2015 PALISADES NUCLEAR PLANT

INITIAL LICENSE RETAKE EXAMINATION

PROPOSED EXAM FILES

Entergy

Entergy Nuclear Operations, Inc.
Palisades Nuclear Plant
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PNP 2015-001

January 13, 2015

NUREG-1021

Mr. Michael Bielby
U.S. Nuclear Regulatory Commission
Region III
2443 Warrenville Road
Suite 210
Lisle, IL 60532-4352

Subject: Final Operator License Re-Examination Materials and Final Re-Application

Palisades Nuclear Plant
Dockets 50-255 and 72-07
License No. DPR-20

Dear Mr. Bielby:

Entergy Nuclear Operations, Inc. (ENO) is submitting the final license re-examination materials for the Palisades Nuclear Plant (PNP), in accordance with NUREG-1021, "Operator Licensing Examination Standards for Power Reactors," Revision 9, Supplement 1, for Kristopher A. Ruetz, docket number 55-33801. Additionally, ENO is attaching the final Senior Reactor Operator and Reactor Operator Personal Qualification Statement – Licensee (NRC Form 398) for Kristopher A. Ruetz, docket number 55-33801.

Mr. Ruetz is undergoing remediation, which will be completed prior to taking the examination. An updated ENO remedial training plan is attached.

The initial license examination is scheduled for the week of January 19, 2015. The following materials are enclosed:

- ES 201-3, Examination Security Agreement (updated)
- ES 201-4, List of Applicants
- ES 401-2/3, PWR Examination Outline (updated)

JAN 14 2015

- ES 401-4, Record of Rejected K/As (updated) with K/A Selection Methodology and Suppression Report

- ES-401-6, Written Examination Quality Checklist (Original)

In accordance with NUREG-1021, ES-201, Attachment 3, enclosed materials shall be withheld from public disclosure until after the examinations are complete.

Attachment 1 contains personally identifiable information. Therefore, it is requested that Attachment 1 be withheld from public disclosure in accordance with 10 CFR 2.390.

Please contact Steve Botimer at (269) 764-2975 if you have any questions regarding this submittal.

This letter contains no new commitments and no revisions to existing commitments.

Sincerely,

Robert ShWinter
for WENelson

wen/jpm

Attachment 1: Final NRC Form 398
Attachment 2: Updated ENO Remedial Training Plan (TQF-201-IM05)
Attachment 3: Enclosed Final Operator License Re-Examination Materials

CC Administrator, Region III, USNRC (w/o attachments and enclosures)
Project Manager, Palisades, USNRC (w/o attachments and enclosures)
Resident Inspector, Palisades, USNRC (w/o attachments and enclosures)
Document Control Desk, USNRC (w/o attachments and enclosures)

Facility: Palisades | Date of Exam: January 2015 | Exam Level: RO ☐ SRO ☒

	Item Description				Initial a	Initial b*	Initial c#
1.	Questions and answers are technically accurate and applicable to the facility.				sb	BAB	MEB
2.	a. NRC K/As are referenced for all questions. b. Facility learning objectives are referenced as available.				sb	BAB	MEB
3.	SRO questions are appropriate in accordance with Section D.2.d of ES-401				sb	BAB	MEB
4.	The sampling process was random and systematic (If more than 4 RO or 2 SRO questions were repeated from the last 2 NRC licensing exam, consult the NRR OL program office).				sb	BAB	MEB
5.	Question duplication from the license screening/audit exam was controlled as indicated below (check the item that applies) and appears appropriate: __ the audit exam was systematically and randomly developed; or __ the audit exam was completed before the license exam was started; or __ the examinations were developed independently; or _x_ the licensee certifies that there is no duplication; or __ other (explain)				sb	BAB	MEB
6.	Bank use meets limits (no more than 75 percent from the bank, at least 10 percent new, and the rest new or modified); enter the actual RO / SRO-only question distribution(s) at right.	Bank: NA / ~~18~~ 14 MEB	Modified: NA / ~~8~~ 1 MEB	New: NA / ~~7~~ 10 MEB	sb	BAB	MEB
7.	Between 50 and 60 percent of the questions on the RO exam are written at the comprehension /analysis level; the SRO exam may exceed 60 percent if the randomly selected KAs support the higher cognitive levels; enter the actual RO / SRO question distribution(s) at right.	Memory: NA / 3	C/A: NA / 22		sb	BAB	MEB
8.	References/handouts provided do not give away answers or aid in the elimination of distractors.				sb	BAB	MEB
9.	Question content conforms with specific K/A statements in the previously approved examination outline and is appropriate for the Tier to which they are assigned; deviations are justified				sb	BAB	MEB
10.	Question psychometric quality and format meet the guidelines in ES Appendix B.				sb	BAB	MEB
11.	The exam contains the required number of one-point, multiple choice items; the total is correct and agrees with value on cover sheet				sb	BAB	MEB

	Printed Name / Signature	Date
a. Author	Steve Botimer / SBotimer	01/08/15
b. Facility Reviewer (*)	Brett Baker / BABaker	01/8/15
c. NRC Chief Examiner (#)	Michael Bielby / Michael E Bielby Sr.	1/8/15
d. NRC Regional Supervisor	Hironori Peterson / [signature]	1/20/15

Note: * The facility reviewer's initials/signature are not applicable for NRC-developed examinations.
Independent NRC reviewer initial items in Column "c"; chief examiner concurrence required.

dm for HP 01/08/2015

WRITTEN QUESTION DATA SHEET

Source of Question: **NEW**
K/A: 008 Pressurizer (PZR) Vapor Space Accident (Relief Valve Stuck Open)
AA2.01 Ability to determine and interpret the following as they apply to a Pressurizer Vapor Space Accident: RCS pressure and temperature indicators and alarms.

Tier: 1 Group: 1 SRO Imp: 4.2
Applicable 10CFR55 Section: 43.5, Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations. ***This exam question meets the criteria for an SRO-only question because the operator must determine the correct procedure to enter and what action should be taken.***

Palisades Learning Objective: TBCORE_CP04.0, Given plant conditions involving Emergency Operating Procedures, determine correct follow-up EOP utilizing Event Diagnostic Flow Chart in accordance with in use EOP.

References: EOP-4.0 Basis step 10f

Question:

Given the following:

A Reactor trip occurs due to a turbine trip at full power

- During the transient Annunciator EK-0744, PRESSURIZER SAFETY VALVE RV-1039 DISCH HI TEMP alarm comes in.
- RV-1039 downstream tailpipe temperature is 240°F.
- Pressurizer (PZR) pressure is 2000 psia and lowering.
- PZR level is 48% and rising.
- PCPs are running.
- T_{AVE} is 535°F and stable.

Following the actions of EOP-1.0, "Standard Post Trip Actions," (1) to which procedure would you transition and (2) what mitigating strategy would be used?

a. (1) Go to EOP-2.0, Reactor Trip Recovery.
   (2) Use AOP-28, Pressurizer Pressure Control Malfunctions, to correct the PPCS malfunction.
b. (1) Go to EOP-2.0, Reactor Trip Recovery.
   (2) Use AOP-23, Primary Coolant Leak, to control the PCS leak.
c. (1) Go to EOP-4.0, Loss of Coolant Accident Recovery.
   (2) Lower pressure to less than 1605 psia to actuate SIAS.
d. (1) Go to EOP-4.0, Loss of Coolant Accident Recovery.
   (2) Lower pressure to 1700 - 1800 psia to try to reseat the valve.

DISTRACTOR ANALYSIS

a. INCORRECT Plausible because the PCS pressure value given in the stem is within the band to go to EOP-2.0 but because its trend is going lower EOP-4.0 is required.
b. INCORRECT Plausible same as 'a' above
c. INCORRECT Plausible because this is the correct procedure but there is no benefit (or direction) to actuate SIAS if it is avoidable
d. **CORRECT This action is directed in EOP-4.0 at step 10f if there are indications of a PZR relief valve being lifted.**

Level of Knowledge: ANALYSIS **Difficulty: 2**

WRITTEN QUESTION DATA SHEET

Source of Question: MODIFIED BANK
K/A: 00009 Small Break LOCA
G.2.4.41 Knowledge of emergency action level thresholds and classifications.

