P-14

The **NO** answer to either DW/P-12 or DW/P-15 requires performing steps DW/P-16 and DW/P-17, a less desirable vent path using the Drywell (**see below**).


Can
the Supp Chamber
be vented
DW/P-15
NO
YES
BEFORE
Drwl Press reaches **65 psig**
DW/P-16
Irrespective of offsite radioactive release, **VENT** the **Drwl** using OP-EO.ZZ-318 to restore and maintain Drwl Press below **65 psig**.
DEFEAT isolation interlocks if necessary
DW/P-17
C

The Containment water level calculation is referenced to the bottom of the Torus, Suppression pool indicated level is based on the range and location of the level indication. Figure HC.EP-AM.ZZ-0001, RPV & Containment Information displays the relationship between the containment level and suppression pool indicated level. The RPV & Containment Information chart **was not provided to the students**. (**see below**)



Students are not expected to correlate between Primary Containment level, as calculated, and Suppression Pool indicated level from memory, this type of calculation and determination of correct vent path would be expected to come from the Technical Support Center (TSC) during Emergency Plan Activation.

All candidates chose to vent the Primary Containment from the Drywell path because they were unable to determine if Suppression Pool Water level was below the 180-inch limit. Without the available figure to determine actual Suppression Pool level or direction from the TSC the correct and only choice was to vent the Primary Containment from the Drywell versus the Suppression Pool.

Recommend - Change answer key to “d” as the correct answer
References: HC.OP-EO.ZZ-0102, HC.EP-AM.ZZ-0001
Record Number 43
RO Question # 32

SRO Question# 38

Given the following conditions:

- The reactor is operating at 100% power.
- Annunciator B1-B3 (RCIC PUMP ROOM FLOODED) alarms with the following alarm message presented on the CRIDS display: D2887 RCIC PUMP RM 4110 LSH 4151-1 HI.
- An investigation reveals that Reactor Building Floor Drain Sump pumps have been running continuously for 20 minutes.
- The Reactor Building Operator reports the RCIC, B and D RHR Pump rooms have about 6 inches of water on the floor when he checked the elevation.
- CST level is lowering.

In addition to running the sump pumps, which of the following action(s), if any, is required by EOP 103/4?

I --- Isolate RCIC
II -- Immediately commence a normal reactor shutdown
III -- Runback reactor recirculation and manually scram the reactor
IV - Emergency depressurize the reactor

