October 2, 1997

NOTE TO: NRC Document Control Desk
Mail Stop 0-5-D-24

FROM: Virgil Curley, Licensing Assistant
Operating Licensing Branch, R I

SUBJECT: OPERATOR LICENSING EXAMINATION ADMINISTERED ON July 7-11, 1997, AT Millstone Unit 3, DOCKET #50-423

On July 7-11, 1997 Operator Licensing Examinations were administered at the referenced facility. Attached, you will find the following information for processing through NUDOCS and distribution to the NRC staff, including the NRC PDR:

Item #1 - a) Facility submitted outline and initial exam submittal, designated for distribution under RIDS Code A070.

b) As given operating examination, designated for distribution under RIDS Code A070.

Item #2 - Examination Report with the as given written examination attached, designated for distribution under RIDS Code IE42.

9710070031 971002
PDR ADOCK 05000423
V PDR

UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION I
475 ALLENDALE ROAD
KING OF PRUSSIA, PENNSYLVANIA 19406-1415

August 12, 1997

Mr. Bruce D. Kenyon
President and Chief Executive Officer
Northeast Nuclear Energy Company
P.O. Box 128
Waterford, Connecticut 06385

SUBJECT: MILLSTONE UNIT 3 REACTOR AND SENIOR REACTOR OPERATOR INITIAL EXAMINATION REPORT 50-423/97-04 (OL)

Dear Mr. Kenyon:

This report transmits the findings of the reactor operator (RO) and senior reactor operator (SRO) licensing examinations conducted by the NRC during the week of July 7 - 11, 1997, at the Millstone Unit 3 Nuclear Power Plant. Based on the results of the examinations, all RO and SRO applicants passed all portions of the examinations. At the conclusion of the examination, Mr. L. Briggs discussed the preliminary findings with Mr. B. Carns and Mr. J. Thayer, as well as other members of your staff.

The examination addressed areas important to public health and safety and were developed and administere[illegible] der Interim Revision 8 of the Examiner Standards (NUREG-1021). Millstone Unit [illegible] personnel developed all segments of the examination, while the NRC provided ove[illegible] and final approval prior to its administration. MS3 training personnel subsequently administered the NRC approved written examination, while the operating examination was administered by the NRC.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter and its enclosures will be placed in the NRC Public Document Room.

No reply to this letter is required, but should you have any questions regarding this examination, please contact me at 610-337-5211, or by E-mail at GWM@NRC.GOV.

Sincerely,

Paul H Bissett for

Glenn W. Meyer, Chief
Operator Licensing and
Human Performance Branch
Division of Reactor Safety

Docket No. 50-423

Enclosure: Initial Examination Report No. 50-423/97-04 (OL) w/Attachments 1 through 4

IE42

~~9708280218~~ 4pp.

<u>cc w/encl; w/o Attachments 1-4:</u>
N. S. Carns, Senior Vice President and Chief Nuclear Officer
M. H. Brothers, Vice President - Millstone, Unit 3
D. M. Goebel, Vice President, Nuclear Oversight
J. K. Thayer, Recovery Officer, Nuclear Engineering and Support
P. D. Hinnenkamp, Director, Unit Operations
F. C. Rothen, Vice President, Work Services
J. Stankiewicz, Training Recovery Manager
R. Johannes, Director - Nuclear Training
L. M. Cuoco, Esquire
J. R. Egan, Esquire
V. Juliano, Waterford Library
J. Buckingham, Department of Public Utility Control
S. B. Comley, We The People
State of Connecticut SLO Designee
D. Katz, Citizens Awareness Network (CAN)
R. Bassilakis, CAN
J. M. Block, Attorney, CAN
S. P. Luxton, Citizens Regulatory Commission (CRC)
Representative T. Concannon
E. Woollacott, Co-Chairman, NEAC

<u>cc w/encl and Attachments 1-4:</u>
H. Haynes, Director, Nuclear Training
D. Lazarony, Supervisor, Operator Training

Distribution w/encl and Attachments 1-4:
DRS Master Exam File
PUBLIC
Nuclear Safety Information Center (NSIC)
V. Curley, DRS
J. Munro, NRR/DRCH/HOLB

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D. Screnci, PAO
W. Travers, Director, SPO, NRR
J. Wiggins, DRS
L. Briggs, Chief Examiner, DRS
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A. Cerne, SRI
DRS OL Facility File
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M. Callahan, OCA
R. Correia, NRR
W. Dean, OEDO
S. Dembek, PM, SPO, NRR
G. Imbro, Deputy Director of ICAVP Oversight, SPO, NRR
D. McDonald, PM, SPO, NRR
P. McKee, Deputy Director of Licensing, SPO, NRR
L. Plisco, Chief, SPO, NRR
S. Reynolds, Technical Assistant, SPO, NRR
R. Frahm, Jr., NRR
S. Richards, OLB/NRR
V. Rooney, PM, NRR
D. Screnci, PAO
Inspection Program Branch (IPAS)
DOCDESK

U. S. NUCLEAR REGULATORY COMMISSION
REGION 1

Docket No.:	50-423
Report No.:	97-04
License No.:	NPF-49
Licensee:	Northeast Nuclear Energy Company P. O. Box 128 Waterford, CT 06385
Facility:	Millstone Nuclear Power Station, Unit 3
Location:	Waterford, Connecticut
Dates:	July 7 - 11, 1997
Chief Examiner:	L. Briggs, Senior Operations Engineer/Examiner, Region I
Examiners:	J. D'Antonio, Operations Engineer/Examiner, Region I F. Jaggar, NRC Contract Examiner, LITCO
Approved By:	Glenn W. Meyer, Chief, Operator Licensing and Human Performance Branch Division of Reactor Safety

EXECUTIVE SUMMARY

Millstone Nuclear Power Station, Unit 3
Inspection Report No. 50-423/97-04

Operations

Four Millstone Unit 3 senior reactor operator (SRO) upgrade candidates and four reactor operator (RO) candidates were administered initial licensing examinations. All candidates passed all portions of the license examination.

Overall, candidate performance during the exam was determined to be good. There were no candidate weaknesses or actions that would compromise safe plant operations observed by the examiners. The NRC examiners observed good three point communications among the operating crews. The three point communications were effective and ensured that information and directions were understood by the operating crews but appeared somewhat cumbersome during lengthy information exchanges.

Report Details

I. Operations

05 Operator Training and Qualifications

05.1 Reactor Operator and Senior Reactor Operator Initial Examinations

a. Scope

The exam was prepared by Millstone Unit 3 personnel in accordance with the guidelines in Interim Revision 8, of NUREG-1021, "Examiner Standards." The examiners administered initial operating licensing exams to four Unit 3 senior reactor operator (SRO) upgrade candidates and four Unit 3 reactor operator (RO) candidates. The written examinations were administered by the facility's training organization.

b. Observations and Findings

The results of the SRO and RO exams are summarized below:

	SRO Pass/Fail	RO Pass/Fail	Total Pass/Fail
Written	4/0	4/0	8/0
Operating	4/0	4/0	8/0
Overall	4/0	4/0	8/0

The written examinations, job performance measures (JPMs) and simulator scenarios were developed by Millstone Unit 3 representatives in accordance with Interim Revision 8 of NUREG-1021 "Examiner Standards." The exam development team was comprised of Millstone Unit 3 training and operation's representatives. All individuals signed onto a security agreement once the development of the examination commenced. The NRC subsequently reviewed and validated, along with Millstone Unit 3 personnel, all portions of the proposed examinations. Various changes and/or additions to the proposed exams were requested by the NRC following their review. Millstone Unit 3 personnel subsequently incorporated the NRC's comments and finalized the examinations.

The written exam was administered on July 7, 1997. Both the SRO and RO written examinations consisted of 100 multiple choice questions. There was one comment by the utility concerning one question on the RO written exam that had two correct answers. The NRC reviewed the utility's comment and justification and accepted the question with two correct answers. There was also a typographical error on the SRO answer key for one question which was identified and corrected during the grading of the exam.

The operating exams were conducted from July 7 - 10, 1997. The operating exams consisted of two simulator scenarios conducted twice and ten JPMs for the RO candidates and five JPMs for the SRO upgrade candidates. All JPMs were followed up with two system-related questions. All candidates were also examined using a mix of questions and JPMs to evaluate the administrative requirement portion of the exam.

Based on the grading of the written exam, the following question subject areas were missed by more than half of the RO, SRO, or a combination of both applicants for the common questions. This indicated a weakness in the general understanding of the subject area.

- Controls at the hot shutdown panel (RO)
- Auxiliary feedwater flow versus steam pressure (RO)
- Shrink and swell effects of a reactor coolant pump trip (RO)
- Loss of Bus 34A and impact on plant operations (RO)
- ATWS actions to add negative reactivity (RO & SRO)
- Operator at the controls (administrative) (RO & SRO)
- Auxiliary feedwater flow versus decay heat level (RO & SRO)
- Pressurizer pressure heater operations (SRO)

The facility should review each of the above subject areas (and several additional areas identified during the written exam grading) to correct possible weaknesses and implement programmatic changes if necessary. The facility should pay particular attention to the safety related auxiliary feedwater system for both RO and SRO candidates.

Simulator performance by the candidates was good. The examiners noted that crew briefings were routinely performed by the SROs. Three point communications, in general, were good; however, lengthy information exchanges appeared somewhat cumbersome.

Simulator deficiencies that occurred are discussed in detail in Attachment 4 of this report. The facility should place increased emphasis on maintaining simulator fidelity and problem free operation to mitigate the negative training impact on licensed operators.

In the administrative segment of the operating portion of the examination, a combination of questions and administrative job performance measures (JPMs) were used to address administrative topics. The examiners determined that the candidate performance was good.

c. Conclusions

The candidates performed well on both the written and operating exams. The SRO upgrades were issued licenses. The RO candidates will be issued licenses when NRC and utility program requirements, that cannot be completed while the plant is shutdown, are completed. The candidates appeared to be well prepared for the exams. The training department did a good job in adhering to the examiner standards and in developing the exam materials needed to administer the exams. The simulator experienced more problems than normally observed during an NRC exam.

E8 Review of UFSAR commitments

A recent discovery of a licensee operating their facility in a manner contrary to the updated final safety analysis report (UFSAR) description highlighted the need for a special focused review that compares plant practices, procedures and /or parameters to the UFSAR descriptions. While performing the preexamination activities discussed in this report, the inspectors reviewed applicable portions of the UFSAR that related to the selected examination questions or topic areas. No discrepancies were identified as a result of this review.

V. Management Meetings

X1 Exit Meeting Summary

On July 11, 1997, the examiners discussed their observations from the exam with Millstone Unit 3 operations and training management representatives. The examiners discussed candidate performance, detailed in paragraph 5.1.b above, concerning communications, preliminary written exam results, and simulator performance during the exam. The examiners also expressed their appreciation for the cooperation and assistance that was provided during both the preparation and examination week by licensed operator training personnel and operations personnel. Millstone personnel present at the exit meeting included the following partial listing in alphabetical order:

Millstone

M. Brothers, Vice President, Millstone Unit 3
B. Carns, Senior Vice President, Millstone
D. Lazarony, Supervisor Operator Training, Millstone Unit 3
R. Lueneberg, Senior Instructor, Millstone Unit 3
C. Miller, Operator Instructor, Millstone Unit 3
L. Palone, Assistant Operations Manager, Millstone Unit 3
R. Royce, Licensed Operator Initial Training Coordinator, Millstone Unit 3
J. Smith, Manager Operator Training
J. Stankiewicz, Recovery Training Manager
R. Stotts, Assistant Supervisor Operator Training, Millstone Unit 2
J. Thayer, Vice President, Nuclear Engineering and Support

NRC

Larry Briggs, Senior Operations Engineer, Chief Examiner
Joseph D'Antonio, Operations Engineer
Russell Arrighi, Resident Inspector, Millstone Unit 3

Attachments:
1. Millstone Unit 3 SRO Written Examination w/Answer Key
2. Millstone Unit 3 RO Written Examination w/Answer Key
3. NRC Resolution of Millstone Unit 3 Written Exam Comment
4. Simulation Facility Report

U. S. NUCLEAR REGULATORY COMMISSION
REGION 1

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Licensee:	Northeast Nuclear Energy Company P. O. Box 128 Waterford, CT 06385
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August 12, 1997

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Glenn W. Meyer, Chief
Operator Licensing and
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Docket No. 50-423

Enclosure: Initial Examination Report No. 50-423/97-04 (OL) w/Attachments 1 through 4

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Attachments:
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Attachment 1

Millstone Unit 3 SRO WRITTEN EXAM W/ANSWER KEY

SRO 1

PLANT CONDITIONS:

- A loss of all AC power has occurred
- SBO diesel is supplying "A" Train 4160 V buses
- Charging and letdown are secured
- Pressurizer level is 10%
- RCS subcooling is 35°F
- RCS pressure is 1600 psia
- Highest CET is 574°F
- Containment temperature is 115°F

Which of the following actions should be taken by the crew?

A. Initiate RCS cooldown by depressurizing Steam Generators.

B. Consult with the DSEC and depressurize the RCS to inject the accumulators.

C. Perform ECA-0.2, Loss of All AC Power Recovery With SI Required.

D. Consult with the DSEO and align one charging pump in the injection mode.

ANSWER:
D. Consult with the DSEO and align one charging pump in the injection mode.

REFERENCE: ECA-0.3 Caution prior to Step 6.

JUSTIFICATION: Distracters "A" and "C" are incorrect because these actions would be performed if the crew could not energize any AC Emergency Bus from the SBO Diesel at Step 7 of ECA-0.0.

Distracter "B" is incorrect because depressurizing the RCS is a strategy used past Step 7 in ECA-0.0. The operators will transition from Step 7 of ECA-0.0 to ECA-0.3 when the bus is re-energized.

Per the Caution prior to Step 6 in ECA-0.3 Distracter "D" is correct.

K/A: 062 K2.01 Loss of power to major loads

Exam Item 687

SRO 2

The plant is in MODE 6. Core offload is in progress.

The alarm circuitry for one of the Spent Fuel Pool area monitors fails. I & C is investigating. No other operator actions have been taken.

Which of the following describes the required ACTION, if any, to be taken regarding the fuel movement in progress?

A. No ACTION required, all LCOs are satisfied, fuel movement may continue.

B. Fuel movement may continue for up to 4 hours while adjusting the setpoint to within the limit.

C. Fuel movement may continue for up to 30 days.

D. Fuel movement must be suspended until an appropriate portable continuous monitor with the same Alarm Setpoint is provided in the fuel storage pool area.

ANSWER:

D. Fuel movement must be suspended until an appropriate portable continuous monitor with the same Alarm Setpoint is provided in the fuel storage pool area.

REFERENCE: Technical Specification 3.3.3.1, ACTION 28

JUSTIFICATION: A incorrect, Tech Spec Minimum channels required is 2, currently only 1.

B incorrect, the setpoint is not the problem, the 4 hour limit does not apply.

C incorrect, ACTION required is to provide the backup or suspend fuel movement.

D correct, until the backup is provided, no fuel movement can occur, then the monitor must be returned to OPERABLE status in 30 days or suspend fuel movement.

K/A: 072 K3.02 Effects on fuel handling operations

OBJECTIVE: RMS05C, RMS07C, RMS08C
New question

SRO 3

Containment purge and exhaust are in operation.