Tier: 1 Group: 1 SRO Imp: 4.6
Applicable 10CFR55 Section: 43.5 - Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations. ***This exam question meets the criteria for an SRO only question because it is an SRO only duty to classify emergency events during implementation of the emergency plan. (SRO task # PL-344 019 05 03).***

Palisades Learning Objective: PL-N00113_E01.09 ***Given EI-1, EAL Basis and SEP Supplement 1, along with plant emergency condition(s), classify the emergency, per given procedures.***

References: EI-1 Attachment 1, EAL Wall Charts (PROVIDE)

Question:

Given the following:

- Reactor manually tripped due to Primary Coolant System (PCS) unidentified leakage > 10.0 gpm.
- PCS leak rate is 16 gpm.
- RIA-2326, Stack Normal Range monitor, is reading 350 cpm.
- 'B' S/G pressure is 750 psia and lowering rapidly.
- Containment pressure is 4.5 psig and rising.
- Containment Spray did not automatically actuate.
- The NCO manually opened CV-3001 and 3002 and started all Containment Spray pumps.

Based on the above conditions, how would you classify this event?

a. No Classification

b. Unusual Event SU8.1

c. Alert FA1.1

d. Site Area Emergency FS1.1

DISTRACTOR ANALYSIS

a. INCORRECT Plausible because the operator may confuse the identified leak classification threshold of 25 gpm with the unidentified threshold.
b. **CORRECT this meets the threshold of > 10 gpm unidentified leakage.**
c. INCORRECT Plausible because there is a PCS leak for which the reactor is tripped but it is below the threshold of 50 gpm and the PCS leak did not cause the ECCS actuation so the PCS barrier is intact.
d. INCORRECT Plausible because the operator may see the leak and believe erroneously that the PCS barrier is lost and that Containment is also potentially lost due to the steam break since Containment Spray did not auto actuate.

Level of Knowledge: APPLICATION **Difficulty: 3**

WRITTEN QUESTION DATA SHEET

Source of Question: NEW

K/A: 000011 Large Break LOCA AA2.06 Ability to determine and interpret the following as they apply to a Large Break LOCA: Actions to be taken, based on RCS temperature and pressure – saturated and superheated

Tier: 1 Group: 1 SRO Imp: 4.7

Applicable 10CFR55 Section: 43.5, Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations. ***This exam question meets the criteria for an SRO-only question because the operator must determine which are the correct automatic actions and select the procedure that gives the needed guidance.***

Palisades Learning Objective: TBCORE_CP03.0, ***Given plant conditions involving Emergency Operating Procedures, determine Primary Coolant System (PCS) subcooling using all available methods in accordance with in use EOP.***

References: EOP-4.0 (PROVIDE EOP 4.0 SFSCs)

Question:

The Plant is at full power with P-67B, Low Pressure Safety Injection (LPSI) Pump, out of service for motor replacement.

Then, the following occurs:

- A Reactor trip occurs due to a large break loss of coolant accident
- Off-site power is lost (LOOP)
- D/G 1-2 trips
- EOP-4.0, "Loss of Coolant Accident Recovery," has been implemented.
- PZR pressure is 70 psia
- All RVLMS lights are red
- Containment water level is 589.5'
- Average of qualified CETs is 302°F
- Both S/Gs are 10% and rising with P-8A Flow Control valves in cascade feeding at 165 gpm per S/G
- Both S/G pressures are 300 psia and lowering.

Based on this information, (1) what is the status of Safety Functions and (2), what should be done regarding procedure use?

a. (1) All Safety Functions are satisfied.
   (2) Stay in EOP-4, Loss of Coolant Accident Recovery.

b. (1) PCS Inventory Control is NOT satisfied.
   (2) Go to EOP-9, Functional Recovery Procedure.

c. (1) Core Heat Removal is NOT satisfied.
   (2) Go to EOP-9, Functional Recovery Procedure.

d. (1) PCS Heat Removal is NOT satisfied.
   (2) Go to EOP-9, Functional Recovery Procedure.

DISTRACTOR ANALYSIS

a. INCORRECT Plausible because the operator could easily miss the one indicator in the stem that would render Inventory Control being satisfied.
b. **CORRECT PCS Inventory is below the threshold because all the RVLMS red lights are on indicating that level is NOT 11 inches above the bottom of the fuel alignment plate.**
c. INCORRECT Plausible because the ECCS flow is low in conjunction with the low RVLMS level but CETs are less than superheated.
d. INCORRECT Plausible because S/G level is below the desired range but it is being restored.

Level of Knowledge: APPLICATION **Difficulty: 2**

WRITTEN QUESTION DATA SHEET

Source of Question: USED ON 2008 PALISADES EXAM.
K/A: 000025 Loss of RHR System
G2.4.11 - Knowledge of abnormal condition procedures.

Tier: 1 Group: 1 SRO Imp: 4.2
Applicable 10CFR55 Section: 43.5 – Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations. ***This exam question meets the criteria for an SRO-only question because it requires assessment of plant conditions (normal, abnormal, or emergency) and then selecting a procedure or section of a procedure to mitigate, recover, or with which to proceed. Interpretation of Safety Function Status Checks is an SRO Only function.***

Palisades Learning Objective: IOTF_CK15.0. ***Given an Abnormal Operating event and control room reference, describe the effect of the Abnormal Operating condition on affected plant systems and components without error.***

References: AOP-30 attachment 4

Question:

Given the following conditions during a Loss of Shutdown Cooling event in MODE 5:

- PCS Level is 620' 4"
- Average of Qualified CETs indicates 192°F and rising
- Containment air temperature is 110°F
- EY-40, Preferred AC Bus, is de-energized due to failure of Inverter #4

Based on the above conditions, which one of the following safety functions does NOT meet the acceptance criteria of AOP-30, Loss of Shutdown Cooling, Safety Function Status Checks?

a. Maintenance of Vital Auxiliaries – Electric

b. Containment Atmosphere

c. PCS/Core Heat Removal

d. PCS Inventory Control

DISTRACTOR ANALYSIS

a. INCORRECT - Plausible however only 3 of 4 preferred AC buses need to be energized to meet MVAE.
b. INCORRECT – Plausible however as long as Containment temperature is <125°F CA is met.
c. **CORRECT – Acceptance criteria for PCS/Core heat Removal is: Qualified CETs less than 200°F and not rising**
d. INCORRECT – Plausible as this is the reduced inventory PCS level, however, as long as level is > 618' 2.5" IC is met.

Level of Knowledge: ANALYSIS **Difficulty: 4**

WRITTEN QUESTION DATA SHEET

Source of Question: BANK
K/A: CE/E05 Excess Steam Demand EA2.1 Ability to determine and interpret the following as they apply to the (Excess Steam Demand) Facility conditions and selection of appropriate procedures during abnormal and emergency operations.
Tier: 1 Group: 1 SRO Imp: 4.0
Applicable 10CFR55 Section: 43.5 – Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations. ***This exam question meets the criteria for an SRO-only question because it requires knowledge of when to implement attachments and appendices, including how to coordinate these items with procedure steps.***

Palisades Learning Objective: TBAD_E03.01) ***Given indications of an Excess Steam Demand Event, determine which Steam Generator is the faulted generator in accordance with EOP 6.0.***

References: EOP-1.0, Event Diagnostic Flow Chart; EOP-6.0 basis step 14 and 16

Question:

Given the following conditions:

- Reactor tripped due to Containment High Pressure (CHP) signal
- PCS Pressure is 1190 psia
- PZR Level is 12% and lowering
- 'A' S/G Pressure is 850 psia, 'B' S/G Pressure is 520 psia
- T_C in both loops is lowering
- PCS subcooling is 95°F
- EOP-1.0 "Standard Post Trip Actions" have been completed