a. I - ONLY

b. II - ONLY

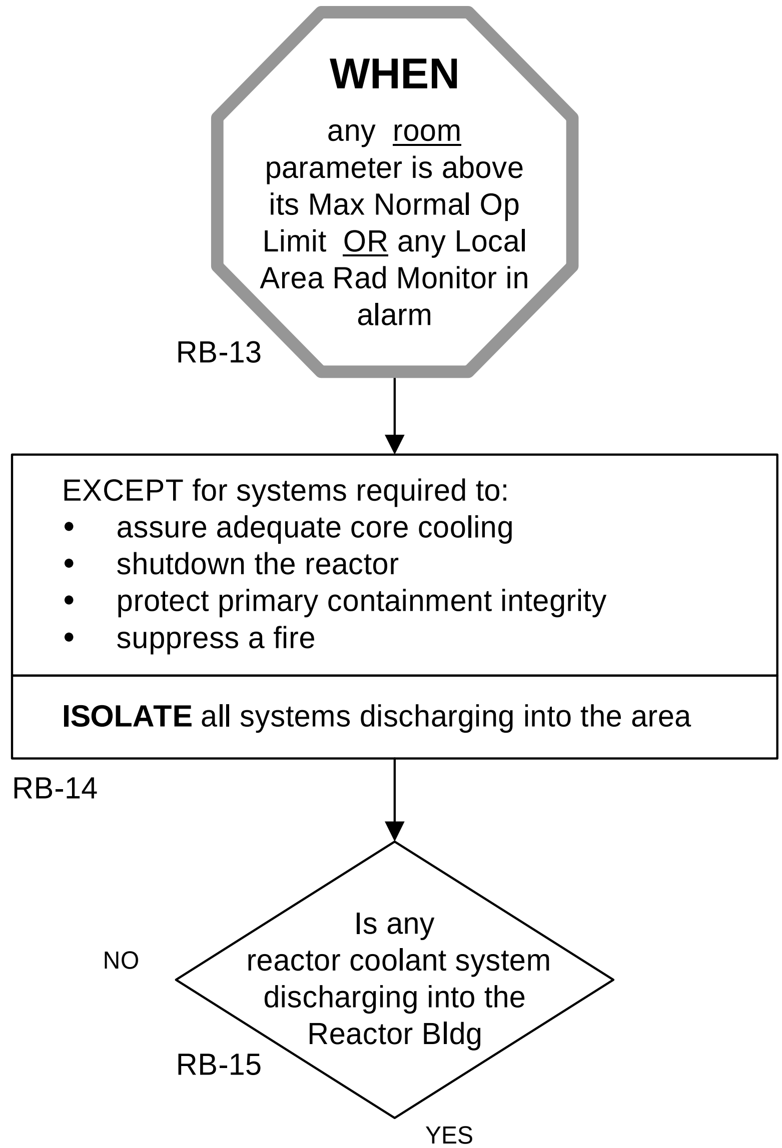
c. I and II

d. I, III, and IV

Answer: c

The question identified a leak into RCIC was occurring and had flooded the RCIC room and the B and D RHR room. The implication was the source being the CST, a NON Reactor Coolant System. The question asked the required actions for this condition.

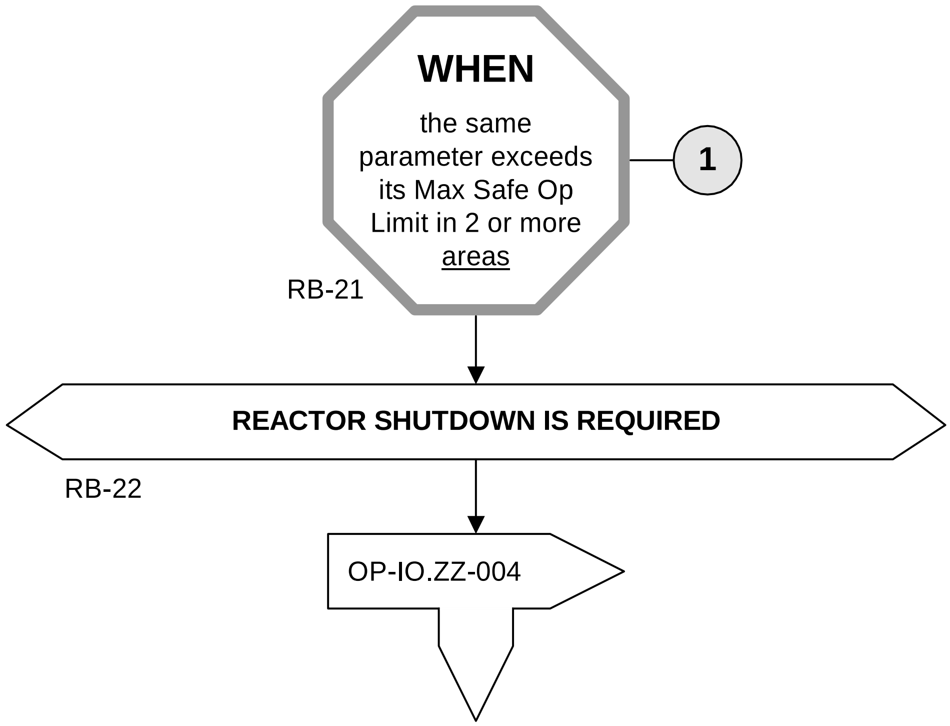
Step RB-14 of EOP-103/4 requires isolating systems discharging to the area.
Step RB-15 of EOP-103/4 requires the determination if a Reactor Coolant System is discharging into the areas affected. (**see below**)


WHEN
any room parameter is above its Max Normal Op Limit OR any Local Area Rad Monitor in alarm
RB-13
EXCEPT for systems required to:
• assure adequate core cooling
• shutdown the reactor
• protect primary containment integrity
• suppress a fire
ISOLATE all systems discharging into the area
RB-14
Is any reactor coolant system discharging into the Reactor Bldg
NO
RB-15
YES

The **“NO”** response based on a leak from the CST (Non Reactor Coolant System) leaking in the Reactor Building directs actions per RB-21.