Fuel Drop monitor, 3RMS*RE41 goes into high alarm.

Which of the following describes the automatic response of the system to the alarm?

A. All four containment purge supply and exhaust valves close only.

B. All four containment purge supply and exhaust valves close and the running exhaust fan, 3HVR-FN4A or 4B stops.

C. One supply and one exhaust valve closes only.

D. One supply and one exhaust valve closes and the running exhaust fan, 3HVR-FN4A or 4B stops.

ANSWER:

C. One supply and exhaust valve closes only.

REFERENCE: LSK 22.27B

JUSTIFICATION: Purge isolation takes place if EITHER fuel drop monitor goes into alarm. (One monitor isolates the inside containment valves, the other monitor isolates the outside containment valves.) The fans do not get tripped. (C correct only)

OBJECTIVES: RMS05C; RMS07C, RMS08C

K/A: ape 061 A1.01, Automatic actuation of ARM 3.1/3.6

modified from 398 and 2117

SRO 4

Plant Conditions:

- An ATWS has occurred due to a fire in the switchgear room
- The reactor could not be tripped locally because the breakers are fused together
- Reactor power is 40% and decreasing
- Tav is increasing
- Pressure is being maintained by the PORVs cycling around 2350 psia

The PREFERRED sequence to add negative reactivity to shutdown the reactor is:

A. Perform an immediate boration, drive rods in auto or manual, initiate a safety injection

B. Drive rods in auto or manual, shift charging pump suction to the RWST, perform an immediate boration.

C. Perform an immediate boration, maximize charging flow, drive rods in auto or manual.

D. Shift charging pump suction to the RWST, drive rods in auto or manual, perform an immediate boration,.

ANSWER:

C. Perform an immediate boration, maximize charging flow, drive rods in auto or manual.

REFERENCE: FR-S.1

JUSTIFICATION: The first 4 steps of FR-S.1 contain the preferred sequence for adding negative reactivity in an ATWS situation.

OBJECTIVES: FS102C

K/A: 029 EK3.12 Actions in EOP

New question

SRO 5

The plant is in Mode 1. The following personnel are in the control room The CO who is the "Operator at the Controls", the US and the SM. The other CO is on a plant tour.

When, if at all, may the US be considered the "Operator at the Controls?"

A. If the "Operator at the Controls" must leave the red carpeted area (Operations Area).

B. If the "Operator at the Controls" is around back of the main control board acknowledging an annunciator.

C. If the Shift Manager is in the surveillance area and a proper turnover occurs.

D. The US cannot be considered the "Operator at the Controls".

ANSWER:

C. If the Shift Manager is in the surveillance area and a proper turnover occurs.

REFERENCE: COP-200.1 Section 1.7 & 1.6
COP-200.1 Attachments 5 & 6
Tech. Specs 6.2.2

JUSTIFICATION: "A" incorrect due to CO not restricted to Operations area (COP 200.1, Section 1.6)
"B" incorrect because CO allowed behind boards to acknowledge main control board annunciators and is still the "Operator at the Controls"(COP 200.1, Section 1.6)
"C" correct. With the SM in the surveillance area and with a proper relief the US can become the "Operator at the controls"
"D" incorrect. Section 1.7 of COP200.1 states 1 licensed operator in the surveillance area at all times in addition to the SM or the US in the control room. If other personnel are to be considered in the surveillance area, they shall meet the normal relief requirements.

OBJECTIVES: NAD407

K/A: 2.1.2, Operator responsibilities

Major rewrite of Question 2189

SRO 6

Safety-related plant equipment is known to have been operated in a manner which had the potential to damage the equipment.

Which of the following describes an action which must be taken in accordance with COP 200.1 CONDUCT OF OPERATIONS?

A. Shift Manager shall notify the Duty Officer and initiate a CR.

B. A notification must be made to the NRC within twenty-four hours of the event.

C. The Unit Supervisor shall initiate a priority 1 AWO for maintenance to investigate.

D. The Operations Manager shall notify the Technical Services Engineering Manager and cognizant system engineer.

ANSWER:

A. Shift Manager shall notify the Duty Officer and initiate a CR.

REFERENCE: C OP 200.1

JUSTIFICATION: If plant equipment is known to have been operated in a manner which had the potential to damage the equipment, the following actions shall be taken:

Operator shall inform the US of the problem.
US shall direct equipment or plant be placed in a safe condition.
SM shall describe the problem in the SM log.

If potentially damaged equipment is safety-related, the following additional actions shall be taken:

SM/US shall check applicability of Technical Specifications or Technical Requirements.
SM shall inform the Duty Officer.
SM shall initiate an CR.
SM shall initiate appropriate investigative action to determine status of potentially damaged equipment.

OBJECTIVES:	NAD413	
K/A:	2.1.1	3.7/3.8
	2.1.7, Evaluate performance/judgment	3.7/4.4

Question #3093

SRO 7

Which of the following is a difference between a dual verification and independent verification?

A. An Independent Verification can be performed by the same individual performing the initial verification. Dual Verification requires two individuals.

B. An Independent Verification is performed prior to the task being performed. Dual Verification is performed upon completion of the task.

C. Dual Verification can be performed using the "time and distance" method Independent Verification is usually performed concurrently with the initial verifier.

D. Dual Verification is performed concurrently with the task. Independent Verification is performed after the task or evolution has been completed.

ANSWER:

D. Dual Verification is performed concurrently with the task. Independent Verification is performed after the task or evolution has been completed.

REFERENCE: WC-6, Attachment 2

JUSTIFICATION: DUAL VERIFICATION: Dual verification is performed concurrently with the task. In general, **dual verification is performed when an action could result in an immediate threat to safe and reliable plant operation.** For dual verification, concurrently means the "performer" and the "verifier" together determine and agree the work location and component are correct for the specified action. And, the task, to the best of their knowledge, will result in the desired outcome. The person performing the task and the person performing the verification both must positively identify the component, determine the actual and required position or state, and agree on the method to be used prior to the action taking place. The verifier essentially completes the verification of the intended action, before the action is undertaken and then witnesses the task performed. It should be noted that if a specific task requires a dual verification during its performance, the need for a separate independent verification of the same evolution is not required. [v Comm 3.1]

INDEPENDENT VERIFICATION: **Independent verification is performed after a task or evolution has been completed to ensure it has been performed in the correct location, on the correct equipment, or the desired results have been obtained.** An independent verification is also performed periodically to ensure systems, equipment, components, etc. are still in the condition they were left following their last manipulation or verification. To satisfy the requirements for an independent verification, when plant conditions or the situation allows, a good rule of thumb to ensure real independence, is to apply the "time and distance" method. This requires the independent verifier to not visually observe the person who initially perform the task. Conducting the verification in this manner will alleviate the possibility that the performer goes to the wrong item or place and the independent verifier, watching the performer, simply goes to the same (wrong) location and verifies the performer's actions were completed correctly (at the wrong location!). In lieu of using the time and distance technique, as with all verifications, using attentiveness and attention to detail during their performance will prevent inadvertent problems. It is the responsibility of the verifier to ensure he or she is in the correct location, checking the required equipment or components, and determining if they meet the specified acceptance criteria.

OBJECTIVES: NAD613

K/A: 2.1.29, Conduct/verify valve lineups

Question #3268

SRO 8

When is a second Main Condensate Pump required to be operating?

A. When pumping the Main Condenser Hotwell.

B. When a Main Feed Pump is operating.

C. When the Main Circulating Water Pumps are operating.

D. Anytime steam is being released from the Steam Generators.

ANSWER:

B. When a Main Feed Pump is operating.

REFERENCE: OP 3319A section 4.1

JUSTIFICATION: Pumping the hotwell requires only one condensate pump and is only performed when the plant is in a shutdown condition and in long recycle (no MFPs operating) - ('A' incorrect)

Operation of the Circ pumps requires that tube sheet seal water be applied. This can be supplied from the CNS system during plant shutdown conditions and does not require use of the condensate pumps ('C' incorrect)

Steam may be released from the S/Gs to atmosphere with feed being supplied.

OP 3319A section 4.1 states "A minimum of two condensate pumps must be running while feeding steam generators with main feed pumps." This requires that a 2nd condensate pump be operating any time a main feed pump is operating ('B' correct)

OBJECTIVES: CNM06C (a)

K/A: 2.1.32, System limitations/precautions

Question #2964

SRO 9

The following plant conditions exist:

- The plant was operating at 100% power when a large LOCA occurs.
- A Containment Depressurization Actuation Signal (CDA) is generated due to high containment pressure.
- All safeguards equipment operates as designed except the B EDG fails to auto start and cannot be started.

15 minutes later, while performing E-0, Reactor Trip or Safety Injection, all offsite power is lost.

Approximately 2 minutes later, the RO checks power availability to the Safeguards equipment.

What should be the status of the RSS pumps?

A. All four pumps should be running.

B. Only RSS pumps "A" and "C" should be running.

C. Only RSS pump "A" should be running.

D. None of the pumps should be running.

ANSWER:

B. Only RSS pumps "A" and "C" should be running.

REFERENCE: LSKs 24-9.4A, 24-9.4B, 24-9.4Q, and 27-11J

JUSTIFICATION: The CDA signal starts the 11 minute timer to start the pumps. Once started, if an LOP occurs, the pumps are restarted by the sequencer after 60 seconds.

A is incorrect, because the "B" EDG is not running

C is incorrect. Both "A" train pumps will start. Only the "A" pump would start if in the SI/Recirc mode.

D is incorrect because the "A" train pumps will start.

OBJECTIVES: CDA06C (1); CDA07C

K/A: 103 K1.08 SIS/CDA including reset

Bank item

SRO 10

A Safety Injection has occurred and the crew was conducting a brief at the end of Step 14 of E-0 when a complete loss of Off-Site power occurs.

The Emergency Diesel Generators start, their output breakers close to restore power to the vital busses, and the sequencers complete their sequencing on of loads.

What is the status of Containment Air Recirculation (CAR) fans?

A. All the CAR fans will be running.

B. The "A" and "B" CAR fans will be running.

C. Only the "C" CAR fan will be running.

D. None of the CAR fans will be running.

ANSWER:

B. The "A" and "B" CAR fans will be running.

REFERENCES: LSK 22-27.C, LSK 24-9.4a

JUSTIFICATION: The crew has stopped the "C" CAR fan at Step 12 of E-0 ("A" and "C" wrong).

The "A" and "B" fans will start on the SI signal, and when the LOP occurs, the "A" and "B" fans will trip and will sequence back on at 39 seconds. ("B" correct, "C" and "D" wrong.).

OBJECTIVE: CVS03C (6.4)

K/A: 022 A3.01 Initiation of Safeguards

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SRO 11

During an uncontrolled rod withdraw from 175 steps on "D" bank, the final steady state actual reactor power will ________, and RCS Tave will ________. (Assume no operator action/reactor does not trip)

A. Increase, increase

B. Increase, remain the same

C. Remain the same, decrease

D. Remain the same, increase

ANSWER:

D. Remain the same, increase

REFERENCES:

JUSTIFICATION:	The p reactivity addition of the control rods will cause power and Tave to initially increase. The increasing Tav will drive power down to approximately the initial level. The end result approximately at the same power level with an elevated temperature. The temperature increase will feedback negative reactivity to offset the positive reactivity originally added by rod motion.
OBJECTIVES:	ROD06C (e)
K/A:	001K1.03, Relationship of reactivity and Rx power to rod movement 3.9

New Question

SRO 12

Given the following conditions:

- MP3 is holding at 75% power following a refueling outage.
- Rod control is in automatic.
- Rod height is 220 on Bank D.

Power range channel N-44 fails high.

WHICH ONE of the following describes the response on the rod control system?

A. Rods will drive in until a Tav-Tref error develops which will result in the rods driving out.

B. Rods will not move.

C. Rods will continuously drive in unless stopped by operator intervention.

D. Rods will drive in until the power mis-match circuit output decays away.

ANSWER:

D. Rods will drive in until the power mis-match circuit output decays away.

REFERENCES:

JUSTIFICATION: The over power rod stop will prevent outward motion (part a is incorrect).

Rod stops don't prevent inward auto motion (B is incorrect).

Temperature error and power mismatch circuit output decaying away will stop rod motion. (D is correct)

C is incorrect because the rods will eventually stop driving in without operator intervention.

OBJECTIVES: NIS07C (g)

K/A: 001 K1.05, Cause/effect between CRDS and NIS and RPS 4.5

Modified item 2207

SRO 13

During a Reactor Startup, you must ensure the reactor goes critical above the rod insertion limit. The reason that this is safety significant at this time is to ensure:

A. Peak centerline fuel melt temperatures are not exceeded if you ejected a rod during startup.

B. Sufficient positive reactivity is available by the control rods to offset power defect on the power escalation.

C. No power tilt is introduced in the core because Bank C is partially withdrawn with Bank D still on the bottom.

D. The reactor will have adequate shutdown margin for a steam line break accident.

ANSWER:

D. The reactor will have adequate shutdown margin for a steam line break accident.

REFERENCES:

JUSTIFICATION: Tech Spec Bases. The hot zero power Rod Insertion Limit is to ensure reactor DNB limits are not exceeded on large Steam Line Break and the reactor doesn't return to critical on small steam line break. (D is correct)

A is incorrect because HCP rod ejection do not cause fuel melt concerns, rod ejection is limiting at HFP.

Power defect must be offset by rods and dilution on power escalations. Normal rod height for criticality is about Bank D at 150 steps. (B is incorrect).

QPTR is not a concern less than 50%. QPTR insures FQ (LOCA) calculations are within limits. This is not a concern at hot zero power. (C is incorrect).

OBJECTIVES: ROD03C (c); ROD08C (b)

K/A: 001 K5.08, RIL Setpoint

New Question

SRO 14

PLANT CONDITIONS:

- A small break LOCA has occurred
- The crew is in ES-1.2, "Post-LOCA Cooldown and Depressurization"
- An RCS cooldown has been initiated by dumping steam to the atmosphere.

Which of the following statements describes the optimum reactor coolant pump configuration, and the basis for this configuration?

A. All RCPs should be stopped to minimize RCS inventory loss following break uncover, and prevent steam voiding in the reactor vessel on subsequent RCS depressurization.

B. One RCP should be run to produce effective heat transfer and RCS pressure control, yet minimize RCS heat input.

C. One RCP should be run to produce effective heat transfer and RCS pressure control, yet minimize RCS inventory loss.

D. Two RCPs should be run to ensure symmetric heat transfer to the intact SGs, to enhance RCS pressure control, and to prevent steam voiding in the reactor vessel head on the subsequent RCS depressurization.

ANSWER:

B. One RCP should be run to produce effective heat transfer and RCS pressure control, yet minimize RCS heat input.

REFERENCE: ES 1.2 Background

JUSTIFICATION: Forced coolant flow is the preferred mode of operation to allow for normal RCS cooldown and provide PZR spray. All but one should be stopped to minimize heat input into the RCS. "A" is incorrect because if all RCPs are stopped voiding could occur in the vessel head during the depressurization. "C" is incorrect because the RCS inventory loss is based on the existing differential pressure and not on the forced flow through the RCS. While the reasons in distracter "D" are correct, the procedure does not address running two (2) RCPs.