Which one of the following describes (1) the appropriate procedure to enter, and (2) the actions that need to be taken?

a. (1) EOP-6.0, "Excess Steam Demand Event"
(2) Isolate the 'B' Steam Generator per EOP supplement 18, 'B' S/G ESDE Isolation Checklist, AND maintain 'A' S/G pressure within 50 psi of 'B' S/G as soon as possible.

b. (1) EOP-6.0, "Excess Steam Demand Event"
(2) Isolate the 'B' Steam Generator per EOP supplement 18, 'B' S/G ESDE Isolation Checklist, AND maintain 'A' S/G pressure within 50 psi of 'B' S/G after 'B' S/G indicates it has boiled dry.

c. (1) EOP-9.0, "Functional Recovery Procedure"
(2) Isolate the 'B' Steam Generator per EOP supplement 18, 'B' S/G ESDE Isolation Checklist when level reaches -84% AND stabilize PCS temperature with 'A' Steam Generator.

d. (1) EOP-9.0, "Functional Recovery Procedure"
(2) Isolate the 'B' Steam Generator per EOP supplement 18, 'B' S/G ESDE Isolation Checklist, AND maintain 'A' S/G pressure within 50 psi of 'B' S/G after 'B' S/G indicates it has boiled dry.

DISTRACTOR ANALYSIS

a. **CORRECT –This strategy ensures that when the faulted S/G blows dry, the unaffected S/G will remain coupled with the PCS and prevent PCS heatup and re-pressurization (PTS Concern)**
b. INCORRECT – Plausible if the student believes that 'A' S/G should NOT be steamed until after 'B' S/G is dry to prevent adding to the cooldown.
c. INCORRECT – Plausible if the student believes that EOP-9.0 is the correct procedure to enter. The student could believe that there is an ESDE in 'A' S/G also since T_C in both loops is lowering.
d. INCORRECT – same as 'b' above.

Level of Knowledge: APPLICATION **Difficulty: 2**

WRITTEN QUESTION DATA SHEET

Source of Question: NEW
K/A: 000062 Loss of Nuclear Service Water
AA2.06 Ability to determine and interpret the following as they apply to the Loss of Nuclear Service Water: The length of time after the loss of SWS flow to a component before that component may be damaged.

Tier: 1 Group: 1 SRO Imp: 3.1
Applicable 10CFR55 Section: 43.5, Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations. ***This exam question meets the criteria for an SRO-only question because the operator must determine which are the correct automatic actions and select the procedure that gives the needed guidance.***

Palisades Learning Objective: TBAR_E3.01 ***Given plant conditions, describe the actions necessary to minimize thermal and hydraulic shock to various plant components in accordance with EOP Supplement 24***

References: EOP-3; AOP-35

Question:

The plant was operating at full power.

- At 1510 the plant tripped and offsite power was lost.
- EK-0532, BUS 1C OR 1D OVERCURRENT LOCKOUT, is lit and Bus 1D is deenergized.
- D/G 1-1 is running but its output breaker did not automatically close.
- At 1522 the transition to EOP-3.0, Station Blackout Recovery, is made.
- At 1523 an NPO was able to close the 1-1 D/G output breaker and Bus 1C is now energized.
- The sequencer did NOT operate.
- Jacket water and Lube oil temperature on both D/Gs are 105°F and rising very slowly.

What action should now be taken and why should it be done?

a. Direct the NPO to immediately trip both of the D/Gs to minimize the likelihood of damaging the D/Gs.

b. Direct the NCO to immediately start P-7B SW pump so that Maintenance of Vital Auxiliaries –Water is satisfied.

c. Close CV-1359 Non Crit SW Hdr Isol and throttle service water pump discharge valve 2 turns open then start P-7B SW pump to minimize hydraulic shock.

d. Exit EOP-3.0, Station Blackout Recovery, and immediately return to EOP-1, Standard Post-Trip Actions, to rediagnose because we no longer satisfy the entry criteria.

DISTRACTOR ANALYSIS

a. INCORRECT Plausible because neither D/G has cooling water but the given temperatures allow for the time to restore the SW before component damage occurs.
b. INCORRECT Plausible because the pump needs to be started but because 10 minutes have elapsed, damage could occur to the piping so we are instructed to take actions to prevent the resultant hydraulic shock.
c. **CORRECT EOP-3.0 at step 4 directs us to restore SW cooling per AOP-35 which has these actions stipulated to prevent component damage if the loss of service water has exceeded 10 minutes which it has in this question.**
d. INCORRECT Plausible since we are just entering the procedure when we have power restored but once the transition has occurred we continue in the procedure.

Level of Knowledge: APPLICATION **Difficulty: 3**

WRITTEN QUESTION DATA SHEET

Source of Question: NEW
K/A: 000028 Pressurizer Level Malfunction
G2.2.22-Knowledge of limiting conditions for operations and safety limits.

Tier: 1 Group: 2 SRO Imp: 4.7
Applicable 10CFR55 Section: 43.2 –Facility operating limitations in the technical specifications and their bases.
This exam question meets the criteria for an SRO-only question because the candidate must have knowledge of the application of Required Actions (Section 3) and Surveillance Requirements (SR) (Section 4) in accordance with rules of application requirements (Section 1)

Palisades Learning Objective: **PLCS_CK21.0,** ***Given plant conditions and Technical Specifications, 3.3.7, 3.3.8, and 3.4.9 determine the. status of the associated LCO Condition(s) and applicable Required Actions and Completion Times for the Pressurizer Level Control System in accordance with Technical Specification 3.3.7, 3.3.8, and3.4.9 BASES***

References: TS LCO 3.4.9 Pressurizer

Question:

The plant is operating in a normal full power lineup.

- At 1315 the controlling pressurizer level channel fails low.
- At 1320 pressurizer level is 63% and rising.
- Assume no operator action occurs.

What is the most limiting Technical Specification ACTION, (i.e. shortest COMPLETION time), that applies?

a. Be in MODE 3 with the reactor tripped within 6 hours.

b. Take action within 1 hour to place the plant in MODE 3 within 7 hours.

c. Restore Pressurizer level within 6 hours or be in MODE 3 with the reactor tripped within the next 6 hours.

d. Restore Pressurizer heaters within 72 hours or be in MODE 3 within the next 6 hours.

DISTRACTOR ANALYSIS

a. **CORRECT at 1320 the pressurizer level exceeds its LCO and Condition A requires this action and the associated COMPLETION time.**
b. INCORRECT Plausible because the two inoperable heater power supplies places the plant in LCO 3.0.3 but it is less limiting that the correct answer.
c. INCORRECT Plausible if the operator confuses the completion time with the allowed outage time.
d. INCORRECT Plausible because Condition B completion time applies to this situation but is not most limiting.

Level of Knowledge: APPLICATION **Difficulty: 3**

WRITTEN QUESTION DATA SHEET

Source of Question: BANK
K/A: 000033 Loss of Intermediate Range NI
AA2.10-Ability to determine and interpret the following as they apply to the Loss of Intermediate Range Nuclear Instrumentation: Tech-Spec limits if both intermediate-range channels have failed

Tier: 1 Group: 2 SRO Imp: 3.8
Applicable 10CFR55 Section: 43.5, Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations ***This exam question meets the criteria for an SRO-only question because the operator must use the TS Bases to determine which NI channels meet the LCO 3.3.9 criterion for neutron flux monitor. Also the applicability criteria for MODE 5 are contained in a footnote.***

Palisades Learning Objective: NI_CK22.0 ***From memory, describe the Technical Specification bases for the Nuclear Instrumentation System in accordance with Technical Specification 2.1.1, 2.2.1, 3.2.1, 3.2.2, 3.2.3, 3.2.4, 3.3.1, 3.3.7, 3.3.8, 3.3.9, and 3.9.2 Bases.***

References: LCO 3.3.9 and LCO 3.3.1

Question:

Given the following after a Plant shutdown five days ago:

- The Plant is in MODE 5
- All CRDM Clutch Toggle Switches are Caution Tagged OFF
- Source Range Nuclear Instrument 1A is indicating 30 cps
- Source Range Nuclear Instrument 2A is indicating 35 cps
- Wide Range Nuclear Instrument 3A is INOPERABLE due to I&C maintenance
- Wide Range Nuclear Instrument 4A then de-energizes due to an internal power supply fault

Which one of the following describes the status of LCO 3.3.1, "RPS Instrumentation," and LCO 3.3.9, "Neutron Flux Monitoring Channels," for these conditions?

a. LCO 3.3.1 is NOT <u>applicable</u>.
LCO 3.3.9 is NOT met due to one or more Neutron Flux Monitoring Channels INOPERABLE.

b. LCO 3.3.1 is NOT met due to two ZPM Bypass Removal Channels INOPERABLE.
LCO 3.3.9 is NOT met due to one or more Neutron Flux Monitoring Channels INOPERABLE.

c. LCO 3.3.1 is NOT met due to two ZPM Bypass Removal Channels INOPERABLE.
LCO 3.3.9 is NOT <u>applicable</u>.

d. LCO 3.3.1 is NOT <u>applicable</u>.
LCO 3.3.9 is met.

DISTRACTOR ANALYSIS

a. Plausible if the operator misapplies the definition of a neutron flux monitoring channel from the LCO bases.
b. Plausible if the operator misapplies the definition of a neutron flux monitoring channel from the LCO bases and believes that LCO 3.3.1 is applicable.
c. Plausible if the operator believes that LCO 3.3.1 is applicable because it could be applicable in MODE 5 if the rods were capable of being withdrawn.
d. **CORRECT - None of the RPS trips associated with Wide Range NIs are applicable in LCO 3.3.1 with either boron greater than refueling or no more than one control rod capable of being withdrawn.**

Level of Knowledge: APPLICATION **Difficulty: 4**

WRITTEN QUESTION DATA SHEET

Source of Question: MODIFIED BANK
K/A: 000059 Accidental Liquid Radwaste Release
AA2.05 Ability to determine and interpret the following as they apply to the (Accidental Liquid Radwaste Release). The occurrence of automatic safety actions as a result of a high PRM system signal.

Tier: 1 Group: 2 SRO Imp: 4.5

Applicable 10CFR55 Section: 43.5 – Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations. ***This exam question meets the criteria for an SRO-only question because the operator must assess plant conditions (normal, abnormal, or emergency) and then select a procedure or section of a procedure to mitigate, recover, or with which to proceed.***

Palisades Learning Objective: RMS_CK09.0 ***From memory, describe the design features and interlocks that provide the following Radiation Monitoring System functions: Release terminations when radiation exceeds setpoint***

References: ODCM and ARP-8

Question:

Given the following:

- A liquid radwaste batch release of T-91, Utility Water Storage Tank, is in progress.
- Ten minutes into the release the following alarm annunciates: EK-1365, PROCESS LIQ MONITORING HIGH RADIATION
- The NCO verifies RIA-1049, Liquid Radwaste Monitor, in alarm.

Which of the following (1) actions will occur automatically as result of RIA-1049, Liquid Radwaste Monitor exceeding its high alarm setpoint and (2) which procedure will direct your actions?

a. (1) Both CV-1049, 3" Radwaste Discharge Line to Mixing Basin", and CV-1051, "Treated Waste Monitor P-58 A/B Bypass" close.
(2) Use ODCM Appendix A, "Relocated Technical Specifications per NRC Generic Letter 89-01 (TAC 75060)."

b. (1) Both CV-1049, 3" Radwaste Discharge Line to Mixing Basin", and CV-1051, "Treated Waste Monitor P-58 A/B Bypass" close.
(2) Use ARP-8, "Safeguards Safety Injection and Isolation Scheme EK-13 (EC-13)."

c. (1) Utility Water Tank T-91 pumps, P-91A/B trip, and CV-1054, "Discharge to Mixing Basin" closes.
(2) Use ODCM Appendix A, "Relocated Technical Specifications per NRC Generic Letter 89-01 (TAC 75060)."

d. (1) Utility Water Tank T-91 pumps, P-91A/B trip, and CV-1054, "Discharge to Mixing Basin" closes.
(2) Use ARP-8, "Safeguards Safety Injection and Isolation Scheme EK-13 (EC-13)."

DISTRACTOR ANALYSIS

a. Plausible because the automatic actions are correct however the ODCM discusses RIA-1049 but only for operability purposes.
b. **Correct – automatic actions are correct and ARP-8 contains appropriate guidance.**
c. Plausible because tripping transfer pumps and isolating discharge valve stop release but these are not automatic as a result of the high Rad alarm and ODCM is not correct as stated above.
d. Plausible because tripping transfer pumps and isolating discharge valve stop release but these are not automatic as a result of the high Rad alarm.

Level of Knowledge: APPLICATION **Difficulty: 3**

WRITTEN QUESTION DATA SHEET

Source of Question: **BANK USED ON 2012 PALISADES EXAM**
K/A: 000074 (W/E06&E07) Inad. Core Cooling
G2.4.46 Ability to verify that the alarms are consistent with the plant conditions.

Tier: 1 Group: 2 SRO Imp: 4.2
Applicable 10CFR55 Section: 43.5 - Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations. ***This exam question meets the criteria for an SRO-only question because the candidate must assess the facility conditions given in the stem and use those conditions to select the appropriate procedure to mitigate the consequences of a Reactor Head void following a loss of coolant accident.***

Palisades Learning Objective: TBCORE_CK05.0 - ***Given an event involving EOPs, describe the expected plant or instrument response.***

References: EOP-4.0, step 86 and 87

Question:

Given the following:

- The Plant was tripped from full power due to a LOCA coincident with a Loss of Offsite Power.
- EOP-4.0, "Loss of Coolant Accident Recovery," has been implemented
- Pressurizer pressure is 1300 psia
- Core ΔT (CET - T_C) is 55°F and slowly rising
- Corrected Pressurizer level is 15% and rising
- EK-0748, REACTOR WATER LEVEL LOW is in alarm
- The top 5 red sensor lights are LIT on the Reactor Vessel Level Monitoring System

Which one of the following (1) describes an additional indication that a steam bubble has formed in the Reactor Head region for these conditions and (2) the procedure, if any, that the Control Room team will implement to mitigate the bubble formation?

a. (1) Average Core Exit Thermocouple temperature is 575°F
(2) EOP Supplement 26, "PCS Void Removal."

b. (1) Average Core Exit Thermocouple temperature is 575°F
(2) None. EOP-4.0 contains actions to vent the Reactor Head for these conditions.

c. (1) Erratic Nuclear Instrumentation indications
(2) EOP Supplement 26, "PCS Void Removal."

d. (1) Erratic Nuclear Instrumentation indications
(2) None. EOP-4.0 contains actions to vent the Reactor Head for these conditions.

DISTRACTOR ANALYSIS

a. **CORRECT – CET temp indicates subcooled fluid at core exit, Pzr level rising indicates transfer of bubble to head, Core delta T indicates inadequate core cooling, and EOP supplement 26 is directed by EOP 4.0 for void removal**
b. Plausible if student correctly recognizes the indication that a bubble has formed but believes that EOP-4.0 contains guidance to vent the head which would exacerbate the situation.
c. Plausible because erratic NI indications would exist for a bubble in the head but it would also have to uncover the fuel in order to see those indications with NIs. With 3 green lights lit on RVLMS, this indication would not be expected.
d. Plausible for combination of 'b' and 'c'.