The question stem conditions based on visual observation and Reactor Building sump pump run times **placed two areas** above the Max Safe Operating Limit.

Per step RB-22 of EOP-103/4 an immediate shutdown is required. (**see below**)



The identified answer “c” was to isolate RCIC and immediately commence a reactor shutdown.

However, the choice of “Isolate RCIC” was ambiguous because it was not clear from the question that “Isolating RCIC” would isolate the leak. Two students asked if Isolating RCIC would isolate the leak. The proctor’s response was “Answer the question to best of your ability with conditions in the stem.” **See “Exam Proctor Questions” times 0945 and 1348.**

From post exam review comments, the students considered that RCIC did not need to be isolated, instead closing the suction from the CST (BJ-HV-F010) had potential to stop the leak. However, this was not an option nor did the proctor clarify the “Isolate RCIC” choice when asked.

Additional student comments stated the RCIC Isolation taken in the literal sense would not stop the leak, because the CST suction source is not included. All references to **“Isolation”** in system operating procedures do not include closure of the CST suction valve.

Excerpts from HC.OP-SO.BD-0001 rev 23 (**below**) reflect the procedure context of "**Isolation.**"

3.3.4 AUTO isolation occurs upon receipt of any of the following signals:

- Low Reactor pressure (< 64.5 psig w/4 second time delay).
- High Steam Pipe Area temperature
  (≥160°F w/30 min time delay).
- High steam line flow :
  ≥ 598" H_2O w/4 sec. time delay
  - 50" H_2O w/4 sec. time delay.
- High Turbine Exhaust Diaphragm pressure (> 10 psig).
- RCIC Pump Room High temperature
  (≥160°F w/1 sec time delay).
- RCIC Pump Room Ventilation Duct High diff
  temperature (≥ 70°F w/1 sec time delay).
- RCIC Torus Compartment High temperature
  (≥ 128°F w/30 min time delay).

3.3.5 Isolation Logic Train "B" when actuated, closes the following valves:

- FC-HV-F008 RCIC STM OUTBD ISLN VLV
- Turbine trip throttle valve.

3.3.6 Isolation Logic Train "D" when actuated, closes the following valves:

A. FC-HV-F007 RCIC STM INBD ISLN VLV

B. FC-HV-F076 RCIC STM LN WARMUP VLV

C. Turbine trip throttle valve

3.3.7 Exhaust line vacuum breaker isolation valve closure interlock FC-HV-F084 and FC-HV-F062 will close upon receipt of the following:

- Low steam supply line pressure 64.5 psig

AND

- 1.68 psig in the Drywell.

3.3.8 The mechanical overspeed trip of 125% rated speed will close the trip throttle valve and must be reset locally, and the limitorque must be manually run to the full closed position to relatch the valve.

3.3.9 Valve BD-HV-F031, PMP SUCT FROM SUPP CHB Auto Opens on **CST low level** (E51-K69) provided CST level signal is not in manual override.
WHEN BD-HV-F031 reaches full open, valve BD-HV-F010, PMP SUCT FROM CST will Auto Close.

Recommendation: Change answer key to both "b" and "c" as correct answers.

References: HC.OP-EO.ZZ-0103/4, HC.OP-SO.BD-0001

Record – 52
SRO Question – 47

Given the following:

- Reactor power is 83%.
- Neither RBM is bypassed with the joystick.
- Rod 30-31 has just been selected.

Use the attached figure of the 4-Rod Display for LPRM indications
(Ribbon readings are approximates)

Assuming all other LPRMs are operable, which of the following describes the operability status of the RBM CHANNEL A and CHANNEL B?

a. A- Operable; B- Operable

b. A- Operable; B- Inoperable

c. A- Inoperable; B- Operable

d. A- Inoperable; B- Inoperable

Answer: b

The question required determining the operability status of the RBMs based on the operable LPRM inputs. Candidates were provided with the 4-Rod Display surrounding the selected rod, and no additional information or handouts. The answer was based on the Administrative requirement of SH.OP-AP.ZZ-0108, Exhibit 3

The question asked the operability status. Two issues exists when referring to the operability status.