OBJECTIVES: S1203C

K/A: WEST 03 EA1.3, Desired results 4.1

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SRO 15

Initial Plant Conditions:

- The plant is at 100% power at EOL
- Bank D rods are at 225 steps
- All controllers are in automatic

The following MB annunciators are received:

- TAVE/TREF DEVIATION
- ROD POSITION DEVIATION
- POWER RNG CHANNEL DEVIATION
- POWER RNG FLUX RATE HI

Based on the above indications, which of the following events has occurred?

A. The contolling first stage pressure transmitter has failed high.

B. An NIS power range upper detector has failed low.

C. A rod position indication channel has failed low.

D. A control rod has dropped into the core.

ANSWER:

D. A control rod has dropped into the core.

REFERENCE: AOP 3552

JUSTIFICATION: The dropped control rod will generate the negative rate signal and the NIS power channel deviation because of the power tilt in the core. The dropped rod will cause the plant to cooldown causing the Tave/Tref deviation alarm. (D is correct)

OBJECTIVES: A5203C; ROD07C (c)

K/A: 003A2.03, Dropped rod using in-core/ex-core inst., loop temp.

New Question

SRO 16

Which of the following represents the safeguards signal that actuates the listed CVCS system valves?

	RCP Seal Return	Charging Flow Control Isolation	Letdown Isolation
A.	CIB	CIA	SI
B.	CIA	SI	SI
C.	CIA	SI	CIA
D.	CIB	CIA	CIB

ANSWER:

C. CIA SI CIA

REFERENCES:

JUSTIFICATION: The seal return MOVs close on CIA and CVCS letdown and charging flow control isolation valves close on an SI.

OBJECTIVES: ECC03C (6.1)

K/A: 006 K4.09

modified test item 2382

SRO 17

Which of the following describes how the Emergency Diesel Generators are placed in the "Speed Droop" mode of operation?

A. Automatically selected when started Manually from Main Board 8 and no safeguards signals are present.

B. Taking the mode selector switch on Main Board 8 to "Unit".

C. Taking the mode selector switch on Main Board 8 to "Parallel".

D. Automatically selected for all Emergency Diesel Starts.

ANSWER:

C. Taking the mode selector switch on Main Board 8 to "Parallel".

REFERENCE: LSK 24-9.3C

JUSTIFICATION: Speed droop is selected when the mode selector switch is in the "Parallel" position, providing no safeguard signals are present. Speed Droop allows the diesel speed to change allowing load sharing. When the diesel is the sole source of power, (i.e. in Unit) no speed droop is allowed to ensure equipment is running at full speed.

OBJECTIVES: EDG02C (v); EDG06C (f)

K/A: 064 A4.06, manual start, stop of EDG

New question

SRO 18

A large break LOCA occurred on unit 3. All systems responded as designed. When transitioning from E-0, the US directs you to perform an evaluation of the CSFs. You identify an orange path on integrity, a red path on containment and a yellow path on core cooling, a yellow path on heat sink, and a yellow path on inventory.

Based on this information, the operating crew should:

A. Transition to E-1.

B. Transition to FR-C.3.

C. Transition to FR-Z.1.

D. Transition to FR-P.1.

ANSWER:

C. Transition to FR-Z.1.

REFERENCES:

JUSTIFICATION: First transition to the highest priority critical safety function procedure which is FR-Z.1 - the only red path (correct), other answer are incorrect in accordance with EOP rules of usage.

OBJECTIVES: E0004C

K/A: 2.4.1, EOP entry control/immediate actions

Modified Question 533

SRO 19

The following plant conditions exist:

- Plant is in Mode 5 with all loops FULL
- Train "A" of RHR is in service providing shutdown cooling
- Train "B" electrical outage is in progress and is expected to last another 6 hrs.
- Due to a plane crash in the switchyard, all offsite power is lost
- The "A" EDG starts but fails to load onto Bus 34C
- The Unit Supervisor enters the appropriate procedure for loss of shutdown cooling

How should the crew re-establish shutdown cooling?

A. Transition to EOP 3501, Loss Of All AC Power (Mode 5, 6 and Zero), and restore power to be able to restart cooling.

B. Satrt and align the SBO Diesel onto Busses 34A/C and then perform Attachment B, Loss of Shutdown Coolong And/Or RCS Inventory Mode 5, of EOP 3505.

C. Initiate decay heat removal as the RCS heats up by dumping steam from at least one available SG.

D. Inject available SI Accumulators into the RCS.

ANSWER:

A. Transition to EOP 3501, Loss Of All AC Power (Mode 5, 6 and Zero), and restore power to be able to restart cooling.

REFERENCE: EOP 3505 Step 1

JUSTIFICATION: The appropriate procedure for loss of shutdown cooling in Mode 5 is EOP 3505

"A" is correct. Step 1 RNO will direct a transition to EOP 3501 if neither Bus 34C or 34D is energized.

"B" is incorrect because the SBO diesel should only be started using the guidance of EOP 3501. EOP 3501 directs the operators to attempt manual loading of the operating EDG before directing action to start/load the SBO diesel.

"C" and "D" are incorrect because they are strategies used in EOP 3501 after all attempts at energizing operable and degraded busses have been unsuccessful.

OBJECTIVES: E05605C (3)

K/A: 2.4.9, Low power operations in EOP

Exam Bank item 2935

SRO 20

The plant is operating normally at 100% po ver.

Which of the following conditions require a reactor trip followed by a trip of ALL RCPs?

A. VCT temperature increases to 140°F.

B. Isolation of a single train of RPCCW to containment

C. Seal injection flow to each pump decreases to 5.5 GPM.

D. The "A" RCP bearing oil temperature increases to 200°F.

ANSWER:

A. VCT temperature increases to 140°F.

REFERENCE: AOP 3561 Foldout Page

JUSTIFICATION: (A correct) If VCT temperature is greater than 135°F AND RCS temperature is above 400°F, then all RCPs must be stopped (RCPs are also stopped at any time if VCT is above 150°F)

(B incorrect)Both trains of RPCCW to containment must be lost to trip RCPs
(C incorrect) A reactor and pump trip is required if seal injection flow is less than 6 GPM AND thermal barrier cooling is also lost,
(D incorrect)High oil temperature only requires a trip of the affected pump

OBJECTIVE. A61761D (1)

K/A: 2.4.11, Knowledge of abnormal operating procedures

Question # 275

SRO 21

The plant tripped from 100% power. A large break LOCA has occurred. The control room has carried out E-O and is now entered E-1. A computer failure has occurred and the US is performing a manual status tree check.

The following conditions exist:

SR energized with a negative SUR
Core Exit TC's 640 degrees
Subcooling 28 degrees
RVLMS Plenum 100%
S/G NR's 4%
Total AFW flow 500 GPM
Cold leg temp decrease in last hour 20 degrees
RCS temperature 480 degrees
Containment Pressure 60 psia
Pzr level is at 0%
RVLMS upper head is at 100%

What answer best describes the sequence for dealing with the Critical Safety Functions

A. Heat Sink
Containment

B. Containment
Heat Sink

C. Heat Sink
Core Cooling
Containment

D. Containment
Heat Sink
Core Cooling

ANSWER:

A. Heat Sink
Containment

REFERENCES: CSF status Trees. OP3272 EOP Users Guide Section 1.6

JUSTIFICATION:

SR energized with a negative SUR- **Green Path Subcriticality**
Core Exit TC's 640 degrees
Subcooling 28 degrees
RVLMS Plenum 100%- **Yellow Path Core Cooling**
S/G NR's 4%
Total AFW flow 500 GPM **Red Path Heat Sink**
Cold leg temp decrease last hour 20 degrees
RCS temperature 480 degrees **Green Path Integrity**
Containment Pressure 60 psia **Red Path Containment**
Pzr level is at 0%
RVLMS upper head is at 100% **Green Path Inventory**

(A correct) Red paths are first. Heat sink is before Containment
(B incorrect) Containment is after Heat sink
(C is incorrect) Core cooling is a yellow path and is before Containment which is a Red path
(D incorrect) Heat Sink and Containment are out of order of priority

OBJECTIVES: E0004C

K/A: 2.4.21, Knowledge of status trees/logic's for CSF

New question

SRO 22

PLANT CONDITIONS:

Plant is in MODE 3
T_{ave} is 557 F
Main steam pressure is 1092 psig, controlled via turbine bypass valves
Atmospheric steam dump valves (ASDVs), 3MSS*PV20A, B, C and D are in AUTO, set at 1150 psig

What is the proper method for changing ASDV set points?

A. Place turbine bypass valve controller (MSS-PK507) in MANUAL. Place all ASDV controllers in MANUAL. Adjust each ASDV set point to the desired value, then place the ASDV in AUTO. Return MSS-PK507 to AUTO.

B. Place one ASDV controller in MANUAL. Adjust the set point to the desired value, then place the controller in AUTO. Repeat this process for each remaining ASDV.

C. Slowly dial the ASDV thumbwheels to the desired value, ensuring the set point tracks properly and the valves do not open.

D. Place MSS-PK507 in MANUAL. Ensure all ASDVs are closed, then place each ASDV in MANUAL, one at a time, and adjust set points to the desired value.

ANSWER:

B. Place one ASDV controller in MANUAL. Adjust the set point to the desired value, then place the controller in AUTO. Repeat this process for each remaining ASDV.

REFERENCES: OP 3203

JUSTIFICATION: 4.26 Place the Atmospheric Steam Dump controllers in MANUAL one valve at a time prior to making any setpoint changes, then, Return the controller to AUTO. This prevents the Atmospheric Steam Dumps from opening rapidly causing steam pressure transients

OBJECTIVES: NAD106; NAD108

K/A: 2.2.2, Local/manual operations of controllers

Question: 2131

SRO 23

While performing a new Surveillance on the Safety Injection pumps, the CO performs a step as written and notices that the Safety Injection pump does not have adequate recirculation flow.

What is the first action(s) the CO should take?

A. Determine the cause of the problem.

B. Continue with the surveillance, but consult with the Unit Supervisor.

C. Stop the surveillance and place the Safety Injection pump in a stable or safe condition.

D. Initiate a procedure modification in accordance with DC-1, Administration of Millstone Procedures and Forms.

ANSWER

C. Stop the surveillance and place the Safety Injection pump in a stable or safe condition.

REFERENCE: DC-4, Procedural Compliance, 1.9

JUSTIFICATION: 1.9 Inadequate or Unexpected Results

1.9.1 IF procedure appears to be inadequate, OR yields unexpected results while executing work activity, PERFORM the following:

a. STOP work activity.

b. IF applicable, PLACE equipment or system in stable or safe condition. (C correct)

c. CONSULT First Line Supervisor for direction. (B incorrect)

d. DETERMINE cause of problem. (A incorrect)

e. IF necessary, Refer To DC 1, "Administration of Millstone Procedures and Forms" and INITIATE modification to procedure to rectify problem

(Stephen E. Scace memo "Procedure Compliance - Management Expectations" to Millstone Station Personnel, number MP-91-801, dated October 10, 1991, states: "If you think you can't follow the written procedure, consult First Line Supervision to determine what actions (i.e. procedure changes) are necessary before proceeding. If a procedure cannot be followed as written: a. Stop the task and place the equipment or system in a safe condition. b. Change the procedure using the procedure change process. c. Proceed with the task.")

OBJECTIVES: RAD544

K/A: 2.2.12, Knowledge of Surveillance Procedures

Question : 3260 modified

SRO 24

When preparing a clearance, which of the following system/equipment conditions should be isolated from the work area by two closed valves in series?

A. A fluid system which operates at 170°F.

B. A gas system which operates at 50 psig.

C. Caustic or acid systems at any temperature or pressure.

D. Systems from confined work spaces.

ANSWER:

D. Systems from confined work spaces.

REFERENCE: WC-2

JUSTIFICATION: If practical, isolate fluid or gas systems that operate at greater than 200°F or 100 psig from the work area with two closed valves in series. Isolate systems from confined work space with two closed valves in series.

OBJECTIVES: NAD318

K/A: Generic 2.2.13 Tagging/Clearances

97 LOIT Remediation Exam

SRO 25

Given the following conditions:

- MP3 was operating at 100% power when a spurious SI occurs.
- All systems respond as designed with the exception of the "B" reactor trip breaker, which did not open and remains closed.
- The crew responds IAW EOPs, eventually transitioning to ES-1.1, "SI TERMINATION".
- The crew takes both SI reset switches to RESET.

WHICH ONE of the following describes the status of SI?

A. Both trains of SI are reset and automatic initiation blocked.

B. Neither train of SI is reset, nor is automatic initiation blocked.

C. Both trains of SI are reset, only the "A" train automatic initiation is blocked.

D. Both trains of SI are reset, only the "B" train automatic initiation is blocked.

ANSWER:

C. Both trains of SI are reset, only the "A" train automatic initiation is blocked.

REFERENCES:

JUSTIFICATION: Both trains of SI can be reset but only Train "A" is blocked because P-4 is not enabled because B reactor trip breaker is closed. ("C" is correct.)

SI cannot be blocked in Train "B" because P-4 is not present. ("A" and "D" are incorrect.)

SI can be reset in both trains. ("B" is incorrect.)

OBJECTIVES: RPS012C

K/A: 013 K4.01, SIS Reset

New Question

SRO 26

Following a Loss of Coolant Accident, adverse containment conditions exist. The following values have been recorded by the STA.

TIME	CONTAINMENT TEMP	CONTAINMENT RADIATION LEVELS
0800	185°F	5×10^4 R/HR
0815	190°F	2×10^5 R/HR
0830	180°F	2×10^5 R/HR
0845	175°F	1×10^5 R/HR
0900	170°F	9×10^4 R/HR

When, if ever, may the crew suspend the use of adverse containment values?

A. At 0845 because containment temperature has decreased below it's adverse setpoint.

B. At 0900 because both containment temperature and radiation levels are below their adverse values.

C. Adverse values can never be relaxed once they are entered if containment temperature limits were exceeded.

D. Adverse values can never be relaxed once they are entered if containment radiation limits were exceeded.

ANSWER:

D. Adverse values can never be relaxed once they are entered if containment radiation limits were exceeded.

REFERENCE: OP 3272

JUSTIFICATION: "A" is incorrect because containment radiation levels are still adverse and adverse values will always apply.

"B" is incorrect because 10^5 R/HR limits are/were exceeded.

"C" is incorrect because of containment radiation levels haven't exceeded 10^5 R/HR - containment temperature regarding adverse values can be relaxed when temperature drops < 180°F.

OBJECTIVES: E0003C

K/A: W 16 EK 1.3, Hi Rad Alarms and Actions

Modified Exam Item 3213

SRO 27

FR C-1 is entered when core exit thermocouples are greater than 1200°F. Implementation of FR-C.1 is safety significant at this time because additional operator action is required to:

A. Prevent core uncover.

B. Provide core cooling to stop the hydrogen generation due to zircaloy water reaction.

C. Limit containment pressure to less than design pressure.

D. Provide core cooling to prevent exceeding peak clad temperature limits.

ANSWER:

D. Provide core cooling to prevent exceeding peak clad temperature limits.

REFERENCES:

JUSTIFICATION: When FR-C.1 is entered the core is already uncovered. ("A" is incorrect.)

H_2 generation from zircaloy water reaction starts ~ 1800 - 2200F. ("B" is incorrect)

FR-C.1 is written for a small break LOCA with no high head injection. Car fans and spray will limit pressure. FR-C.1 established injection flow to cool core. ("C" is incorrect.)