Level of Knowledge: ANALYSIS **Difficulty: 4**

WRITTEN QUESTION DATA SHEET

Source of Question: MODIFIED BANK
K/A: 004 Chemical and Volume System
A2.17 Ability to (a) predict the impacts of the following malfunctions or operations on the CVCS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Low PZR pressure

Tier: 2 Group: 1 SRO Imp: 3.7
Applicable 10CFR55 Section: 43.5 Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations. . ***This exam question meets the criteria for an SRO-only question because the candidate must have knowledge of the diagnostic steps and decision points in procedures that involve transitions to event specific sub-procedures or emergency contingency procedures.***

Palisades Learning Objective: CVCS_CK13.0 ***From memory, predict how the following conditions will impact operation of the CVCS: PZR pressure and level***

References: AOP-23, Primary Coolant Leak

Question:

Given the following with the plant in MODE 1 operating at 100% power:

- Letdown flow is 40 gpm and stable
- Charging flow is 49 gpm and rising
- Containment Sump level trend on plant process computer, PPC, is rising
- Pressurizer level is 57% and stable
- PCS Pressure is 2050 psia and lowering slowly

Which one of the following statements correctly describes (1) the proper procedure to address the problem, and (2) for the stated conditions, when would you direct the reactor to be tripped?

a. (1) Enter AOP-23, "Primary Coolant Leak"
(2) As soon as unidentified leak rate exceeds 10 gpm.

b. (1) Enter AOP-23, "Primary Coolant Leak"
(2) As soon as identified leak rate exceeds 25 gpm.

c. (1) Enter AOP-28, "Pressurizer Pressure Control Malfunctions"
(2) Either Pressurizer Spray Valve failed open AND Pressurizer pressure not being maintained.

d. (1) Enter AOP-28, "Pressurizer Pressure Control Malfunctions"
(2) Both Pressurizer Spray Valves failed open AND Pressurizer pressure not being maintained.

DISTRACTOR ANALYSIS

a. **CORRECT Since charging flow is still rising and Pzr pressure is lowering, this indicates PCS inventory loss versus Pzr pressure malfunction which leads to AOP-23. Trip criteria is correct for this AOP.**
b. Plausible if the operator confuses the reactor trip criteria with the EAL criteria of SU8.1 (PCS Leakage).
c. Plausible because PCS pressure is seen to be lowering but the cause is a PCS leak on the Pzr Pressure Transmitter sensing line. The trip criteria statement is correct for the procedure.
d. Plausible as in 'c' and the trip criteria are similar to those given in the procedure.

Level of Knowledge: **APPLICATION** **Difficulty:** **3**

WRITTEN QUESTION DATA SHEET

Source of Question: **BANK**
K/A: 006 Emergency Core Cooling
G2.4.8 – Knowledge of how abnormal operating procedures are used in conjunction with EOPs.

Tier: 2 Group: 1 SRO Imp: 4.5
Applicable 10CFR55 Section: 43.5 - Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations. ***This exam question meets the criteria for an SRO-only question because the candidate must assess the facility conditions given in the stem and use those conditions to select the appropriate procedure to mitigate the consequences of having PZR heaters deenergized following a loss of coolant accident***

Palisades Learning Objective: TBCORE_CK05.0 ***Given plant conditions or an event involving Emergency Operating Procedures, describe the expected plant or instrument response in accordance with the in-use Emergency Operating Procedure***

References: EOP-4.0, AOP-28

Question:

Given the following with the Plant at full power:

- A Plant trip occurs due to a small break LOCA coincident with a loss of offsite power
- EOP-4.0, "Loss of Coolant Accident Recovery," is being implemented
- SIAS has actuated and all equipment operated as designed
- Pressurizer (PZR) level is currently 37% and rising slowly
  - Lowest PZR level noted was 15%
- Containment pressure is 0.6 psig and rising slowly
- Containment temperature is 135°F and rising slowly

Which of the following (1) describes the current status of the PZR heaters and (2) the action, if any, required to energize the available PZR heaters in accordance with EOP-4.0?

a. (1) No PZR heaters are energized.
(2) Use AOP-28, "Pressurizer Pressure Control Malfunctions," to energize PZR heaters.

b. (1) Half the PZR heaters are energized.
(2) Use AOP-28, "Pressurizer Pressure Control Malfunctions," to energize PZR heaters.

c. (1) No PZR heaters are energized.
(2) Raise PZR level to > 40% and then energize PZR heaters per EOP-4.0.

d. (1) Half the PZR heaters are energized.
(2) Use SOP-30, "Station Power," to energize PZR heaters.

DISTRACTOR ANALYSIS

a. **CORRECT – 'E' Bus heaters deenergized due to LOOP, 'D' bus heaters can be recovered using AOP-28 as directed from EOP 4.0.**
b. Plausible, if the operator believes 'D' bus heaters automatically reenergized by 1-2 D/G.
c. Plausible, because there are no heaters energized but the level stated is for degraded containment.
d. Plausible, if the operator believes 'D' bus heaters automatically reenergized by 1-2 D/G.

Level of Knowledge: ANALYSIS **Difficulty: 2**

WRITTEN QUESTION DATA SHEET

Source of Question: **MODIFIED BANK**
K/A: **008 Component Cooling Water**
A2.02 - Ability to (a) predict the impacts of the following malfunctions or operations on the CCWS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: High/low surge tank level

Tier: 1 Group: 1 SRO Imp: 3.5
Applicable 10CFR55 Section: 43.5 - Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations. ***This exam question meets the criteria for an SRO-only question because the candidate must assess the facility conditions given in the stem and use those conditions to select the appropriate procedure to mitigate the consequences of a low CCW surge tank level.***

Palisades Learning Objective: IOTF_CK03.0 ***Given off normal plant conditions, select the applicable Abnormal Operating Procedure to mitigate the event without error.***

References: AOP-36.

Question:

Given the Plant at full power, the following conditions are noted:

Indication	Time 1300:00	Time 1302:30
LIA-0916 (CCW Surge Tank level)	74%	49% stable
CV-0918 (CCW Surge Tank makeup fill)	Green indication	Red indication

At 1311 a Nuclear Plant Operator informs the control room of the following:

- There is water flowing out of the 'A' Evaporator Room floor area
- EK-3230, AUX BLDG SUMP LEVEL has annunciated at Rad Waste Panel C-105

Which one of the following (1) describes the correct procedure to mitigate this event and (2) the action that will be directed by Control Room Supervisor?

a. (1) SOP-16, "Component Cooling Water System"
 (2) Open CV-0918 bypass valve and manually fill the CCW Surge Tank

b. (1) SOP-16, "Component Cooling Water System"
 (2) Close CV-0944A, CCW To SFP HXs, RW Evaps.

c. (1) AOP-36, "Loss of Component Cooling"
 (2) Start an additional CCW Pump

d. (1) AOP-36, "Loss of Component Cooling"
 (2) Isolate CCW to SFP Heat Exchangers and Waste Gas Compressors

DISTRACTOR ANALYSIS

a. Plausible if the operator fails to recognize AOP-36 Entry Conditions met.
b. Plausible because this action will partially isolate the leak but not completely..
c. Plausible if the operator believes starting additional CCW Pumps will raise the Surge Tank fill rate.
d. **CORRECT The postulated CCW leak is in the 'A' Evaporator Room; leak size matches the capacity of the fill valve (CV-0918), so CCW Surge Tank level is stable at a level greater than the lo level alarm setpoint; Level not being recovered meets AOP-36 Entry Conditions.**

Level of Knowledge: ANALYSIS **Difficulty: 3**

WRITTEN QUESTION DATA SHEET

Source of Question: NEW
K/A: 059 Main Feedwater
G2.4.20-Knowledge of the operational implications of EOP warnings, cautions, and notes.