1. The RBM will automatically generate an inoperable trip if more than 50% of the available LPRMs are bypassed. Based on the figure provide, the candidates answered the question to determine if the RBM was operable or inoperable based on the automatic INOP trip, they choose answer "a" which states "**both "A" and "B" RBM are operable."**
2. SH.OP-AP.ZZ-0108, Operability Assessments, also includes requirements for determining if a RBM is operable based on the available LPRMs per level. **The candidates were not provided SH.OP-AP.ZZ-0108 to make the determination if instruments were operable per SH.OP-AP.ZZ-0108 requirements**. (**See excerpts below**)

SH.OP-AP.ZZ-0108 rev 11, page 81 of 104 (**Exhibit 3 page 15 of 18**)

- Licensee Controlled AOT - RBM
- Description /Regulatory Basis - Rod Block Monitor is inoperable due to inadequate LPRM inputs.
- Required Action - WHEN LPRMs are bypassed, OR become INOP, EVALUATE the impact on the Rod Block Monitor function. ENSURE that at least 50% of the LPRM detectors are operable for all 4 detector levels for all control rods affected by that LPRM.
    - o IF at least 50% of the operable detectors per level are NOT available**, rod withdrawal is prohibited** unless a condition specific evaluation is performed or the condition is corrected.
    - o NOTIFY Nuclear Fuels and Reactor Engineering IF this condition is not satisfied. [**70005801**]

As shown, **SH.OP-AP.ZZ-0108 does not make the Rod Block Monitor Channel Inoperable. I**t only prevents rod motion for that rod selected until additional evaluations are completed.

Selected answer key choice “b” - “A” RBM is Operable and “B” RBM is Inoperable is **incorrect .**

Recommendation: Change answer key to “a” as the correct answer

References SH.OP-AP.ZZ-0108 , exhibit 3
HC.OP-SO-SF-0002, note 5.1.7

Record – 69
RO Question - 55
SRO Question – 57

Given the following conditions:

- The plant has been manually scrammed.
- A normal reactor cooldown is in progress.
- The reference leg backfill system is out of service.

Then, annunciator (A7-C5) “RPV LEVEL 4” is received. The operator investigates and observes that reactor water level “notching” is occurring.

Which of the following is the most accurate indicated water level from the indicator that is experiencing “notching”?

a. An average of the water levels from the top AND bottom of the “notch”.

b. The water level at the bottom of the “notch”.

c. The water level at the top of the “notch”.

d. An average of the water levels from all indicators that are “notching”.

Answer: c

Answer key was found incorrect due to typographical error. **The correct answer is “b” not “c”.**

Recommendation: Change answer key to “b” as the correct answer.

Record - 97
SRO Question – 78

Given the following conditions:

- The reactor core has been operating with one or more known fuel pin leaks.
- A reactor scram occurred from 100 percent power.
- Both Scram Discharge Volume Drain Valves did NOT go full closed.

Which one of the following rooms would become the most significant radiological hazard?

a. Reactor Building North Equipment Sump Room.

b. HPCI Pump and Turbine Room.

c. Reactor Building South Equipment Sump Room.

d. RCIC Pump and Turbine Room.

Answer: c

The question required the candidates to identify, from memory, where the scram discharge volumes (SDVs) drain. The students recognized that the SDVs drain to the reactor building equipment drain sump but were unable to determine which sump without addition references.

The training the candidates received from the Control Rod Hydraulics (CRD) lesson plan (NOH01CRDHYD) only states that the SDVs drain to clean radwaste (CRW).

M-47-1, Control Rod Drive identifies that the SDVs drain to (CRW). **This reference was provided to the candidates.**

Determination of the specific equipment drain sump that the SDVs drains to requires use of M-61 Sheet 2. **M-61 was not provided to the student**.

Without the additional reference the candidates could not select between the “North” or “South” Equipment Sump Room(s)

Recommendation: Change answer key to both “a” and “c” as correct answers.