OBJECTIVES: MC1004

K/A: 017 A2.02, Mitigating Core Damage

New Question

SRO 28

Given the following conditions:

- The unit is at 45% power with all loops in operation.
- Control Rods M12 and D4 in Bank "D" , group 2, are stuck and misaligned and will not move.
- The affected rods are trippable.
- The Bank D stuck rods indicate 156 and 162 steps respectively on DRPI while the other Bank D rods and step counters indicate 180 steps.

Assuming the rods cannot be repaired within the next week, which one of the following correctly describes the actions required by Technical Specifications for the misaligned rods?

A. Verify shutdown margin requirements are met within 2 hour.

B. Verify QPTR within 1 hour and apply LCO 3.2.4.

C. Align the remaining Bank "D" rods to ± 12 steps of the inoperable rods.

D. Be in HOT STANDBY within the next 6 hours.

ANSWER:

D. Be in HOT STANDBY within the next 6 hours.

REFERENCE: Technical Specification 3.1.3.1, AOP 3552 "Malfunction of the Rod Drive System"

JUSTIFICATION: Technical specification 3.1.3.1 outlines the actions to be taken for a single stuck but trippable control rod.
A is incorrect because shutdown margin must be satisfied in one hour not 2.
B is incorrect because QPTR requirements are not applicable when less than 50% power.
C is incorrect because multiple rods in the same group are misaligned by greater than 12 steps. Alignment is not allowed and the unit must shutdown and be in hot standby because of the multiple misaligned rods in the same group (D is correct).

K/A: 005 K3.06, Actions in EOP

OBJECTIVE: ROD08C

Modified NRC Exam Item 95 LOIT Exam

SRO 29

INITIAL CONDITIONS:

- MODE 5 with the RCS solid.
- Temperature being maintained 140°F to 150°F by the "A" RHR train.
- The plant is currently 4 days into a scheduled 8 day "B" train electrical outage, with 34B and 34D deenergized. "B" train load centers are NOT cross-tied to the "A" train.

Assuming NO operator action, which of the below statements describes the plant response to a loss of the 'A' Instrument Air Compressor?

A. RCP Thermal Barrier cooling flow from RPCCP will decrease.

B. Letdown flow will increase resulting in the RCS depressurizing.

C. RCS temperature will increase due to increased RPCCP flow through the RHR Heat Exchanger.

D. RCS temperature will decrease due to increased RHR flow through the RHR Heat Exchanger.

ANSWER:

D. RCS temperature will decrease due to increased RHR flow through the RHR Heat Exchanger.

REFERENCES: P&ID 104A, 121A, 112A, 121B

JUSTIFICATION: 'B' is incorrect because a loss of IAS will cause HCV128 to fail closed which will result in a loss of letdown and a resultant increase in RCS pressure.

'C' is incorrect because a loss of IAS will cause FV66A to fail AS IS resulting in NO change in RCS temperature from CCP flow. 'D' is correct because FCV 618 fail closed and HCV 606 fails open on a loss of IAS which will result in maximum flow through the RHR HX.

'A' is incorrect because the CCP return valves from the thermal barriers have a LOCK UP feature to prevent them from being affected by a loss if IAS. The CCP CTMT isolation valves are MOVs and therefore are not affected by a loss of IAS.

OBJECTIVES: PAS07C; RHR07C

K/A: 078 K3.02, Pneumatic control valves

96 AOP Exam

SRO 30

Given the following:

- A loss of offsite power has occurred.
- Tave is 552°F
- "Turbine Bypass Tav Interlock Bypassed" is illuminated.
- Steam dumps are in steam pressure mode of control.
- Steam dumps demand is manually INCREASED to begin a cooldown.
- The steam dumps failed to open.

Which ONE (1) of the following explains why the steam dumps will NOT open?

A. P-12, LO-LO Tave, has disarmed the steam dumps.

B. P-4, Reactor trip, has locked out the steam header pressure signal.

C. C-9, Condenser available, interlock is not met.

D. The plant trip controller has not reset.

ANSWER:

C. C-9, Condenser available, interlock is not met.

REFERENCES:

JUSTIFICATION: "C" is correct because power is not available to circ pumps on loss of offsite power. Blocking signals override arming signals.

"A" is incorrect because P-12 has been bypassed.

"B" is incorrect because P-4 locks out the load rejection controller in the Tave mode of control.

"D" is incorrect because plant trip controller was reset when you shifted to pressure mode.

OBJECTIVES: SDS06C

K/A: 051 K3.01, Steam Dump Operation - Loss of Vacuum

Exam Item: 2422

SRO 31

Which of the following conditions occurring concurrently with a large LOCA will required entry into ECA-1.1 - Loss of Emergency Coolant Recirculation?

A. Off-site power is lost and the "B" EDG did not start.

B. The power lockout relays fail to operate and all the white power lockout indicator lights are dim.

C. The A charging pump tripped on overcurrent and the B SI pump is tagged out for maintenance.

D. The "B" & "D" Recirculation spray pumps are damaged and cannot be started.

ANSWER:

B. The power lockout relays fail to operate and all the white power lockout indicator lights are dim.

REFERENCE: ES-1.3 step 2, notes prior to step 2 & 4.

JUSTIFICATION: A is incorrect. The A EDG is still available to supply A train components for cold leg recirc.

C is incorrect - One charging pump and one SI pump are still available for cold leg recirculation.

D is incorrect because the A train of RSS is still operable.

B is correct because without the power lockout operating power not available to operate some of the recirculation valves, ES-1.3 directs the operator to ECA-1.1.

OBJECTIVES: A1101C

K/A: W/E 11 K2.1, Control, function, system

Modified 2853

SRO 32

Which of the following plant conditions will cause the TD AFW pump to auto start?

A. 2/4 SG level detectors at low - low level in two SGs.

B. Safety Injection Signal.

C. Main Feedwater Isolation Signal.

D. Loss of Battery Bus 5

ANSWER:

A. 2/4 SG level detectors at low - low level in two SGs.

REFERENCES:

JUSTIFICATION: 2/4 low low signals in at least 2 SG will start TD AFW pump.

Only the motor driven AFW pumps start on SI (D is incorrect).

Main feedwater isolation will only isolate MFW but does not start any AFW pumps.

TD AFW pump will start on loss of batt Bus 1/2 not Bus 5 (D is incorrect).

OBJECTIVES: FWA04C

K/A: 059 A4.11, Auto start AFW 4.2

Modified 349

SRO 33

Assume that prior to a startup, work on the Intermediate Range nuclear instrumentation resulted in BOTH channels being OVER COMPENSATED.

Which of the following describes the expected system response to this condition?

A. During startup the Intermediate Range indication will be less than actual, and during shutdown the Source Range may not be automatically reinstated.

B. During startup the Intermediate Range indication will be greater than actual, and during shutdown the Source Range may be reinstated prematurely causing an unwanted reactor trip.

C. During startup the Intermediate Range indication will be less than actual, and during shutdown the Source Range may be reinstated prematurely causing an unwanted reactor trip.

D. During startup the intermediate range indication will be greater than actual, and during shutdown the source range may not be automatically reinstated due to the P-10 permissive being active.

ANSWER:

C. During startup the Intermediate Range indication will be less than actual, and during shutdown the Source Range may be reinstated prematurely causing an unwanted reactor trip.

REFERENCE: Funct. Diag. Sht. 3 & 4

JUSTIFICATION: A is incorrect because on shutdown the Intermediate range detectors will read lower than actual and be automatically reinstated prematurely.

B & D are incorrect because readings on startup will be lower than actual not higher.

OBJECTIVES: NIS06C (a); NIS05C

K/A: 032 A2.04 - SR/IR overlap

Bank Item 2217

SRO 34

PLANT CONDITIONS:

- 100% power
- All systems in AUTOMATIC
- LT-459 selected for control of Pressurizer level control selected to LT-459
- PT-456 selected for control of Pressurizer Pressure
- Instrument for "A" and "C" steam generators selected to Channel I
- Instruments for "B" and "D" steam generators selected to Channel II

A loss of VIAC-2 occurs.

Which of the following lists controllers which should be taken to MANUAL as a result of the VIAC-2 failure?

A. Rod control
Pressurizer Pressure
Pressurizer Level

B. Rod Control
Pressurizer Pressure
Master main feed pump controller

C. Pressurizer Pressure
Pressurizer Level
Feed Regulating Valves for "A" & "C" SGs

D. Pressurizer Pressure
Master main feed pump controller
Feed Regulating Valves for "B" & "D" SGs

ANSWER:

D. Pressurizer Pressure
Master main feed pump controller
Feed Regulating Valves for "B" & "D" SGs

REFERENCE: AOP 3564, Process sheets 10 and 11, 25

JUSTIFICATION:

Pressurizer level controller will be affected because the backup channel will result in a letdown isolation

Pressurizer pressure is affected because its controlling channel is channel II

Main feed pump speed control is affected due to loss of two steam flow channels, low.

Rod Control is not affected

The Feed Regulating Valves on only the "B" and "D" SGs will be affected.

Only D correct.

OBJECTIVES: 12005C

K/A: ape 057 A1.06, Manual control of components

New question

SRO 35

The plant is operating at 100% power. The steam dumps are in the Tave mode of control and rods are in AUTO.

3MSS*PT507, Main Steam Header Pressure, fails high. This will:

A. Open the steam dump cooldown valves, and the TDFWPs speed will decrease.

B. Block the load rejection controller from arming the steam dumps, and the TDFWPs speed will increase.

C. Arm the steam dumps but they won't open, and the TDFWPs speed will decrease.

D. Have no effect on steam dumps, and the TDFWPs speed will increase.

ANSWER:

D. Have no effect on steam dumps, and the TDFWPs speed will increase.

REFERENCES:

JUSTIFICATION: PT 506, not PT 507, feeds the load rejection controller and arms the steam dumps (B & C are incorrect).

The PT 507 effect on the cooldown valves function is only available in the pressure mode of control (A is incorrect).

PT 507 is only in effect when in the pressure mode of control (D is correct)

PT 507 feeds the TDFWP speed control circuitry. An increase in pressure will cause the pumps speed to increase.

OBJECTIVES: SDS07C

K/A: 039 A2.04

Modified 2426

SRO 36

INITIAL CONDITIONS:

- The Unit is operating at 48% power with the "NIS POWER RANGE P-9 PERMISSIVE" Blue Light NOT Lit.
- Due to an instrument failure, actual level in the "B" S/G level has increased to 80% resulting in an automatic FWI actuation.
- All equipment operated as designed.

Assuming no actions are taken in the instrument rack room, which of the following must occur to allow resetting the FWI signal from the main boards?

A. Clear P-14 and Reset P-4

B. Clear P-14 and P-9

C. Clear P-14 only

D. Reset P-4 only

ANSWER:

A. Clear P-14 and Reset P-4

REFERENCE: Functional Sheet 13

JUSTIFICATION: Permissive is lit below P-9, if level reaches the turbine trip setpoint, the reactor will trip, therefore to reset the FWI, both P-4 and P-14 will have to clear.

OBJECTIVES: NIS04C (b.4)

K/A: 059 A 4.11 Permissives

Question 1520
96 Quiz 6

SRO 37

Initial Plant Conditions

- Plant is at 60% power
- Rod control is in manual
- Steam Dumps are in Tave mode of control

A turbine trip occurs. The turbine trip fails to cause a reactor trip and actuate the steam dumps on the turbine trip controller.

The next automatic reactor trip signal to be generated for this transient would be:

A. High Pressurizer Level Trip

B. OTΔT

C. High Pressurizer Pressure Trip

D. OPΔT

ANSWER:

C. High Pressurizer Pressure Trip

REFERENCES:

JUSTIFICATION: The power mismatch will cause a rapid increase in pressurizer pressure causing the reactor to trip (C is correct).

Pressurizer level will backup the high pressure trip (A is incorrect)

B is incorrect. The temperature increase will drive power down and pressure up. Both of these factors are benefits with regard to OTΔT.

D is incorrect. Power will decrease during the transient. The margin to the OPΔT trip will be increasing.

OBJECTIVES: A5002C; MC0302

K/A: 045 A1.05 RCS following turbine trip

New Question

SRO 38

Plant History:

- A loss of off-site power occurred 10 minutes ago.
- The crew has stabilized the plant and just completed ES-0.1.
- RCS temperature has stabilized at 557°F
- Steam Generator levels are between 25 - 30% in the narrow range.
- Total AFW flow to the steam generators is 535 GPM.

The Operations Manager has directed the plant be maintained at the current plant conditions for RCS temperature and steam generator levels.

One day from now the total AFW flow should be approximately:?

A. The same.

B. 110 - 140 GPM.

C. 250 - 275 GPM.

D. 400 - 425 GPM.

ANSWER:

B. 100 - 125 GPM.

REFERENCE:

JUSTIFICATION: One minute after a reactor trip the decay heat level is approximately 3-4%. After one hour it is approximately 1.5-2% and after one day it is 0.7-1%. Consequently after one day the existing decay heat is approximately one quarter of its value following the trip. Therefore since the AFW system is maintaining SG level, the flow will reduce by one quarter to approximately 134 GPM.

OBJECTIVES: S0103C

K/A: W/E 9 EK2 2 Relationship between emergency feedwater flow to S/G and decay heat removal for facility heat removal following a trip.

New Question

SRO 39

PLANT CONDITIONS:

- Reactor is operating at 100% rated thermal power
- Annunciator 5-3 on MB4C "PR UP DET HI FLUX DEV/AUTO DEFEAT" has alarmed
- All control rods are positioned within 12 steps of their group demand counters
- Maximum QPTR based on plant computer program 3R5 is 1.04

Assuming QPTR is not reduced, within two hours reactor power must be reduced to ________, and NIS overpower trips reduced to ______ within the next four hours.

A. 50%, 59%

B. 50%, 55%

C. 88%, 97%

D. 94%, 103%

ANSWER:

C. 88%, 97%

REFERENCE: Tech. Spec. 3.2.4 Action Statement c.2; OP 3273

JUSTIFICATION: If QPTR is greater than 1.02 but less than 1.09 then within 2 hrs reduce thermal power 3% of rated power for every 1% greater than 1.0 and similarly reduce the overpower trip setpoints within the next four hours.

OBJECTIVES: NIS08C (b)

K/A: 015 A1.04, NIS/QPTR

Modified 1058

SRO 40

Given the following conditions:

- The unit is critical
- The crew is holding power at 1.0 x 10-8 amps.
- SG levels are being controlled on the bypasses, in automatic.

N-36 control power fuse blows.

WHICH ONE of the following describes the plant's response?

A. An IR high flux rod stop will be received and the reactor will remain critical.

B. The reactor will remain critical with no rod stops.

C. The reactor will trip on SR high flux when they automatically energize.

D. The reactor will trip on IR high flux.

ANSWER:

D. The reactor will trip on IR high flux.

REFERENCES:

JUSTIFICATION: Loss of control power fuses causes a trip signal to be sent to RPS through the Reactor Protection System and will also generate a rod stop and reactor trip on 1/2 coincidence (d is correct)

A is incorrect because the trip will cause the reactor to go subcritical. (This also makes b incorrect).

C is incorrect as source ranges will not automatically energize.