Tier: 2 Group: 1 SRO Imp: 4.3
Applicable 10CFR55 Section: 43.5 – Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations. ***This exam question meets the criteria for an SRO-only question because the candidate must assess plant conditions and then apply specific limitations of feed flow to a nearly dry Steam Generator correctly.***

Palisades Learning Objective: TBCORE_CK02.0, ***Given plant conditions involving Emergency Operating Procedures, describe the bases of any EOP step, note, caution or warning in accordance with the Emergency Operating Procedure Bases Document.***

References: EOP-7 LOSS OF ALL FEEDWATER RECOVERY step 8 Caution

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Question:

Given the following:

- Plant was manually tripped due to a Loss of Feedwater event.
- EOP 7.0, Loss Of All Feedwater Recovery, is in use.
- 2400 VAC Bus 1C is deenergized.
- Annunciator EK-0532 BUS 1C OR 1D OVERCURRENT LOCKOUT, is in.
- P-8C was out of service for corrective maintenance.
- P-8B tripped on overspeed.
- The crew has successfully performed a Hot Restart of P-1A, Main Feedwater Pump.
  - 'A' S/G level is at -65% and lowering
  - 'B' S/G level is at -92% and lowering

Based on the above conditions, select the statement that explains how to properly feed the S/Gs.

a. Feed both S/Gs at whatever rate is needed to raise S/G level.

b. Feed 'A' S/G at whatever rate is needed to raise S/G level and 'B' S/G at < 300 gpm.

c. Feed 'A' S/G at whatever rate is needed to raise S/G level and do not feed 'B' S/G.

d. Feed 'B' S/G at whatever rate is needed to raise S/G level and 'A' S/G at < 300 gpm.

DISTRACTOR ANALYSIS

a. INCORRECT - Plausible if the operator neglects the Caution prior to step 8.
b. **CORRECT Caution statement prior to Step 8 specifically says:** ***Limit feed flow to less than 300 gpm for any S/G with level less than -84%.***
c. INCORRECT – Plausible if the operator believes that you cannot feed a S/G with level < -84%.
d. INCORRECT Plausible if the operator misapplies the Caution prior to step 8.

Level of Knowledge: ANALYSIS **Difficulty: 3**

WRITTEN QUESTION DATA SHEET

Source of Question: BANK
K/A: 103 Containment
A2.03 - Ability to (a) predict the impacts of the following malfunctions or operations on the containment system and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Phase A and B isolation

Tier: 2 Group: 1 SRO Imp: 3.8
Applicable 10CFR55 Section: 43.5 Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations. ***This exam question meets the criteria for an SRO-only question because the candidate must recall what strategy or action is written into a plant procedure, including when the strategy or action is required.***

Palisades Learning Objective: IOTF_CK09.0 ***Given Abnormal Operating plant conditions and control room references, SELECT the applicable Technical Specification LCO REQUIRED ACTIONS and COMPLETION TIMES in accordance with Technical Specifications.***

References: AOP-31, AOP-32, LCO 3.6.3

Question:

The Plant is at full power when a spurious Containment Isolation occurs. During performance of EOP Supplement 6, "Checksheet for Containment Isolation and CCW Restoration," the following indications exist:

Valve	Indication
CV-1002, Primary System Drain Tank T-74 Outlet Isol	Red light OFF, Green light ON
CV-1007, Primary System Drain Tank T-74 Outlet Isol	Red light ON, Green light OFF

A Nuclear Plant Operator in the field reports the following:

- Air supply pressure reading for CV-1002 is 0 (zero) psig
- Air supply pressure reading for CV-1007 is 0 (zero) psig
- CV-1002 indicates CLOSED locally
- CV-1007 indicates OPEN locally

Which one of the following describes (1) the minimum action(s) necessary for the indication associated with CV-1007 and (2) the procedure that contains the action(s)?

a. (1) Place CV-1002 handswitch to CLOSE position only.
   (2) AOP-31, "Spurious Containment Isolation."

b. (1) Place CV-1002 handswitch to CLOSE position and close and lock the air supply valve to the operator.
   (2) AOP-31, "Spurious Containment Isolation."

c. (1) Place CV-1002 handswitch to CLOSE position and close and lock the air supply valve to the operator.
   (2) AOP-32, "Loss of Containment Integrity."

d. (1) Place CV-1002 handswitch to CLOSE position only.
   (2) AOP-32, "Loss of Containment Integrity."

DISTRACTOR ANALYSIS

a. Plausible if the student believes that AOP-31 is the correct procedure and that the valve handswitch only needs to be in close.
b. Plausible if the student believes that AOP-31 is the correct procedure since that was the initiating problem.
c. **CORRECT - AOP-32 directs this action and the actions meet the definition of "de-activated" which is a single active failure will not allow the valve to open.**
d. Plausible, however the air supply is already 0 PSIG and the valve is still open.

Level of Knowledge: APPLICATION **Difficulty: 3**

WRITTEN QUESTION DATA SHEET

Source of Question: MODIFIED BANK
K/A: 015 Nuclear Instrumentation
G2.2.40-Ability to apply Technical Specifications for a system.

Tier: 2 Group: 2 SRO Imp: 4.7
Applicable 10CFR55 Section: 43.2 – Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations. ***This exam question meets the criteria for an SRO-only question because the candidate must apply required actions for the Operating Requirements Manual (ORM).***

Palisades Learning Objective: NI_CK18.0 Given plant conditions and the Operating Requirements Manual, determine the required ORM actions for the NI system in accordance with ORM section 3.17.6.12.

References: ORM 3.17.6.12 (PROVIDE ORM Section 3.17.6)

Question:

The following conditions exist:

- Plant is at 98% power
- 'A' Channel Flux ΔT comparator inoperable

Which one of the following describes the operational limitations if NI-08, Channel 'D' Power Range Instrument, fails?

a. Power operation at current levels may be continued provided that Quadrant Power Tilt Ratio is manually calculated every 12 hours.

b. Within one hour take action to place the plant in MODE 3 within 7 hours.

c. Power must be reduced to < 50% rated thermal power within 4 hours.

d. Power must be reduced to < 70% rated thermal power within 2 hours.

DISTRACTOR ANALYSIS

a. INCORRECT Plausible if the operator believes that 3.17.6.15 applies and that QPTR issue is the only constraint
b. INCORRECT Plausible if the operator believes that 2 inoperable channels would require entry into LCO 3.0.3
c. INCORRECT Plausible if the operator takes the action for high QPTR
d. **CORRECT 'D' Channel Pwr Range NI is an input to 'D' Channel Flux ΔT Comparator so two channels are inoperable. ORM 3.17.6.12 requires that with two channels inoperable, power must be maintained <70% to assure no unobserved flux tilt causes local power limits to be exceeded.**

Level of Knowledge: APPLICATION **Difficulty: 3**

WRITTEN QUESTION DATA SHEET

Source of Question: BANK
K/A: 034 Fuel Handling Equipment
A4.01-Ability to manually operate and/or monitor in the control room: radiation levels

Tier: 2 Group: 2 SRO Imp: 3.7
Applicable 10CFR55 Section: 43.7 – Fuel handling facilities and procedures. *This exam question meets the criteria for an SRO-only question because the candidate must determine the appropriate emergency classification.*

Palisades Learning Objective: PL-N00113_E01.09, *Given plant conditions involving Emergency Operating Procedures, describe the bases of any EOP step, note, caution or warning in accordance with the Emergency Operating Procedure Bases Document.*

References: Palisades Site Emergency Plan Supplement 1 EAL Wall Charts (PROVIDE)

Question:

The Plant is in day 2 of a refueling outage, currently in MODE 5.

- The Spent Fuel Pool Gate is installed.
- The Tilt pit is still drained.
- Annunciator EK-1309 SPENT FUEL POOL HI/LO LEVEL is alarming.
- A report is received that the local Spent Fuel Pool water level reading is 633 ft and stable.
- RIA-5709, Spent Fuel Pool Area Radiation Monitor is reading 10 mR/hr and rising slowly.

Based on the above indications, determine the appropriate classification and EAL for this event.

a. Unusual Event – AU2.1

b. Alert – AA2.1

c. Unusual Event – CU3.2

d. Alert – AA2.2

DISTRACTOR ANALYSIS

a. **CORRECT because SFP level is <646 ft. AND a valid high radiation reading exists on RIA-5709.**
b. INCORRECT Plausible if candidate thinks 10 mR/hr meets Alert threshold. Should be >15 mR/hr for Alert.
c. INCORRECT Plausible if candidate misapplies Note 3 from the Wall Chart.
d. INCORRECT Plausible if candidate believes SFP level has or will uncover the fuel.