References: P&ID M-47 and M-61.
NOH01CRDHYD

Record – 106
RO Question – 87
SRO Question – 83

Given the following conditions:

Station Service Water (SSW) pump status:
- 'A' SSW pump I/S in AUTO.
- 'B' SSW pump I/S in AUTO.
- 'C' SSW pump O/S in AUTO.
- 'D' SSW pump O/S in AUTO.

Which one of the following will result in the automatic start of the 'D' SSW Pump?

a. 'A' SSW Loop low flow.

b. 'A' SSW Pump low flow.

c. 'B' SSW Loop low flow.

d. 'B' SSW Pump low flow.

Answer: d

The question asked what will cause the automatic start of the "D" Service Water pump that is in AUTO. IAW HC.OP-SO.EA-0001, Service Water System Operation, (**see below**)

3.3.1 When in AUTO with NO Process Start Inhibit signal, the Station Service Water Pumps auto start on any of the following signals (B AND D Pumps will NOT auto start if control is transferred to the Remote Shutdown Panel):

- Associated loop low flow < 13,475 gpm (< 1.0 psid across pump strainer)
- Reactor Water low level (< -38 inches)
- Drywell high pressure (> 1.68 psig)
- Reactor Building high Radiation (> 1X10-3 uCi/cc)
- Refueling Floor high Radiation (> 2X10-3 uCi/cc)
- Containment Manual Initiation

The choices provided that will auto start a service water pump is anything that will produce "Associated loop low flow < 13, 475 gpm (< 1.0 psid across pump strainer)." Based on this procedural step either:

Answer "c" –"B" SSW Loop low Flow matches the procedure statement "Associated loop low flow < 13, 475 gpm"

Answer "d" – "B" SSW Pump low flow matches the procedure statement "< 1.0 psid across pump strainer".

This makes both answers correct

Recommendation: Change answer key to both "c" and "d" as correct answers.
References: HC.OP-SO.EA-0001

Record – 118
RO Question - 95
SRO Question -92

Per NC.NA-AP.ZZ-0024, Radiation Protection Program, a 21 year old worker with 11 Rem Lifetime dose from the previous 3 years working at Hope Creek will have an administrative exposure control level of (1)________ mrem TEDE per year. This can be raised to a maximum of (2)_______ mrem TEDE by the Radiation Protection Manager.
(Assume NO delegation of authority)

a. (1) 2000
(2) 3000

b. (1) 2000
(2) 4000

c. (1) 3000
(2) 4500

d. (1) 3000
(2) 4750

Answer: b

The answer key is incorrect.

Attachment 1 of NC.NA-AP.ZZ-0024 provides limitations if lifetime administrative limits of 2(N-17) are exceeded. The question was taken from the INPO exam bank and the justification was inadvertently verified based on an Administrative limit of 5(N-17) due to an oversight.

The worker in the question exceeded the administrative lifetime 2(N-17) limit, therefore the table on page 2 of Attachment 1 to NC.NA-AP.ZZ-0024 must be used.

The worker's administrative exposure control limit is 2000 mrem/year, which can be raised to 3000 mrem/year by the Radiation Protection Manager. (**see table**)

2(N-17) LIFETIME DOSE ACTION LEVEL (Administrative Control Level)			
Dose Control Level	**Description**	**Action at Control Level**	**Increase Approval**
2000 mrem/year TEDE	Initial administrative control level for workers whose lifetime dose exceeds 2(N-17) where n is the age in years	Administrative control level may be increased to 3000 mrem/year	Radiation Protection Manager
3000 mrem/year TEDE	Extended administrative control level for workers whose lifetime dose exceeds 2(N-17) where n is the age in years	Administrative control level may be increased to 4000 mrem/year	Vice President - Operations
4000 mrem/year TEDE	Final administrative control level for workers whose lifetime dose exceeds 2(N-17) where n is the age in years (may not be exceeded in non-emergency situations).	Incremental increase until 4750 mrem control level is reached	Vice President - Operations

Recommendation: Change answer key to “a” as correct answer.