OBJECTIVES: NIS07C

K/A: 015 K2.01, NIS channels, power supplies 3.3

Modified item 2269

SRO 41

The following conditions exist:

- Crew is in EOP 3503, Shutdown Outside The Control Room
- Control room is filling with dense smoke
- Control room is ordered evacuated
- The reactor is tripped from 100% power
- The turbine is tripped

SI occurs after trip due to the steam dumps malfunctioning

Which of the following describes the procedural flow path under these conditions·

A. Complete EOP 3503 and then enter E-0.

B. Exit EOP 3503 and enter E-0.

C. Perform E-0 in parallel with EOP 3503.

D. Complete EOP 3503, then perform cooldown in accordance with EOP 3504.

ANSWER:

D. Complete EOP 3503, then perform cooldown in accordance with EOP 3504.

REFERENCES:

JUSTIFICATION: EOP rules of usage - if SI or Rx Trip occurs in EOP 3503, you should remain in EOP 3503.

OBJECTIVES: EOU (1733)

K/A: 067 K3.04 Actions in EOPs

New Question

SRO 42

Given the following conditions:

- Unit is at 100% power
- Pressurizer level control is selected to 459/461
- VCT makeup control is in Automatic

A reference leg leak occurs in pressurizer level transmitter LT-459.

Assume no other operator action, which of the following occurs:

A. Letdown isolation occurs
3CH-FCV-121 ramps open
Auto makeup occurs
Unit will eventually trip on high pressurizer level.

B. 3CH-FCV-121 ramps closed
VCT diverts
Letdown isolation occurs
Pressurizer level will begin to increase
Unit will eventually trip on high pressurizer level.

C. Letdown isolation occurs
3CH-FCV-121 ramp open
Auto make-up occurs
VCT swaps over to RWST
Plant cools down
Unit trips on low pressurizer pressure because heaters are de-energized.

D. 3CH-FCV-121 ramps closed
VCT diverts
Letdown isolation occurs
Pressurizer level will continue to decrease due to seal leakoff.
Unit trips on low pressurizer pressure

ANSWER:

B. 3CH-FCV-121 ramps closed
VCT diverts
Letdown isolation occurs
Pressurizer level will begin to increase
Unit will eventually trip on high pressurizer level.

REFERENCES:

JUSTIFICATION: Reference leg failure will cause indicated level to fall high this causes FCV-121 to ramp close. Thus A & C are incorrect).

Act pressurizer level will decrease and the renaming channels will cause letdown isolation. Seal injection will fill pressurizer and cause eventually a high level trip (B is correct).

D is inconect because seal injection still occurs even if the FCV-121 is closed and level will begin to increase.

OBJECTIVES: A5503C; PPL06C; PPL07C

K/A: 011 K3.01, Loss of PZR Level effect on CVCS.

Modified 375

SRO 43

Given the following conditions:

- The unit is at 8% power.
- Plant startup is in progress
- Pzr level instrument LT-459 has failed LOW.
- All actions of AOP 3571 "Instrument Failure" Attachment C are complete.

Which of the following describes the course of action the crew should take if a subsequent failure of Pzr level instrument LT-460 HIGH?

A. Verify reactor trip.

B. Stop the startup, and restore one of the failed channels of pressurizer level to OPERABLE status prior to increasing power above 10%.

C. Stop the startup, and restore both of the failed channe's of pressurizer level to OPERABLE status prior to increasing power above 10%.

D. Within one hour initiate ACTION to be in at least HOT STANDBY within the next 6 hours.

ANSWER:

B. Stop the startup, and restore ONE of the failed channels of pressurizer level to OPERABLE status prior to increasing power above 10%.

REFERENCE: AOP 3571 "Instrument Failure" Attachment C, Pzr Level and Pressure Control Lesson Plan, Technical specification 3.3.1 and functional sheet 11

JUSTIFICATION: With all actions of the AOP complete, the bistable associated with the high Pzr level Rx. trip has been placed in a tripped condition

When the second channel fails high, the coincidence for a high pressurizer level reactor trip is met, however, the trip is blocked less than 10%. (A incorrect)

Technical specifications require 2 channels to be OPERABLE, however, this is required below P-7 (10%), and to increase above 10%, the bistables must be tripped within 6 hours, B correct, D incorrect.

It is not required to have both channels OPERABLE to increase above 10%, (C incorrect)

OBJECTIVES: PPL07C

K/A: 028 A1.01, Pressurizer level bistables

modified from 1995 MP3 NRC exam

SRO 44

Given the following conditions:

- The reactor is tripped.
- A Loss of Offsite Power has occurred
- Safety Injection is actuated from a small LOCA.
- All ECCS equipment is operating as expected.
- Pressurizer level is 48% and increasing on all channels.
- RCS pressure is 1700 psia and decreasing slowly on all channels.

Which one of the following describes a leak location that is consistent with the indications given?

A. A leaking pressurizer safety valve.

B. The letdown line relief valve lifting

C. A reference leg break on pressurizer level instrumentation.

D. A failed open spray valve.

ANSWER:

A. A leaking pressurizer safety valve.

REFERENCES:

JUSTIFICATION: A PORV or safety valve failing open will cause pressurizer level to increase and pressure to decrease on all channels. ("A" is correct.)

"B" is incorrect because this break location will be isolated by the SI/CIA signal.

"C" is incorrect. On the effected reference leg the indicated level channel would increase and pressure would decrease. However, on the non-affected channel level will decrease as well as pressurizer pressure.

"D" is incorrect - a failed open spray valve will only cause pressurizer pressure to decrease.

OBJECTIVES: A5503C

K/A: 008 A1.01, Operation Monitoring Instrumentation from for PORV, sprays

New question

SRO 45

During performance of the shift control room rounds, the Control Operator discovers that 3HVQ-RE49, ESF Building Normal Ventilation Monitor, indicates OFF-LINE for both Data-A and Data-B at the RMS Console.

Which of the below statements describes the operating status of the radiation monitor?

A. The radiation monitor may be considered operational once its data and operation is verified at the Local Indicating Control panel (LIC).

B. The radiation monitor must be considered inoperable.

C. The radiation monitor continues to indicate properly at the RMS Console but all radiation monitor control functions must be performed manually at the RMS Console.

D. The radiation monitor may still be considered operational because it will still notify control room staff of high radiation conditions by actuating the "RADIATION ALERT" and "RAD HI" annunciators on MB2.

ANSWER:

A. The radiation monitor may be considered operational once its data and operatior is verified at the Local Indicating Control panel (LIC).

REFERENCES: P&ID 152A, RMS073T and RMS073C Handouts, PIR 391-043, MP3 Memo MP-3-0-385 dated 3/25/91, Kaman Instrumentation Operation - Maintenance Manual volumes 1 thru 3 and RMS Console Help display.

JUSTIFICATION: 'B' is incorrect because each local unit is completely self contained, requiring the computer room computer ONLY for transmitting data to the control room.

'C' is incorrect because if off-line from both data-A and data-B, all communications between control and the RMU is terminated.

'D' is incorrect because the alarms at MB2 are a function of the computer. If the RMU computer is not communicating with the computer room computer, it can not cause the MB2 alarms to actuate.

'A' is correct because although not communicating with the control room, each RMU is completely a stand alone unit and is designed to function without the control room computer. Once the data and operation has been verified correct for the RMU at its LIC, the unit may be considered operational per the SS (Memo MP-3-0-385 and PIR 391-043 and RMS073 handouts)

OBJECTIVES: RMS08C

K/A: 073 A4.02, RMS Control Panels/Indications

Exam Item 2407

SRO 46

Given the following:

- A twenty five (25) year old Maintenance Contractor with complete exposure records has the following exposure record for the current calendar year:

• Shallow Dose Equivalent -	2.55 REM
• Committed Dose Equivalent -	0.75 REM
• Deep Dose Equivalent -	2.13 REM
• Lens Dose Equivalent -	3.08 REM
• Committed Effective Dose Equivalent -	1.95 REM

WHICH ONE (1) of the following is this individuals Total Effective Dose Equivalent (TEDE) for the current calendar year?

A. 2.88 REM

B. 4.08 REM

C. 5.21 REM

D. 5.43 REM

ANSWER:

B. 4.08 REM

REFERENCE: RPM 1.3.1
Get RAD Worker Training

JUSTIFICATION: TEDE = CEDE + DDE = 1.95 REM + 2.13 REM = 4.08

OBJECTIVES: GET Radworker training

K/A: 2.3.1 , 10CFR20 Radiation Limits

New Question

SRO 47

The rad waste PEO is dispatched to change LWS-FLT3. This PEO has not performed this task before. The HP technician informs the PEO that the dose rate on the outside of the filter housing is 1 R/hr.

Which one of the following is not an example of ALARA techniques for reducing exposure for filter replacement.

A Long handled tools to remove the old filter -

B Place the filter in a shielded drum upon removal to reduce exposure

C. Have the rad waste PEO be assisted by the Turbine Building PEO who has done the task several times before.

D. Have the new PEO perform the filter replacement on a mockup first.

ANSWER:

C. Have the rad waste PEO be assisted by the Turbine Building PEO who has done the task several times before.

REFERENCE: RPM 5.2.4 Section 1.2,1.3,1.4

JUSTIFICATION: RPM 5.4.2 list 3 main areas to reduce Radiation exposure. Time Distance and Shielding
A is Distance
B is Shielding
D is Reduced Time by practice on a mockup prior to the job.

RPM 5.2.3 Section 1.1 states that individual exposures within a work group are balanced consistent with experience.
C will not balance the exposure if the experienced person always does the job.

OBJECTIVES: NAD721; NAD722; NAD723

K/A: 2.3.10, ALARA - procedures to reduce radiation exposure

New Question

SRO 48

Given the following conditions:

- MP3 is at 35% power.
- "RCP B STANDPIPE HI LEVEL" has lit.
- "B" seal injection flow is 8.2 GPM.
- "B" seal leak-off flow is 0.2 GPM.
- Seal return temperature is 150°F and rising steadily.
- Pump radial bearing - rising slowly @ 145°F

Based on the above indications, the operating crew should:

A. Trip the unit, secure "B" RCP and close its No. 1 seal leakoff valve within 2 minutes.

B. Trip the "B" RCP and close its No. 1 seal leakoff valve after the pump has been stopped for five minutes.

C. Close the "B" RCP's No. 1 seal leakoff valve within 5 minutes and shutdown the unit within the next 30 minutes then secure the "B" RCP.

D. Trip the "B" RCP and close its No 1 seal leakoff after the pump has been tripped for two minutes.

ANSWER:

D. Trip the "B" RCP and close its No 1 seal leakoff after the pump has been tripped for two minutes.

REFERENCE: OP 3554

JUSTIFICATION: Since power less than P-8, the RCP can be stopped without tripping the unit (a is incorrect)

0P 3554 requires tripping RCP and closing the seal leakoff valve within two minutes (D is correct) The RCP must be removed from service within 5 minutes of failure (not within 5 minutes of closing seal leakoff valve). (B and C are incorrect)

OBJECTIVES: A5403C

K/A: 015 A2.01, Cause of RCP failure

Modified question 1104

SRO 49

WHICH ONE of the following interlocks must be satisfied to start an RCP?

A. RCP #1 seal ΔP must be greater than 200 psid.

B. The overcurrent trip selector switches must be in the cold position.

C. Cold leg and hot leg isolation valves must be open.

D. Cold leg isolation valve must be open and the loop bypass and hot leg isolation valve must be closed.

ANSWER:

C. Cold leg and hot leg isolation valves must be open.

REFERENCES: RCP text; OP 3301B

JUSTIFICATION: A and B are incorrect because they are procedure administrative requirements but are not part of the interlock circuitry

C is correct.

D is incorrect. The cold leg stop valve must be closed with the bypass fully open to satisfy the RCP interlock.

OBJECTIVES: RCS04C

K/A: 003 K6.14, RCP starting requirements 2.9

Modified question 2159

SRO 50

A small break Loss of Coolant Accident has occurred.

The current plant conditions exist at the completion of E-0 step 14:

- SI has occurred.
- All SI equipment started.
- Containment Temperature is 185°F.
- RCS pressure is 1800 psia and stable.
- CET's are 520°F.
- Pressurizer level is 50% and slowly increasing.

Assuming conditions do not significantly change, you would expect to stop one charging pump in:

A. E-0 Reactor Trip on Safety Injection

B. ES-1.1 - SI termination.

C. ES-1.2 - Post LOCA Cooldown and Depressurization

D. ES - 1.3 - Transfer to Cold Leg Recirculation

ANSWER:

C. ES-1.2 - Post LOCA Cooldown and Depressurization

REFERENCES:

JUSTIFICATION: A is incorrect because adequate subcooling doesn't exist and RCS pressure is less than 1950 psia (adverse containment) to stop a charging pump in E-0.

B is incorrect because adequate subcooling doesn't exist to make the transition to ES-1.1.

C is correct because a 100°F/hr cooldown will be started which will increase subcooling major to allow stopping a charging pump in ES-1.2.

D is incorrect because charging pump will be stopped in ES-1.2, and cooldown will place plant on RHR, ES-1.3 will not be entered for a small break LOCA.

OBJECTIVES: S1203C

K/A: E02 EK2.1, SI Termination

Modified Exam Item 1533

SRO 51

Given the following conditions:

- The Uni: was operating at 75% power.
- A small break LOCA occurred in coincidence with a loss of off-site power.
- RVL!S indicates that a void exists in the reactor vessel head.
- The cooldown was stopped and RCS pressure raised to regain subcooling margin. Increasing RCS pressure will _______ the size of the void and _____ the leakage from the RCS.

A. Increase increase

B. Decrease; increase

C. Increase; decrease

D. Decrease; decrease

ANSWER:

B. Decrease; increase

REFERENCE:

JUSTIFICATION: Raising the pressure will decrease the size of the void but increase the leakage for the RCS. ("B" is correct)

OBJECTIVES: MC0703 (c)

K/A: 009 K3.06, Inventory Balance During Small Break Loss.

New Question

SRO 52

PLANT CONDITIONS:

- Plant is in Mode 6
- No fuel movements are in progress
- "A" Train Electrical outage is in progress
- Computer is available
- "B" Spent Fuel Pool cooling pump caught fire and tripped 1 hour ago
- Spent Fuel temperature - 115°F and slowly increasing
- Spent Fuel Pool level - 37% and decreasing slowly
- Reactor Cavity seal - intact
- Fuel Building Monitoring Group Histogram - NORMAL
- RWST level - 1,000,000 gallons

For the existing plant conditions, which one of the following corrective actions should be taken?

A. Align RWST to gravity feed the spent fuel pool.

B. Establish emergency makeup using the fire water system.

C. Supply makeup to the Spent Fuel Pool from the Primary Grade Water System

D. Establish emergency makeup to the Spent Fuel Pool from Service Water.

ANSWER:

A. Align RWST to gravity feed the spent fuel pool.

REFERENCE: EOP 3505A, Att. A, Step 3

JUSTIFICATION: Gravity feed is the preferred method to the spent fuel pool. (A is correct)

Emergency makeup using the fire water system is only used if the additional attempt of emergency makeup from the RWST is attempted after the gravity feed method does not work (B incorrect).

C and D are least preferred and are only done if RWST is not available (C & D are incorrect).

OBJECTIVES: E0503C; E05A3C

K/A: 033 A2.02, Loss of Spent Fuel Cooling EOP actions

Exam Item 1295

SRO 53

The following events have occurred:

- A SGTR has occurred subsequent to a steam break inside containment.
- The crew has transitioned to E-3, "SGTR", from E-2, "Faulted Steam Generator Isolation."
- They have identified and isolated the ruptured Steam Generator, which is not faulted, and are preparing to initiate RCS cooldown.