Level of Knowledge: APPLICATION **Difficulty: 3**

WRITTEN QUESTION DATA SHEET

Source of Question: **NEW**
K/A: 035 Steam Generator
A2.06- Ability to (a) predict the impacts of the following malfunctions or operations on the S/GS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Small break LOCA.

Tier: 2 Group: 2 SRO Imp: 4.6
Applicable 10CFR55 Section: 43.5 – Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations. ***This exam question meets the criteria for an SRO-only question because the candidate must evaluate the mitigation strategy and determine the expected outcome from the actions that will be taken in the procedure.***

Palisades Learning Objective: TBCORE_CK01.0, Given plant conditions involving Emergency Operating Procedure, describe the mitigating strategy of the in use Emergency Operating Procedure in accordance with the Emergency Operating Procedure Bases Document

References: EOP-4 LOSS OF COOLANT ACCIDENT RECOVERY step 43 RNO. EOP-4.0 Basis document page 3.

Question:

The plant was operating at full power when the following occurred.

- The reactor tripped due to lowering PCS pressure
- Off-site Power (LOOP) was lost concurrently
- Safety injection automatically actuated

The crew has transitioned to EOP-4.0, Loss of Coolant Accident Recovery, and the following conditions are noted.

- PCS pressure is 1000 psia and rising slowly
- HPSI flow is 125 gpm per loop
- CETs are 545°F and rising slowly
- T_cs are 538°F and rising slowly

Based on this information, what mitigation strategy should be employed in EOP-4.0 and for what reason?

a. Raise PCS Pressure to improve natural circulation.

b. Raise steam flow to enhance natural circulation.

c. Raise PCS Pressure to improve subcooling.

d. Raise steam flow to establish subcooling.

DISTRACTOR ANALYSIS

a. INCORRECT Plausible because natural circulation flow is the desired effect but because the S/Gs are the PCS heat sink, the problem with the plant conditions is poor natural circulation which can only be corrected by adjusting the heat sink.
b. **CORRECT because natural circulation cannot be established unless the S/Gs have adequate steam and feed flow. So if natural circulation is not present, that can be corrected by better control of S/Gs.**
c. INCORRECT PCS pressure, and hence subcooling, is low because of the small break LOCA so this strategy will not be effective.
d. INCORRECT Plausible because the PCS is saturated so the lack of subcooling is not correctable by better control of S/Gs. While it would lower temperature, it would also lower pressure therefore the PCS would remain saturated.

Level of Knowledge: Analysis **Difficulty: 3**

WRITTEN QUESTION DATA SHEET

Source of Question: **BANK**

K/A: **G2.1.5 - Ability to use procedures related to shift staffing, such as minimum crew complement, overtime limitations, etc.**

Tier: **3** **Group:** **SRO Imp:** **3.9**

Applicable 10CFR55 Section: **43.1 - Conditions and limitations in the facility license.** ***This exam question meets the criteria for an SRO only question because the candidate must apply knowledge contained in 10CFR26, "Fitness for Duty Programs," subpart I to determine who must determine that a waiver is necessary and who must authorize the waiver. This is a condition of the Palisades license because the license states that the facility must operate the plant within the regulations of the commission (NRC). See Palisades Renewed License No. DPR-20 section 1.D. This item also is <u>not</u> required knowledge for Reactor Operators.***

Palisades Learning Objective: **APCO_E14.01** ***Given references explain the following:***

a. Covered Worker
b. Covered Work
c. Exception to work hour controls
d. Opt-in/Opt-out of work hour controls
e. Work Hours
f. Work hour controls
g. Work hour controls waivers
h. Work hour limits for covered individuals

in accordance with EN-OM-123.

References: **10CFR26.207; EN-OM-123, 5.9**

Question:

Which one of the following is <u>NOT</u> a requirement regarding administrative controls to limit the working hours of personnel who perform safety-related functions?

a. Any individual who will exceed fatigue rule guidelines must be capable of performing the assigned duties without impairment due to fatigue.

b. Any deviation from the fatigue rule guidelines shall be authorized by the Operations Manager.

c. Any deviation from the fatigue rule guidelines shall be authorized in advance.

d. Any deviation from the fatigue rule guidelines shall have documentation of the basis for granting the deviation.

DISTRACTOR ANALYSIS

a. Correct statement which makes the distracter incorrect.
b. **CORRECT because the Operations Manager is NOT allowed to authorize this. It must be SVP or GMPO.**
c. Correct statement which makes the distracter incorrect.
d. Correct statement which makes the distracter incorrect.

Level of Knowledge: **COMPREHENSION** **Difficulty:** **2**

WRITTEN QUESTION DATA SHEET

Source of Question: NEW
K/A: G2.1.32 - Ability to explain and apply system limits and precautions

Tier: Group: SRO Imp: 4.0
Applicable 10CFR55 Section: 43.2 - Facility operating limitations in the technical specifications and their bases.
This exam question meets the criteria for an SRO only question because the candidate must apply knowledge from TS Bases 3.5.2 that describes how to satisfy the required action from the LCO.

Palisades Learning Objective: SIS_CK22.0 ***From memory, describe the Technical Specification bases for the Safety Injection System in accordance with Technical Specification***

References: TS 3.5.2 Bases page 8

Question:

Given the plant is operating at full power. P-66B 'B' HPSI pump is inoperable.

Which of the following equipment being declared inoperable would require the plant to be placed in MODE 3 within seven hours per LCO 3.0.3?

a. CV-3070 HPSI Pump P-66B Subcooling Valve

b. CV-3071 HPSI Pump P-66A Subcooling Valve

c. P-67A, 'A' LPSI pump

d. P-67B, 'B' LPSI pump

DISTRACTOR ANALYSIS

a. INCORRECT Plausible because the inoperable subcooling valve renders its associated HPSI pump incapable of performing its design function but the B HPSI pump is already inoperable (given in the stem).
b. **CORRECT the inoperable subcooling valve renders the A HPSI pump incapable of performing its design function so there is no high head injection available for a large break LOCA therefore we do not have 100% ECCS capacity.**
c. INCORRECT Plausible because neither the A Train nor the B Train is completely available. But there is 100% availability without having to consider a further active failure which is consistent the LCO Bases document.
d. INCORRECT Plausible because all of the Train B ECCS flow is inoperable but the A Train is available.

Level of Knowledge: APPLICATION **Difficulty: 2**

WRITTEN QUESTION DATA SHEET

Source of Question: BANK USED ON 2010 PALISADES EXAM

K/A: G2.2.18 - Knowledge of the process for managing maintenance activities during shutdown operations, such as risk assessments, work prioritization, etc.

Tier: 3 Group: SRO Imp: 3.9

Applicable 10CFR55 Section: 43.5 - Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations. ***This exam question meets the criteria for an SRO only question because the candidate must assess the facility conditions given in the stem, i.e., Plant MODE, and use those conditions to select the appropriate procedure for performing a risk assessment. Plant MODE is required in the stem because the risk assessment procedure changes if the Plant is below MODE 3. This exam question also meets the criteria for an SRO only question because performing a risk assessment is an SRO only task.***

Palisades Learning Objective: ADAO_E02.02 ***Given plant conditions and the need to perform a Risk Assessment, perform a Risk Assessment in accordance with AP-4.02.***

References: ADMIN 4.02, attachment 3, section 2.0

Question:

Given the following conditions:

- The Plant is in MODE 5
- A severe thunderstorm warning is in effect
- Irradiated fuel moves are in progress in the Spent Fuel Pool
- EY-30, Preferred AC Bus, is planned to be removed from service on the next shift

Which one of the following procedures will be used to perform a risk assessment for removing EY-30 from service?

a. Admin 4.02, "Control of Equipment."

b. Admin 4.11, "Safety Function Determination Program."

c. GOP-14, "Shutdown Cooling Operations."

d. GOP-11, "Refueling Operations and Fuel Handling."