Reference: NC.NA-AP.ZZ-0024 rev 12 Attachment #1

Record 3
RO Question 2
SRO Question 3

Given the following conditions:

- The plant is operating at 100 percent power.
- TACS is on the ’A’ Loop of SACS.
- ’A’ 1E 4.16 KV bus 10A401 has de-energized due to bus fault.

Which one of the following describes a result of the bus fault and the reason for the result?

a. All RACS Pumps trip due to LO-LO Head tank level.

b. ’B’ SACS Expansion Tank overflows due to power loss to TACS return valves.

c. RACS Head Tank overflows due to makeup valve power loss.

d. ’A’ & ’C’ SACS Pump trip due to LO-LO-LO Expansion Tank level.

Answer: c

The question stem indicates a loss of the 10A401 bus. This loss would cause an operator to enter HC.OP-AB.ZZ-0170 “Loss of 4.16kv Bus - 10A401 A Channel”.

This procedure contains Nine (9) attachments written to identify the loads lost off each level of power supply from; the entire bus/unit substation, individual Motor Control Centers and supplied lighting panels.

Each system affected by the power loss will be addressed by entry into another Abnormal procedure, as shown in the Subsequent Actions section **(see table)**

4.0 SUBSEQUENT OPERATOR ACTIONS

4.1 On a loss of the 10A401 Bus the following major equipment will be lost AND the appropriate Abnormals should be entered concurrently. SH.OP-AP.ZZ-0108(Q) should also be referred to

COMPONENTS	ABNORMAL
Loss of A SACS Pump (IF running) WITH no auto start of C SACS Pump	HC.OP-AB.COOL-0002(Q)
Loss of TACS IF supplied by A SACS loop	HC.OP-AB.COOL-0002(Q)
Loss of A SSW Pump	HC.OP-AB.COOL-0001(Q)
Loss of 50% of the Drywell Cooling Fans causing drywell pressure to increase	HC.OP-AB.CONT-0001(Q)
Loss of HPCI Jockey Pump	
Loss of PCIG (IF a loss of A SACS loop occurs) due to lockout on A PCIG Compressor	HC.OP-AB.COMP-0002(Q)
Loss of FPCC	HC.OP-AB.COOL-0004(Q)
Loss of Drywell Equipment and Floor Drain Sump Pumps	
Loss of A TSC Chiller	HC.OP-AB.HVAC-0001(Q)
Loss of A RACS Pump and RACS Panel 10C202 - Make-up valve fails open	HC.OP-AB.COOL-0003(Q)
Loss of power to Instrument Air Control Panels- Dryers should fail in service but will not regenerate	HC.OP-AB.COMP-0001(Q)
All AC supplies to 1A-D-481, 1A-D-482, 1A-D-483, 10-D-410, 10-D-470 and 10-D-450. Loads will be carried by the batteries (Batteries have a four hour design capacity).	HC.OP-AB.ZZ-0135(Q) HC.OP-AB.ZZ-0136(Q) HC.OP-AB.ZZ-0150(Q)

The RACS make-up valve is supplied from a lighting panel (10-Y-205) and is addressed on Attachment 8 of this procedure. (**see below**)

ATTACHMENT 8
Page 1 of 1

10B252 MCC

1A1-1H1V212 Drywell Cooling Fans	HC.OP-AB.CONT-0001(Q)
10-E-276 SLC Tank Operating Heater ! SLC will become inoperable after a period of time due to low tank temperature. CYCLE Mixing Heater to maintain temperature.	
1C-D-483, Inverter Backup power	HC.OP-AB.ZZ-0136(Q)
1A-D-483 Inverter Normal supply	HC.OP-AB.ZZ-0136(Q)
1A-D-484 Inverter Backup supply	HC.OP-AB.ZZ-0136(Q)
1A-K-402 A EDG Air Compressor ! CROSS CONNECT air receivers to another EDG Air System.	
1C-P-267 Drywell Floor Drain Sump Pump ! Will receive Sump High Level alarm on alternating pump starts.	
1A-P-267 Drywell Equipment Drain Sump Pump ! Will receive Sump High Level alarm on alternating pump starts.	
1B1D-474 Battery Charger	HC.OP-AB.ZZ-0150(Q)
1A2-D-473 Battery Charger ! MONITOR voltage on 10D470. ENSURE 1A1-D-473 is in service.	HC.OP-AB.ZZ-0150(Q)
10-Y-205 Reactor Bldg 120VAC Panel ! 1A-C-201 SACS Control Panel UPS – Non-1E UPS is rated for approximately one hour on the battery. ! 10-C-202 RACS Control Panel - The RACS Exp Tank M/U valve and FCV will fail open when the panel is de-energized. The RACS Pumps will trip when the panel is restored.	HC.OP-AB.COOL-0002(Q) HC.OP-SO.NQ-0002(Q) HC.OP-AB.COOL-0003(Q)