Current Plant Conditions:

- Containment temperature is 185°F
- RCS pressure is 1420 psia
- Ruptured Steam Generator pressure is 895 psig
- Intact Steam Generator pressures are 850 psig
- Faulted Steam Generator is 600 psig
- Core exit temperature is 485°F

Using the provided reference, determine the required core exit temperature to be achieved by the RCS cooldown, if necessary.

A. A cooldown is not necessary, core exit temperature is already less than the required temperature.

B. 413°F

C. 434°F

D. 476°F

ANSWER:

C. 434°F

PROVIDE ATTACHMENT TO STUDENTS

REFERENCE: E-3 Step 14a graph (Adverse CTMT parameters to be used)

JUSTIFICATION: CTMT 185°F Adverse parameters. Step states not to interpolate, therefore, 885 psig on graph to be used. A incorrect since RCS is above the required temperature.
476°F = non adverse number for 850 psig (D incorrect).

OBJECTIVES: E3003C

K/A: 038 A1.34, Cooldown to specific temperature

97 EOP Exam
681

SRO 54

PLANT CONDITIONS

The unit is running down in power from 100% to take the unit off-line.

Loop 3 Tave fails unobserved to a constant output of 572°F.

Which one of the following describes where pressurizer level will stabilize under these plant conditions? Assume no operator action taken.

Pressurizer level will stabilize at:

A. 22%

B. 28%

C. 45%

D. 89%

ANSWER:

C. 45%

REFERENCES:

JUSTIFICATION: Pressurizer level is programmed with auctioneered high Tav. Pressurizer level will decrease, then control at program level for 572°F which is 45%. (C is correct).

B is incorrect because pressurizer level will not be decreased no load value.

A is incorrect because pressurizer level will not decrease to cause let down isolation.

D is incorrect as pressurizer level will not increase above program value.

OBJECTIVES: PPL07C

K/A: 0094 A1.02, CVCS/Tav/Pressurizer level

SRO 55

Which of the following signals will cause 3LWS-HV77, Waste to Discharge Tunnel Stop valve to CLOSE?

A. High radiation, only.

B. High radiation and Low flow from the Circ Water system.

C. High radiation and High flow through the monitor.

D. High radiation and Low flow through the monitor.

ANSWER:

A. High radiation, only.

REFERENCE: P & ID 106A; LWS 068T Text

JUSTIFICATION: The stop valve closes if radiation reaches 2x the setpoint, only.

OBJECTIVES: WFB05C

K/A: ape 059 K3.01, Termination of release

modified from exam bank item 3429

SRO 56

PLANT CONDITIONS:

- Plant startup in progress
- Reactor power is 42%
- Turbine load is 425 MWe

For the current plant conditions which of the following would immediately start the Motor Driven AFW pumps?

A. • Safety Injection
 • Low-Low level in one steam generator

B. • Safety Injection
 • AMSAC

C. • Loss of DC Bus 1
 •AMSAC

D. • Loss of DC Bus 1
 • Low-Low level in one steam generator

ANSWER:

A. • Safety Injection
 • Low-Low level in one steam generator

REFERENCE: Functional Sheet 15

JUSTIFICATION: The MDAFW pumps start on SI or Low -Low level in 1 of 4 SGs

Based on the given conditions, turbine load less than 40%, AMSAC will not be armed. Therefore B and C are not correct.

Loss of DC Bus 1 will result in the TDAFW pump starting but not the MDAFW pumps.

OBJECTIVES: FWA04C

K/A: 061 K4.02, auto start of Aux. feed

modified from 2815

SRO 57

With the plant at 100% power, the "B" RPCCW pump trips.

The crew enters AOP 3561 LOSS OF REACTOR PLANT COMPONENT COOLING WATER.

The crew immediately OPENS the RPCCW CTMT header cross-connect valves 3CCP*AOV179A, B, C, and D; and CLOSES the RPCCW CTMT Supply and Return header isolation valves for the "B" train.

If no further operator action is taken, how long can the plant remain at full power and satisfy Technical Specifications?

A. 1 hour

B. 6 hours

C. 72 hours

D. Indefinitely

ANSWER:

C. 72 hours

REFERENCE: Technical Specification 3.7.3

JUSTIFICATION: As long as only the "B" pump is running, only one loop is OPERABLE, both are required, the crew has 72 hours to restore to two OPERABLE loops. (C correct)

OBJECTIVES: CCP08C

K/A: 008 A2.01, Loss of RPCCW pump

MODIFIED FROM 3437

SRO 58

The crew is attempting to determine the location of a small leak, approximately 10 GPM, in the charging system. All systems are in automatic.

30 minutes after the leak begins, the PEO reports an indication of leakage from 3CHS*MV8438C, Charging Pump A/C Discharge Isolation valve.

What would be a positive indication the operator would have on the main board that 3CHS*MV8438C is the source of the leak?

A. VCT level decreasing with letdown flow constant.

B. No change in indicated charging flow from its initial value.

C. VCT level decreasing with increased seal injection flow.

D. Increased charging flow indicated and pressurizer level decreasing.

ANSWER:

B. No change in indicated charging flow from its initial value.

REFERENCE: AOP 3555
P&ID 104A
CVCS Text

JUSTIFICATION: A leak downstream of CHS*FCV121 would INITIALLY result in a decrease in pressurizer level. FCV 121 will open to restore level and would remain at this amount to maintain level.

A leak upstream of FCV121 would have the same initial decrease in pressurizer level, however, the increase in charging flow to restore pressurizer level would only be indicated as the original value because the leak is not "seen" by the flow indicator FT-121. (B correct)

C incorrect because a leak at 3CHS*MV8438C would not cause seal injection flow to increase

D incorrect, charging flow will not be increased and pressurizer level will be constant.

A is incorrect. VCT level decreasing with letdown flow constant only confirms that the leak is not in the letdown system. It does not provide any information relative to the location of the leak in the charging system.

OBJECTIVES: A5501C

K/A: 004 A3.11, Auto ops. charging/letdown 3.4/3.1

modified from 2581

SRO 59

The plant is operating at 100% power, the following leak rate data exists:

- Total RCS leakage, 10.0 GPM
- Known leakage is as follows:
- Secondary leakage:
  - A steam generator, 0.25 GPM
  - B and C, none detected
  - D steam generator, 0.33 GPM
- Leakage into the PRT, 4.0 GPM
- Leakage into the Primary Drain Transfer Tank, 4.5 GPM
- Leakage into the Containment Drain Transfer Tank, 0.5 GPM

Which of the following RCS Leakage Technical Specifications, if any, have been exceeded?

A. Identified.

B. Unidentified.

C. Reactor to Secondary.

D. None, all leakage is within Technical Specifications.

ANSWER:

D. None, all leakage is within Technical Specifications.

REFERENCES:	Tech Spec 3.4.6.2 and T.S. definitions for leakage.
JUSTIFICATION:	There is 0.58 GPM leakage total to the SGs, and less than 500 gpd thru any 1 SG (about 0.35 GPM), C incorrect, Identified leakage is 9.73 GPM (4 + 4.5 + 0.5 + 0.58), TS requires greater than 10, A incorrect. Total leakage is 10.0 therefore Unidentified is 0.42, less than Spec of 1.0 B incorrect, D correct.
OBJECTIVES:	RCS09C (b); A5502C
K/A:	2.1.33, Recognize entry into LCO

modified from 652

SRO 60

The following plant conditions exist:

- The plant was operating at 100% power when a large LOCA occurs.
- The crew is preparing to swap over to cold leg recirculation
- The "A" and "B" RSS pumps failed to start and are currently considered out of service.
- The crew has elected to use the "C" RSS pump for cold leg recirculation.

Which of the following actions minimizes the potential for overheating the "C" RSS pump?

A. The flowpath from the "C" RSS pump to the containment spray header will not be isolated until a flowpath to the Charging/SI pumps is established.

B. The "A" Train Sequencer will be placed in "Test 2" mode, with the "C" RSS pump "Test/Inhibit" switch in "Inhibit".

C. The "C" RSS pump is equipped with a motor operated recirculation valve that will return a minimum amount of flow back to the containment sump.

D. The "B" train charging and SI pumps are aligned in the Injection Mode.

ANSWER:

A. The flowpath from the "C" RSS pump to the containment spray header will not be isolated until a flowpath to the Charging/SI pumps is established.

REFERENCE: ES-1.3, "Transfer to Cold Leg Recirculation": step 2.f and Att. C
P&ID EM-112C

JUSTIFICATION: Step 2.f RNO directs the operator to align the "C" RSS pump using Att. C of ES-1.3 only if the "A" and "B" RSS pumps are unavailable. This means the "B" train flowpath will not be realigned for Cold Leg Recirculation. However, distracter D is incorrect because the "C" RSS pump does not provide flow to the "B" train flowpath until after flow is established to train "A". (See ES-1.3, Att.C steps 3-9). This means the pump would heat up unless flow through the pump is assured as soon as flow is established in train "A".

C is incorrect bec ause the "C" RSS pump does not have a mini-flow recirculation valve/line.

B is incorrect because the sequencer lineup has nothing to do with protecting the pump from overheating. The steps ensure the pump restarts on an LOP

The lineup the operators follow specifically ensures that the flowpath to the charging/SI pumps is established prior to isolating flow to the containment spray header. (A is correct)

OBJECTIVES: CDA05C

K/A: 026 A2.04, Failure of spray pump

modified from 2473

SRO 61

Which of the following action(s) is necessary to transfer control of the steam generator atmospheric dump valves to the Auxiliary Shutdown Panel and enable valve position indication?

A. Select "LOCAL" on the controller selector switch to both transfer control and enable valve position indication.

B. Select "LOCAL" on the push button below each controller to transfer control and select "LOCAL" on the controller selector switch to enable valve position indication.

C. Select "LOCAL" on the push button below each controller to transfer control and enable valve position indication.

D. Select "LOCAL" on the controller selector switch to transfer control and select "LOCAL" on the push button below each controller to enable valve position indication.

ANSWER:

D. Select "LOCAL" on the controller selector switch to transfer control and select "LOCAL" on the push button below each controller to enable valve position indication.

REFERENCE: EOP 35.13

JUSTIFICATION: Step informs the operator to perform 2 steps, select LOCAL on the controller selector switch to transfer control and select LOCAL on the push-button below the controller to enable valve position indication.

OBJECTIVES: ASP-7

K/A: APE 068 EK2.03, Relationship controller/positioners

New question

SRO 62

The control switch for PORV 3RCS*PCV456 at MB4 is taken to C.OSE and the LOCAL/REMOTE switch at the TSP is placed in LOCAL.

Which of the following describes the response of the PORV if RCS pressure increased to 2350 psia?

A. The PORV will not open until the switch at MB4 is taken back to AUTO.

B. The PORV will open but will not close until the switch at the ASP is placed in CLOSE.

C. The PORV will open and close normally.

D. The PORV will not open until the REMOTE/ISOLATE switch at the FTSP is also taken to LOCAL.

ANSWER:

C The PORV will open and close normally.

REFERENCE: LSK 25-1.2B

JUSTIFICATION: In LOCAL or REMOTE, the PORV will open as long as the switch at the ASP (when in LOCAL) or MB4 (REMOTE) is in AUTO. (A incorrect, C correct)

. incorrect, going to close will close the PORV however, this action is not required, the PORV will close once RCS pressure is below the HI-HI setpoint.

D incorrect, there is not a FTSP switch for this PORV.

OBJECTIVE: ASP08

K/A: 068 EA1.21, Transfer control panel

SRO 63

The plant is at full power.

The normal power supply to Inverter 6 trips open.

10 minutes later, the operators locally aligns to the alternate supply to IAC-6 using the "ALTERNATE SOURCE TO LOAD" push-button.

25 minutes after transferring to the alternate supply, an LOP occurs.

Which of the following describes the status of inverter 6?

A. Inverter 6 is de-energized.

B. Inverter 6 is energized but will de-energize in 5 minutes.

C. Inverter 6 is energized but will de-energize in 20 minutes.

D. Inverter 6 is energized but will de-energize in 30 minutes.

ANSWER:

A. Inverter 6 is de-energized.

REFERENCE: EE-1BA, 063T

JUSTIFICATION: When the normal power supply is lost, a timer in the breaker to the inverter begins to time out, after 30 minutes the breaker trips open, isolating the DC supply (and the normal AC supply) from the inverter. Aligning the alternate supply does not change or reset the timer. In this case, when the LOP occurs the 30 minute has timed out and therefore the inverter will be de-energized. (A correct)

OBJECTIVES: 12507C

K/A: 063 K4.02, auto swap

modified from exam bank item 2490

SRO 64

The RWST is at the Technical Specification minimum when a design basis LOCA occurs. Only one train of ~~ECCS~~ SAFEGUARDS components actuate.

Approximately how long will it take to reach the automatic RHR pump trip setpoint?

A. 20 - 30 minutes

B. 35 - 45 minutes

C. 50 - 60 minutes.

D. 85 - 95 minutes.

ANSWER:

C. 50 - 60 minutes.

REFERENCE: ECC006T

USTIFICATION: Runout Flow from:
1 CHS pump 550 GPM at 650 psia
1 SIS pump 650 GPM at 650 psia
1 QSS pump 5000 GPM at 120 psia
1 RHR pump 5000 GPM at 120 psia

12,000 GPM
num level: 1,166,000 gallons
r setpoint: 520,000 gallons
e injected: 646,000 gallons
646/12 = 54 minutes

ESF13C

006 K5.06, ECCS flow/pressure

bank item 2643

SRO 65

The "A" EGLS is in Test 1 to perform testing of the RHR pump. The "A" RHR pump toggle switch aligned to the "TEST" position, all other components are in "INHIBIT".

A loss of offsite power occurs.

Which of the following describes the response of the "A" EGLS?

A. The EGLS sees both the LOP and the Test signal, resets and starts all equipment in the required mode and the RHR pump is OFF.

B. The EGLS sees both the LOP and the Test signal, but only starts the RHR pump.

C. The EGLS comes out of Test, resets and starts all equipment in the normal LOP mode, the RHR pump is OFF.

D. The EGLS comes out of Test and resets, however, none of the equipment in "Inhibit" is sequenced in the normal LOP mode and the RHR pump is running.

ANSWER:

C. The EGLS comes out of Test, resets and starts all equipment in the normal LOP mode, the RHR is OFF.

REFERENCE: LSK 24-9.4H & 27-7K

JUSTIFICATION: If an actual accident occurs during a Test 1 sequence, the EGLS will automatically reset to normal and carry out its automatic actions. Test/Inhibit toggle switch position will have no effect on equipment operation. (A, B, and D incorrect)

C is correct, the RHR pump is not started for the LOP and will not be started by the sequencer. All other equipment, such as the service water pumps will start in their normal sequence.

Test 2 functions identically to Test 1 with two major exceptions. The first is that in addition to responding to the local Test pushbuttons it will also respond to external input signals, whether generated by I&C or by actual plant conditions. The second is that an actual external input signal will not reset the sequencer to normal; therefore, the Test/Inhibit switches remain in effect and will determine if each piece of sequenced equipment will actually start (test position) or not (inhibit position).