DISTRACTOR ANALYSIS

a. Plausible since this is the procedure that would be used with the plant in MODE 1, 2, or 3.
b. Plausible since this procedure addresses safety functions.
c. **CORRECT - This procedure is used to perform a risk assessment anytime shutdown cooling is in service.**
d. Plausible since this procedure is in effect and the student believes that there is a process for performing a risk assessment during fuel moves.

Level of Knowledge: COMPREHENSION **Difficulty: 3**

WRITTEN QUESTION DATA SHEET

Source of Question: **BANK**
K/A: G2.2.20 - Knowledge of the process for managing troubleshooting activities.

Tier: 3 Group: SRO Imp: 3.8

Applicable 10CFR55 Section: 43.5 - Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations. ***This exam question meets the criteria for an SRO only question because the candidate must have knowledge of the diagnostic steps and decision points in the emergency operating procedures (EOP) that involve transitions to event specific sub-procedures or emergency contingency procedures.***

Palisades Learning Objective: ADAO_E02.02 ***Given plant conditions and the need to perform a Risk Assessment, perform a Risk Assessment in accordance with AP-4.02.***

References: ADMIN 4.02, attachment 3, section 2.0

Question:

A Station Blackout has occurred. As the Control Room Supervisor, which procedure is used to determine which relays must be checked to evaluate which power source to restore first?

a. SOP-32, "345KV Switchyard" attachment which lists all Switchyard relays and expected status of those relays for a Station Blackout event.

b. EOP Supplement 28, "Supplementary Actions for Loss of Power" which directs use of EOP Supplement 22, "Switchyard Relay/Target List".

c. EOP Supplement 21, "Restoration of 'F' or 'R' Buses" which directs use of EOP Supplement 22, "Switchyard Relay/Target List".

d. EOP-3.0, "Station Blackout Recovery" Step 16 which directs use of EOP Supplement 29, "Restore Buses 1C, 1D, 1E Power from Off-Site Source".

DISTRACTOR ANALYSIS

a. Candidate incorrectly assumes by virtue of the procedure title that it contains the needed guidance. However, SOP-32 is primarily directed towards routine activities and not specific to troubleshooting.
b. Candidate incorrectly assumes by virtue of the procedure title that it contains the needed guidance. However, this procedure has very little to do with the electrical system, and is primarily compensatory actions for equipment which had lost power.
c. **CORRECT - These two supplements are used together for the required actions.**
d. Candidate incorrectly assumes by virtue of the procedure title that it contains needed guidance, but also incorrectly applies procedure intent. This procedure is used after an offsite power source is restored.

Level of Knowledge: APPLICATION **Difficulty: 3**

WRITTEN QUESTION DATA SHEET

Source of Question: **BANK**
K/A: G2.3.4 - Knowledge of radiation exposure limits under normal or emergency conditions

Tier: 3 Group: SRO Imp: 3.7
Applicable 10CFR55 Section: 43.4 - Radiation hazards that may arise during normal and abnormal situations, including maintenance activities and various contamination conditions. ***This exam question meets the criteria for an SRO-only question because the candidate must have knowledge of the radiological hazards associated with protecting plant equipment during an emergency. This exam questions also meets the requirements of an SRO-only question because performing the duties of the Emergency Plant Manager is an SRO job that cannot be delegated to an RO.***

Palisades Learning Objective: N00153_E18.0 ***Identify the individual by their Emergency Response title that must authorize exceeding Administrative and 10CFR Dose Limits in Emergency Conditions before allowing personnel into high exposure areas for any of the reasons given in EI-2.1, in accordance with EI-2.1***

References: EI-2.1, 5.11

Question:

An emergency entry must be made into an area in the Auxiliary Building that has high radiation levels to isolate a leak which is causing extensive damage to valuable company property.

Acting as the Emergency Plant Manager, the Shift Manager can authorize an individual selected to perform this task up to a MAXIMUM of:

a. > 2.0 rem but < 5.0 rem.

b. 5 rem.

c. 10 rem.

d. 25 rem.

DISTRACTOR ANALYSIS

a. Plausible because these dose limits are the Dose Control Limits that are established by the Emergency Plant Manager.
b. Plausible because this is the limit for all activities except those that are explicitly identified with higher limits.
c. **CORRECT Since the Shift Manager is the Emergency Plant Manager, then he/she can authorize this exposure to protect valuable plant equipment.**
d. Plausible because this is the limit for lifesaving activities or large population protection.

Level of Knowledge: COMPREHENSION **Difficulty: 3**

WRITTEN QUESTION DATA SHEET

Source of Question: USED ON 2007 PALISADES EXAM
K/A: G2.3.11 Ability to control radiation releases.

Tier: 3 Group: SRO Imp: 4.3
Applicable 10CFR55 Section: 43.4 Radiation hazards and contamination conditions that may occur during normal and abnormal situations, including maintenance activities and various contamination conditions. ***This is an SRO question because the SRO does the analysis and interpretation of radiation and activity readings as they pertain to selection of emergency procedures. This would also entail when the strategy or action is required per 43.5.***

Palisades Learning Objective: TBAF_E01.01, ***Given Plant conditions involving a Steam Generator Tube Rupture, describe the operator actions necessary to minimize Primary to Secondary leakage in accordance with EOP-5.0.***

References: EOP-5.0 Basis, page 62

Question:

During a Steam Generator Tube Rupture, why do we use the turbine bypass valve (TBV) instead of the atmospheric dump valves (ADVs) when steaming the ruptured S/G?

a. Minimizes the release of radioactivity through an unmonitored pathway.

b. Ensures availability of Steam Driven Aux. Feedwater P-8B.

c. Minimizes PCS shrinkage due to excess cooldown.

d. Ensures finer control over PCS temperature and pressure.

DISTRACTOR ANALYSIS

a. **CORRECT – If the Main Condenser is available, it is the preferred steaming path since any releases will be monitored on RIA-0631 Condenser Off Gas monitor.**
b. INCORRECT – Plausible if the student believes that P-8B exhaust is directed to the Main Condenser. When the TBV is used, condenser vacuum is maintained;
c. INCORRECT – Plausible due to difference in steam flow (15%) from ADVs associated with one S/G versus steam flow from TBV (4.5%).
d. INCORRECT – Plausible from same reason as 'c'.

Level of Knowledge: COMPREHENSION **Difficulty: 2**

WRITTEN QUESTION DATA SHEET

Source of Question: USED ON 2008 PALISADES EXAM

K/A: G2.4.30 - Knowledge of events related to system operation/status that must be reported to internal organizations or external agencies, such as the State, the NRC, or the transmission system operator.

Tier: 3 Group: SRO Imp: 4.1

Applicable 10CFR55 Section: 43.5, Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations. ***This exam question meets the criteria for an SRO-only question because the candidate must apply knowledge contained in 10CFR50.72, "Immediate Notification Requirements for Operating Nuclear Reactors." This question also requires the candidate to classify the event and determine the notification requirements which is SRO Only knowledge..***

Palisades Learning Objective: PL-N00109_E02 - ***State the maximum amount time allowed from the time an emergency is declared until an emergency communication must be made.***

References: EI-1, attachment 1; 10CFR50.72(a)(3)

Question:

Which one of the following requires an immediate report within one (1) hour, to the NRC in accordance with 10CFR50.72?

a. A plant shutdown is initiated in accordance with LCO 3.0.3.

b. The Reactor is automatically tripped by the Reactor Protection System.

c. A contaminated individual had to be taken off-site for medical treatment.

d. The Control Room is evacuated due to hazardous conditions.

DISTRACTOR ANALYSIS

a. Plausible because it is a required report by 50.72 but it is a 4 hour report.
b. Plausible because it is a required report by 50.72 but it is a 4 hour report.
c. Plausible because it is a required report by 50.72 but it is an 8 hour report.
d. **CORRECT - Control Room evacuation requires classification as an Alert which requires NRC notification within one hour per 50.72.**

Level of Knowledge: APPLICATION **Difficulty: 3**