The procedure addresses that the valve fails open, resulting in overflow of the head tank into the reactor building floor drain system. The procedural direction is based on re-energization of the lighting panel and the loss of the operating RACS pumps.

In the scheme of what an operator would be focusing their attention on with a complete loss of the bus, this would be one of the minor problems and addressed long after other systems such as SACS, TACS, DW Cooling, HPCI keep-fill, Primary Containment Instrument Gas, Fuel Pool Cooling, Technical Support Center Chilled Water have been addressed.

For determination of how a Non-1E component fails on a loss of power, the candidates should be supplied with the applicable print, (M-13) or the procedure HC.OP-AB.ZZ-0170.

Neither procedure HC.OP-AB.ZZ-0170, nor the M-13 print were supplied to the candidates.

Recommendation: Delete question.

References: M-13
HC.OP-AB.ZZ-0170

Record-114
RO Question-92
SRO Question-90

The core has been off-loaded to the fuel pool. Per HC.RE-AP.ZZ-0049, Hope Creek Conduct of Fuel Handling, what is the MINIMUM permissible complement of personnel in the crew involved in fuel movement NOT involving core alterations?

a. Fuel Handling Operator
Radiation Protection Technician
Reactor Engineer, acting as spotter

b. Fuel Handling Operator
Refueling Bridge Operator as spotter
Radiation Protection Technician
Reactor Engineer

c. Fuel Handling Operator
Refueling Bridge Operator
SRO acting as spotter
Radiation Protection Technician

d. Fuel Handling Operator
Refueling Bridge Operator
Radiation Protection Technician
Reactor Engineer
Control Room Refuel Monitor

Answer: a

Question asked what was the minimum permissible complement of personnel for moving fuel NOT involving core alterations.

Choice “a” was written based on HC.RE-AP.ZZ-0049 section 5.3 alone, however more restrictive requirements exist in HC.OP-SO.KE-0001, Refueling Platform and Fuel Grapple Operation. The use of HC.OP-SO.KE-0001 is directed by section 5.3.2.C.1.

Note 5.8 of HC.OP-SO.KE-0001 requires all irradiated fuel moves to be supervised by an SRO or SRO limited to fuel handling. **(see below)**

NOTE 5.8

All irradiated fuel moves or core alterations must be directly supervised by a licensed SRO or SRO limited to fuel-handling. Non-irradiated fuel handling not involving core alterations and blade guide movement do not require direct supervision by an SRO and can be annotated by the spotter directly involved with the evolution. **[CD-168A]**

> This procedure Section describes any combination of fuel/blade guide in Spent Fuel Storage Pool movements between any of the following fuel storage locations:

Non-irradiated fuel is brought onto the site and loaded into the pool prior to commencing a Refueling outage.

The question specifically stated the entire core was offloaded, which means any moves inside the fuel pool would have to be assumed to be irradiated fuel.

The question stem did not specify non-irradiated fuel was being handled.

Based on lack of information in the question, there is no correct answer.

Recommendation: Delete the question

References: HC.RE-AP.ZZ-0049,
HC.OP-SO.KE-0001

ATTACHMENT 3

NRC RESOLUTION OF LICENSEE COMMENTS

RO 18/SRO 20

Refer to Attachment 2 for details of the question and basis for facility recommendation.

Comment accepted. The question stem does not provide enough information regarding suppression pool (SP) parameters, particularly SP temperature. Given an ATWS condition from 100% power, with subsequent MSIV closure, it is credible that "suppression pool temperature is rising towards HCTL." Consequently, Boron injection is required, and, when combined with the other conditions given in the stem, MSIVs may be re-opened. Therefore, distractor B is also a valid choice.