OBJECTIVES: EDS06C

K/A: 064 K4.11, Auto load sequencing

modified from exam bank item 258

SRO 66

The crew has entered FR-S.1, "Response to Nuclear Power Generation/ATWS" and is aligning for immediate boration of the RCS through one gravity feed boration valve.

Which of the following would be an acceptable flowrate?

A. Any flow between 5 and 50 gpm.

B. Any flow between 25 and 75 gpm.

C. Any flow between 40 and 90 gpm.

D. Any flow greater than 100 gpm.

answer:

C. Any flow between 40 and 90 gpm.

REFERENCE: FR-S.1, step 4

JUSTIFICATION: Step 4b. RNO states the limit when using at least one gravity feed boration valve is less than 100 gpm. Step 4e. has the operator verify flow equal to or greater than 33 gpm.

A and B incorrect, minimum flow is less than required.

C correct, both values are acceptable.

D incorrect, flow is greater than allowed.

K/A: APE 024 AK2.01, Relationship boration flow/valves

OBJECTIVE: FS103C

New question

SRO 67

Plant Conditions

- 100% power
- It is discovered that the crew has violated the Reactor Core Safety Limit.

What ACTION is required?

A. Take action to be within the limit within 1 hour, be in HOT STANDBY within the next hour, and comply with the requirements of Specification 6.7.1.

B. Be in HOT STANDBY within 1 hour and notify the NRC within the following hour.

C. Be in HOT STANDBY and notify the NRC within 1 hour.

D. Be in HOT STANDBY and notify the NRC, the Senior Vice President and CNO-Millstone within 1 hour.

ANSWER:

C. Be in HOT STANDBY and notify the NRC within 1 hour.

REFERENCE: Technical Specifications 2.1.1, 2.1.2, the 2.1 bases, and section 6.7.2

JUSTIFICATION: If Specification 2.1.1 (Reactor Core) is violated the plant must be in HOT STANDBY within 1 hour and comply with Specification 6.7.1. Specification 6.7.1 requires the following:

- Unit in HOT STANDBY within 1 hour
- NRC Operations Center notified as soon as possible and in all cases within 1 hour. Senior VP Millstone and the Chairperson of NSAB notified within 24 hours.
- Safety Limit Report prepared
- Report submitted to the NRC within 14 days
- Operations not to resume until authoorized by the NRC

OBJECTIVES: SAD821

K/A: 2.2.25, Knowledge of Safety Limits

New question

SRO 68

The SIGMA refueling machine operator has just removed a fuel assembly from the core and started moving toward the upender.

A failure of the refueling cavity seal occurs. The fuel transfer cart is inside containment and the water level in the refueling cavity is dropping quickly.

What action should the SIGMA refueling machine operator take?

A. Return to the core and place the assembly in a core location.

B. Place the fuel assembly in the upender, and return to horizontal position.

C. Place the fuel assembly in the north saddle area of the refueling cavity and unlatch the assembly.

D. Place the fuel assembly in the north saddle area of the refueling cavity and lower the assembly until the cable is slack.

ANSWER:

A. Return to the core and place the assembly in a core location.

REFERENCES: EOP 3572

JUSTIFICATION: Desired location for an assembly is back in the core if the SIGMA is near the core, if the operator has just started moving, then this would be the appropriate location. (A correct)

B incorrect, this is the location for an assembly when the upender is in containment and the refueling machine is away from the core.

C is incorrect, it is never desired to unlatch the assembly.

D incorrect, this location is appropriate if the refueling machine is away from the core and the transfer carriage is not in containment

OBJECTIVES: FHS06C

K/A: ape 036 A1.04, refueling operations/fuel handling accident 2.2.31

modified from exam bank item 209

SRO 69

The plant is at full power.

ESF testing results in inadvertent HI-2 actuation. All equipment operates as designed.

Which of the following reactor trip signals will be generated?

A. Safety Injection.

B. High Steam Pressure Rate.

C. Low-Low Steam Generator Level.

D. Low Pressurizer Pressure.

ANSWER:

C. Low-Low Steam Generator level.

REFERENCE: Westinghouse functional sheet 8

JUSTIFICATION: A, B, and D incorrect, HI-2 will not actuate SI. The High Steam Pressure Rate will only cause a MSLI. Pressurizer pressure will not go low in this situation.

C correct, MSIVs will shut, causing steam generators to shrink off scale, causing a reactor trip. Another possibility would be high pressurizer pressure.

OBJECTIVES: RPS04C (g)

K/A: 013 K1.01, ESF initiation signals

modified from 2657

SRO 70

A loss of refueling cavity seal has occurred and level in the refueling cavity is decreasing slowly.

An operator on the spent fuel side begins to close the transfer tube gate valve prior to a spent fuel assembly being transferred from containment to the spent fuel building. When the valve is moved back to the open position, the operator cannot return it to the full open position.

Which of the following interlocks, if any, would have to be bypassed to allow the assembly to be transferred to the spent fuel building?

A. None.

B. The "valve interlock" only.

C. The "traverse interlock" only.

D. Both the "valve interlock" and the "traverse interlock".

ANSWER:

B. The "valve interlock" only.

REFERENCE: OP3303C

JUSTIFICATION: The bypass interlock key for "Valve Interlock," allows the cart to be traversed with the transfer tube valve not full open. (A incorrect, B correct)

The bypass interlock key for "Traverse Interlock," allows the cart to be traversed with no power to containment side control panel and/or the "Traverse Control," switch in the "OFF" position on the containment side control panel. (C and D incorrect)

OBJECTIVES: FSH05C

K/A: ape 036 K2.01, Fuel handling equipment

new (based on 1582)

SRO 71

PLANT CONDITIONS:

- The unit is in MODE 5
- RWST level is 450,000 gallons

The crew is evaluating conditions for filling the refueling cavity from the RWST.

Should the crew make the decision to fill the cavity?

A. Yes, there is adequate level in the RWST to fill the cavity, there is no minimum RWST level required in MODE 5, and the water in the cavity can be used to satisfy the shutdown risk minimum inventory requirements.

B. No, to fill the refueling cavity requires more than 450,000 gallons.

C. Yes, there is adequate volume in the RWST to fill the cavity and still be above the required Technical Specification level.

D. No, filling the cavity would result in an RWST level less than the minimum amount required by Technical Specifications and shutdown risk minimum inventory requirements.

ANSWER:

D. No, filling the cavity would result in an RWST level less than the minimum amount required by Technical Specifications and shutdown risk minimum inventory requirements.

REFERENCE: TS 3.1.2.5 & SFC033T, OP3305, OP3260A-4

JUSTIFICATION: The cavity holds about 260,000 gallons (B incorrect) and is filled with water from the RWST in preparation for refueling. TS requires 250,000 gallons, filling the cavity will leave 190,000 gallons, (C incorrect).

If the Boric Acid storage system is OPERABLE, the RWST could be drained without entry into Technical specification 3.1.2.5, however, OP3260 requires the RWST be maintained greater than 250,000 gallons. (A incorrect, D correct)

OBJECTIVES. 2522, 2523 and 2041

K/A: 034 A1.02, Level in refueling canal

SRO 72

PLANT CONDITIONS:

- CET's are 1250°F
- A loss of offsite power has occurred
- "A" EDG did not start
- "B" Charging and "B" SI pumps have tripped and cannot be restarted
- No AFW pumps are running, narrow range levels in all steam generators is off scale low
- Wide range levels in all four SGs are approximately 45% and trending down slowly

Which of the following is the PREFERRED method for cooling the core for the existing plant conditions?

A. Depressurize the secondary system at a rate not to exceed a cooldown rate of 100°F/hour.

B. Dump steam to the condenser at the maximum rate.

C. Restart reactor coolant pumps, one at a time until CETs are less than 1200°F.

D. Open all pressurizer PORVs and reactor head vents.

ANSWER:

D. Open all pressurizer PORVs and reactor head vents.

REFERENCE: FR-C.1

JUSTIFICATION: No offsite power, RCPs unavailable.
No AFW pumps and narrow range less than 6%, cannot depressurize the secondary
No High head pumps, cannot restart high head injection.
only available option is opening PORVs and reactor head vents (D correct)

OBJECTIVES: FC102C

K/A: 074 K1.03, Process of removing heat

modified from exam bank item 2520

SRO 73

In accordance with Millstone 3 procedures, when is the appropriate time for transferring to hot leg recirculation?

A. 8 hours after the LOCA occurred

B. 8 hours after Emergency Core Cooling has been placed in Cold Leg Recirculation.

C. 9 hours after the LOCA has occurred

D. 9 hours after the Emergency Core Cooling Systems have been placed in Cold Leg Recirculation.

ANSWER:

C. 9 hours after the LOCA has occurred

REFERENCE: Millstone 3 Basis Document for E-1, Step 22

JUSTIFICATION: The requirement in the MP3 procedures is to shift to Hot Leg Recirculation 9 hours after the initiation of the event (LOCA)(D is incorrect). The procedures do require the operators to prepare for Hot Leg Recirculation 8 hours after the event started (A and B are incorrect)

OBJECTIVES: ESF11C

K/A: 011 A.1.11, Long term core cooling

modified from exam bank item 866

SRO 74

The crew is responding to a LOCA in the Auxiliary Building using ECA-1.2, "LOCA Outside Containment."

If the break is isolated, which procedure will the crew transition to from ECA-1.2?

A. ES-1.2, "Post LOCA Cooldown and Depressurization."

B. E-1, "Loss of Reactor or Secondary Coolant."

C. ECA-1.1, "Loss of Emergency Coolant Recirculation."

D. ES-1.1, "SI Termination."

ANSWER:

B. E-1, "Loss of Reactor or Secondary Coolant."

REFERENCE: ECA-1.2, Step 5

JUSTIFICATION: Step 5 of ECA-1.2 has the operator check if the break has been isolated. If the break is isolated the crew transitions to E-1. If the break is not isolated the crew is transitioned to ECA-1.1 per the step 5 RNO. (B correct, A, C and D incorrect)

"A" is incorrect because there is no entry into ES-1.2 from ECA-1.2

"B" is correct because if the break is isolated the crew should go E-1.

"C" incorrect, this procedure is the choice if the break is not isolated.

"D" is incorrect because the crew must first go to E-1 and subsequently to ES-1.1

OBJECTIVES: A1204C

K/A: West E04, EK 1.2, Associated normal, abnormal, EOPs

New question

SRO 75

Operation of which of the following RPCCW valves, if any, would be DIRECTLY affected by Safety Injection actuation?

A. No RPCCW valves are directly affected by SI actuation.

B. RPCCW system cross-connect to chilled water system valves, 3CCP*MOV222 thru 229.

C. Containment header cross-connect valves, 3CCP*AOV179A&B and 3CCP*AOV180A&B.

D. The A/B RPCCW non-safety train headers, 3CCP*AOV197A/B, 10A/B, 194A/B, 19A/B.

ANSWER:

C. Containment header cross-connect valves, 3CCP*AOV179A&B and 3CCP*AOV180A&B.

REFERENCE: P&ID 121A & B

JUSTIFICATION: B and D incorrect, valves operate on CIA, not SI
C correct, the containment cross-connect valves, will close, if open on SI.
A incorrect, the cross-connect valves are directly affected by the SI.

OBJECTIVES: CCP04C

K/A: ape 026 EK3.02, Auto actions on ECCS actuation

modified from millstone LOIT exam 1995

SRO 76

FR-Z.1, Response To High Containment Pressure, places the hydrogen recombiners in service:

A. 24 hours after a large break LOCA

B. When Hydrogen concentration reaches 4%

C. Following any reactor vessel head venting

D. When Hydrogen concentration reaches 0.5%

ANSWER:

D. When Hydrogen concentration reaches 0.5%

REFERENCE: FR-Z.1, Step 12.c, starts hydrogen recombiners if hydrogen concentration reaches 0.5%

JUSTIFICATION: The FSAR analysis assumes the hydrogen recombiners are started within 24 hours following the LOCA with an estimated hydrogen concentration of 1.6%. The FSAR notes that by procedure the recombiners are started well before this level is reached (A is incorrect)

FR-I.3, Response to Voids In The Vessel, ensures hydrogen recombiners are already operating prior to venting the head. It determines a venting time to reach a concentration of less than 3%. (B and C are incorrect)

OBJECTIVES: FZ103C

K/A: West E14 EK1.2, Loss of containment integrity

new question

SRO 77

PLANT CONDITIONS:

- plant shutdown in progress
- reactor power is 7%

Intermediate range channel N35 fails as indicated by "IR 1 Loss of Compensation Voltage" annunciator alarming at MB4C 5-2.

The crew has entered AOP 3571, "Instrument Failure Response".

Which of the following actions should be taken by the crew?

A. Continue with the shutdown, actuate both SR reset switches when reactor power is believed to be in or near the Source Range.

B. Reduce power to below 5% and remain there until the channel is restored to OPERABLE status.

C. Stop the shutdown and remain in MODE 1, less than 10%, until the channel is restored to OPERABLE status.

D. Continue with the shutdown, and trip the applicable bistables.

ANSWER:

A. Continue with the shutdown, actuate both SR reset switches when reactor power is believed to be in or near the Source Range.

REFERENCE: AOP 3571 Rev. 3 Attachment E

JUSTIFICATION: AOP 3571, step 2 states that if the IR fails during a shutdown OR has symptoms of undercompensation, Actuate both source range reset when reactor power is believed to be in or near the Source Range. (A correct)

The Technical Specification ACTIONS apply to increasing power, (B and C incorrect)

AOP 3571 informs the operator that there are no bistables to be tripped. (D incorrect).

OBJECTIVES: NIS05C; NIS07C

K/A: 033 K3.02, Loss of intermediate range

1995 LOIT SRO exam

SRO 78

An LOP occurs and both diesels start and energize buses 34C and 34D.

Subsequently, the operator actuates CDA by pressing one of the 2 sets of 2 push-buttons for CDA on MB2.

Two minutes later, the operator notices that the "A" RPCCW pump is running and the "B" RPCCW pump is not.

Which of the following describes a probable cause for these indications?

A. The "A" sequencer has malfunctioned and the "B" sequencer has operated correctly because both trains of CDA are actuated by pressing either of the 2 sets of 2 push-buttons.

B. The "A" sequencer has functioned correctly and the "B" sequencer has malfunctioned because all 4 push-buttons must be pushed to actuate both Trains of CDA.

C. Both sequencers have operated correctly because the manual CDA signal is train specific and only the "B" train must have been actuated by the operator.

D. The "A" sequencer has malfunctioned and "B" sequencer has operated correctly because both trains of CDA have actuated and the RPCCW pumps are locked out from starting on a CDA for 6 minutes.

ANSWER:

A. The "A" sequencer has malfunctioned and the "B" sequencer has operated correctly because both trains of CDA are actuated by pressing either of the 2 sets of 2 push-buttons

REFERENCE: LSK 24-9.4A, 27-18A

JUSTIFICATION: For a LOP and LOP/SI, the RPCCW pumps are restarted. For a CDA they are not. Actuating CDA by using either set of 2 push-buttons will actuate both trains of CDA (C incorrect, B incorrect).