SRO 31

Refer to Attachment 2 for details of the question and basis for facility recommendation.

Comment not accepted. The facility correctly noted that applicants could not correlate the calculated containment level to suppression pool (SP) level without references, and/or applicants are not expected to know that containment level is 94" greater than SP level indication. However, the facility is not correct in asserting that SP level is *indeterminate*. Level could be calculated given the needed reference or memory recall of the 94" conversion value. Thus, the only *correct* answer to the question is still choice "B", the original answer. That is, insufficient information given to the applicants is not an acceptable basis to instead accept an answer that is technically and procedurally incorrect. Therefore, the proper resolution is question deletion.

RO 32/SRO 38

Refer to Attachment 2 for details of the question and basis for facility recommendation.

Comment not accepted. The facility noted that isolation of RCIC by procedure would NOT stop the condensate storage tank (CST) leak and provided an excerpt from the RCIC operating procedure that showed isolating RCIC did not isolate the RCIC suction valve to the CST. Furthermore, given (as the facility also noted) that lowering CST level indicated the leak was from the CST, isolation of RCIC is not required *per the EOP*, which directs the operator to "ISOLATE all systems discharging into the area" (i.e., since the CST is discharging, isolate that system and not RCIC). Therefore, the correct answer is changed to reflect that the ONLY answer is "B".

SRO 47

Refer to Attachment 2 for details of the question and basis for facility recommendation.

Comment accepted. The original answer was wrong. That answer was based on *administrative* requirements imposed on the RBM, per an *administrative* procedure, SH.OP-AP.ZZ-0108. However, the question asked for *operability* status of the component, not what administrative limitations were imposed. Thus, the Technical Specification requirements for the RBM needed to be applied, and, for the conditions given in the question, were met by choice "A". Therefore, the correct answer is changed to "A".

RO 55/SRO 57

Refer to Attachment 2 for details of the question and basis for facility recommendation.

Comment accepted. Correct the typographical error; the answer should be “B”.

SRO 78

Refer to Attachment 2 for details of the question and basis for facility recommendation.

Comment not accepted. The facility indicated on the initial draft that choice “A” is a common misconception among students. Also, facility-provided information showed the south equipment pump room is the only place where the SDVs drain. Since distractor “A” is technically incorrect it cannot be an acceptable, alternate choice. The only correct answer is still “C”.

RO 87/SRO 83

Refer to Attachment 2 for details of the question and basis for facility recommendation.

Comment accepted. The facility provided information that showed either loop low flow or pump low flow will result in the automatic start of the “D” service water pump. Therefore, distractor “C” is also correct.

RO 95/SRO 92

Refer to Attachment 2 for details of the question and basis for facility recommendation.

Comment accepted. The original answer key did not incorporate site-specific details. The correct answer is “A”.

RO 2/SRO 3

Refer to Attachment 2 for details of the question and basis for facility recommendation.

Comment accepted. The question requires the recall of knowledge that is too specific for the closed reference test mode, i.e., this knowledge is not required to be known from memory. No facility learning objectives exist that requires *memorization* of power supply or failure mode for this level of minor, non-safety related component. Learning objectives do exist, however, they are prefaced with statements such as, “Given a (drawing/diagram/control room reference) ... .” No references were permitted for this question. Therefore, this question is deleted.

RO 92/SRO 90

Refer to Attachment 2 for details of the question and basis for facility recommendation.

Comment accepted. The question stem is confusing. It indicated the core had been off-loaded. In that context, fuel movement inside the fuel pool would essentially always involve moving irradiated fuel. Although a crew could just move non-irradiated (i.e., new) fuel within the fuel pool - and thus satisfy the conditions of the proposed answer - facility staff indicated such moves would be extremely unlikely and essentially have never been nor would ever be performed. The much more likely scenario - and one which meets the conditions listed in the answer - relates to moving new fuel from the fuel inspection stand to the fuel pool. However, that evolution was not the context for the question. Therefore, given the confusing focus of the stem, this question is deleted.