The "A" pump is running, therefore, the "A" EGLS has not performed properly, the "B" pump should not be running, therefore, the "B" EGLS has perform correctly. (Only A correct)

There is no lockout on the RPCCW pumps, (only the RSS pumps)

OBJECTIVES: CCP06C

K/A: EPE 056 A2.47, Proper OPS, diesel sequencer

New question

SRO 79

What is the basis for isolating the RCP Seal Supply Isolation valves, 3CHS*MV8109A,B,C and D in ECA-0.0, "Loss of All AC Power"?

A. Protect the system from steam formation due to RCP thermal barrier heating.

B. Ensure the necessary amount of ECCS flow is available, if required, when an emergency bus is energized.

C. Allow starting a charging pump in the normal charging mode as part of recovery without concern for damaging the RCPs.

D. Prevent a LOCA outside containment of potentially a few hundred gallons per minute should RCP number 1 seal failure occur.

ANSWER:

C. Allow starting a charging pump in the normal charging mode as part of recovery without concern for damaging the RCPs.

REFERENCE: ECA-0.0 Bkgd Doc, step 8

JUSTIFICATION: Isolating the RCP seal injection lines prepares the plant for recovery while protecting the RCPs from seal and shaft damage that may occur when a charging/SI pump is started as part of the recovery. <u>With the seal injection lines isolated, a charging/SI pump can be started in the normal charging mode without concern for cold seal injection flow thermally shocking the RCPs.</u> Seal injection can subsequently be established to the RCP consistent with appropriated plant specific procedures.

OBJECTIVES: A0002C

K/A: 022 K3.07, Isolating charging

modified from exam bank item 1999

SRO 80

The plant is in MODE 5 at mid-loop in accordance with OP 3270A, "Reduced Inventory Operation Mode 5 (IPTE)". Both trains of RHR are in service at 1000 GPM.

RCS level begins to decrease and amps for the RHR pumps begin to oscillate.

Which of the following actions should be taken by the crew?

A. Maximize charging flow to increase RCS level while maintaining RHR flow at 1000 gpm.

B. Trip both RHR pumps and immediately perform the appropriate actions for venting and restarting the pumps in accordance with OP3310A, "Residual Heat Removal System".

C. Trip both RHR pumps and Go to EOP 3505, "Loss of Shutdown Cooling and/or RCS Inventory".

D. Trip one RHR pump, allow the system to stabilize, vent and restart the tripped pump, then repeat the sequence for the other RHR pump.

ANSWER:

C. Trip both RHR pumps and Go to EOP 3505, "Loss of Shutdown Cooling and/or RCS Inventory".

REFERENCE: OP 3270A

JUSTIFICATION: 4.2.2 At any time, IF impending loss of both RHR trains is evident or an uncontrolled decrease in RCS level occurs, TRIP the RHR pumps and Go To EOP 3505, "Loss of Shutdown Cooling and/or RCS Inventory."

question has both indications, C is correct.

OBJECTIVES: E0502C

K/A: EPE 025 K3.03, Auto actions on ECCS

question modified from 1297, which was probably not used in program (still in 3 distracter format)

SRO 81

Initial Conditions:

- The unit is at 100% power
- The South 345kv bus was lost due to a fault. It will be restored to service in 2 hours

A fault on "B" NSST results in a reactor trip.

Following the reactor trip, all AFW flow is lost. Attempts to establish Auxiliary Feedwater have failed. All steam generator wide range levels are 35% and decreasing.

Assuming no change in plant conditions or trends, which of the following will be the next recovery action taken by the crew?

A. Establish Main Feedwater.

B. Bleed and Feed of the Reactor Coolant System.

C. Depressurize the secondary by dumping steam to the condenser to establish condensate flow to at least one steam generator.

D. Depressurize the secondary by dumping steam to the atmosphere to establish condensate flow to at least one steam generator.

ANSWER:

B. Bleed and Feed of the Reactor Coolant System.

REFERENCE: Potential Core Damaging Event: Loss of Secondary Heat Sink, FR-H.1, EE-1A, LSK 24-2D, 24-3B and 24-2B

JUSTIFICATION: Under the step to establish a secondary heat sink, using main feed is preferred over condensate or bleed and feed. Either condensate or main feedwater will restore secondary heat sink. However, both are unavailable because the fast transfer of 6.9KV buses is defeated by the South bus being de-energized. If the current trend continues, the crew will be initiating bleed and feed once wide range levels are less than 27%.

OBJECTIVE: MC1104

K/A: W/E 05 EK1.2, Procedures for

modified ·· m exam bank item 3197

SRO 82

Inverter 2 output breaker to VIAC-2 trips on a spurious signal.

Which of the following describes how to re-energize the VIAC?

A. The static switch will automatically transfer to the DC source.

B. Manually transfer to the alternate AC power source using the manual bypass.

C. The static switch will automatically transfer to the alternate AC power source.

D. Manually transfer to the alternate AC power source using the static switch.

ANSWER:

B. Manually transfer to the alternate power source using the manual bypass.

REFERENCE: EE-1BG, OP 3345B

JUSTIFICATION: If the output breaker trips, the static switch will not be available, (D, C and A incorrect), the crew will have to manually switch to the alternate source using the manual transfer switch. (B correct).

OBJECTIVES: 12006C

K/A: 057 EA1.01, Manual inverter swapping

taken from static sim exam question 26-4

SRO 83

A plant start-up is in progress. The blue P-10 permissive light has just come "ON".

No operator actions have been taken.

Which of the following will result in an automatic reactor trip?

A. Trip of one RCP.

B. Power range channel N41 fails HIGH.

C. Intermediate Range channel N35B fails HIGH.

D. The operator at MB-4 places the Block/Reset Switch for NIS Channel 31 in the "Reset" position and then depresses the button.

ANSWER:

C. Intermediate Range channel N35B fails HIGH.

REFERENCES: NIS Text (NIS015T); Data Chapter, Section II.A and Functional Sheet 3 and 4. OP3203, step 5.13

JUSTIFICATION: 5.13.1 Observe the P7 permissive blue light is OFF (MB4D, 5-3) and the P10 permissive blue light is ON (MB4D, 4-3) then

5.13.1.1 BLOCK both channels of Power Range Reactor Trip (low range).

5.13.1.2 BLOCK both channels of Intermediate Range Reactor Trip.

Above P-10 and below P-8, reactor trip occurs on 2 RCPs trip (A incorrect)

Above P-10, the Source Ranges are automatically blocked from energizing, (D incorrect)

Power range high flux (high or low) is 2 of 4, even if it assumed that the high failure causes the bypass reg valve to open, high steam generator level does not cause reactor trip, (B incorrect)

A failure of an IR will cause a trip because until the trip is blocked by the operator it will function, (C correct)

OBJECTIVES: NIS04C (a.2)

K/A: 015 K4.08, NIS permissives 3.7/3.8

modified from exam bank questions 988 and 56

SRO 84

INITIAL CONDITIONS:

- The plant is operating at 30% power.
- The 'A' TDFW pump is in service.
- All control systems are operating in Automatic.

The controlling channel of Steam Flow for the 'D' Steam Generator fails HIGH.

Assuming NO operator actions, which of the following describes the expected plant response?

A.
- Main Feed pump speed increases
- "D" Main Feed Reg valve opens
- Actual level in "D" steam generator increases
- FWI and turbine trip occurs
- Reactor trip on Low-Low SG level

B.
- Main Feed pump speed increases
- "D" Main Feed Reg valve opens
- Actual level in "D" steam generator increases
- FWI and turbine trip occurs
- Reactor trip on turbine trip

C.
- "D" Main Feed Reg valve opens
- Level error turns level prior to FWI
- "D" Main Feed Reg valve throttles closed to maintain steam generator level
- "D" steam generator level returns to program level

D.
- Main Feed pump speed increases
- "D" Main Feed Reg valve opens
- Actual level in "D" steam generator increases
- FWI, MSI and turbine trip occurs
- Reactor trip on Low-Low SG level

ANSWER:

A.
- Main Feed pump speed increases
- "D" Main Feed Reg valve opens
- Actual level in "D" steam generator increases
- FWI and turbine trip occurs
- Reactor trip on Low-Low SG level

REFERENCE: SGWLC T.M., Functional and process drawings, Simulator

JUSTIFICATION: Increase in steam flow on 'D' S/G causes total steam flow to increase which causes programmed MFP DP to increase above actual DP and MFP speed to increase. Increase in steam flow causes about 100% flow error (133% steam flow, 30% feed flow) flow error for the 'D' S/G FRV control to increase causing the FRV to open more.

Combination of increased speed and the FRV being more open causes 'D' S/G level to increase rapidly to the P-14 setpoint resulting in a FWI and tripping of the MFP's and Main Turbine. At 80% actual level, FWI will occur on P-14. (no reactor trip, below P-9 and P-14 is not a reactor trip signal, "B" incorrect)

FWI causes the main feed lines to isolate to all S/G's and the main feed pumps to trip with MSIV's still open ("D" incorrect); steam dumps will open to remove decay heat and RCP heat, which results in a decrease in level in all S/G's until the AFW pumps start and Auto Rx Trip occurs due to the Low Low level ("A" is correct). Another main feed pump trip signal will occur after 15 seconds due to the reactor trip.

OBJECTIVES: FWS07C (9.3)

K/A: 016 K3.14, Effects on SG

modified from exam bank item 1519

SRO 85

Which of the following describes the actions, if any, necessary to restore the operation of the Group A and B pressurizer backup heaters following a loss of power or low pressurizer level cutout?

A. For low pressurizer level, heaters will be restored as soon as pressurizer level is restored.
For a loss of power, the heaters must be turned "OFF" and then back to "AUTO"

B. For the low pressurizer level, level must be restored and the heaters must be turned "OFF" and then back to "AUTO".
For a loss of power, the heaters must be turned "OFF" and then back to "AUTO"

C. For the low pressurizer level, level must be restored and the heaters must be turned "OFF" and then back to "AUTO".
For a loss of power, the heaters are restored as soon as the manual start block is removed.

D. For low pressurizer level, heaters will be restored as soon as pressurizer level is restored.
For a loss of power, the heaters are restored as soon as the manual start block is removed.

ANSWER:

A. For low pressurizer level, heaters will be restored as soon as pressurizer level is restored.
For a loss of power, the heaters must be turned "OFF" and then back to "AUTO"

REFERENCE: OP3353.4B, 6-3, LSK 25-1.2F

JUSTIFICATION: loss of power has locked out backup heater group A or B, CYCLE associated control switch to "OFF" and back to "AUTO" (MB4).

for low pressurizer level, as soon as level is restored, the heaters should be restored, A correct.

OBJECTIVES: PPL04C

K/A: 010 K6.03, Effects heaters/spray/PORV

New question

SRO 86

A failure of a pressurizer pressure channel occurred during which the operator mistakenly adjusted the pressurizer pressure master pressure controller, 3RCS-PK455A, down by 100 psia. All actions in AOP 3571, Instrument Failure Response, for the failed pressure channel have been completed. The operator still has PK455A in manual and RCS pressure is 2235 psia, trending up slowly. The control room team is preparing to place the pressurizer pressure master pressure controller, 3RCS-PK455A in Automatic.

The US has directed the RO to use the applicable procedures and restore pressurizer pressure control to automatic at normal RCS pressure.

Which of the following describes the appropriate sequence of steps necessary to place the controller in automatic?

A.
- Set setpoint to desired pressure
- Check output is zero
- Place pressurizer pressure master pressure controller in auto
- Place ALL heaters and spray in auto

B.
- Set setpoint to desired pressure
- Adjust pressurizer pressure to setpoint
- Place pressurizer pressure master pressure controller in auto
- Place heaters (except, turn on "C") and spray in auto

C.
- Adjust pressurizer pressure to desired pressure
- Set setpoint to match pressurizer pressure
- Place heaters (except, turn on "C") and spray in auto
- Place pressurizer pressure master pressure controller in auto

D.
- Adjust pressurizer pressure to desired pressure
- Set setpoint to match pressurizer pressure
- Place pressurizer pressure master pressure controller in auto
- Place ALL heaters and spray in auto

ANSWER:

B.
- Set setpoint to desired pressure
- Adjust pressurizer pressure to setpoint
- Place pressurizer pressure master pressure controller in auto
- Place heaters (except, turn on "C") and spray in auto

REFERENCE: OP3301C, Section 4.1

JUSTIFICATION: Procedure OP3301G is a General Use procedure and contains the following steps:

4.1.4 SET setpoint controller, 3RCS-PK455A (MB4), pressurizer pressure master controller, to desired pressure to be maintained (normal operating pressure is 2,250 psia).

4.1.5 With pressurizer heaters and spray valves in manual, ADJUST pressurizer pressure to match master controller setpoint.

4.1.6 To place Pressurizer Pressure Control System in automatic, PERFORM following.

a. PLACE 3RCS-PK455A (MB4), pressurizer pressure master controller, in "AUTO."

subsequent steps have heaters, A, B, D and E in auto, C "on" and then spray controllers in auto

OBJECTIVES: PPL04C (c)

K/A: 010 K6.03, Controllers and Positioners

New question

SRO 87

The unit is in MODE 1.

Why does OP3337, Radioactive Gaseous Waste System, inform the operator that if the stack discharge stop valve, 3HVR*V42 is OPEN. either a process vent fan, 3GWS-FN1A or 3GWS-FN1B, or a supplementary leak collection and release exhaust fan. 3HVR*FN12A or 3HVR*FN12B, should be in operation?

A. Ensure appropriate dilution air flow.

B. Ensure the calculated radiation nonitor alarm setpoints are based on the actual discharge flowrate.

C. Provide a positive air flow to the Unit 1 stack, preventing backflow from the Unit 1 stack.

D. Provide a recirculation flow path for the GWS system.

ANSWER:

C. Provide a positive air flow to the Unit 1 stack, preventing backflow from the Unit 1 stack.

REFERENCE: OP3337

JUSTIFICATION: Precaution in OP3337

6.5 During periods when the Supplementary Leak Collection and Release System is required to be OPERABLE, and stack discharge stop valve, 3HVR*V42 is OPEN, either a process vent fan, 3GWS-FN1A or 3GWS-FN1B, or a supplementary leak collection and release exhaust fan, 3HVR*FN12A or 3HVR*FN12B, should be in operation so that positive flow exists to the Unit 1 stack. IF none of the fans are available, stack discharge stop valve, 3HVR*V42, should be closed.

OBJECTIVES: WFB06C

K/A: 2.3.3, Radiation control, aux. system outside control room

New question

SRO 88

PLANT CONDITIONS:

- Steam break inside containment occurs from full power
- All systems operate as designed
- Narrow range level for intact steam generators is 15% with AFW flow at 100 GPM to each intact Steam generator
- The crew is now in FR-P.1, "Response to Imminent Pressurized Thermal Shock Condition"
- RCS temperature is stable
- RCS pressure is stable with only the control group of pressurizer heaters energized

The crew has determined a 1 hour soak is required.

Which of the following evolutions could be performed by the crew in the next hour?

A. Energize additional pressurizer heaters.

B. Place auxiliary spray in service.

C. Increase AFW flow to 300GPM per steam generator to raise steam generator levels to 50%.

D. Recirc RHR to establish boron concentration and place it in service.

ANSWER:

B. Place auxiliary spray in service.

REFERENCE: FR-P.1, step 25

JUSTIFICATION: Step 25: Determine if RCS temperature soak is required

b. Perform ALL of the following steps:
- DO NOT cooldown RCS until temperature has been stable for 1 hr (increasing SG level would not be allowed, C incorrect)
- DO NOT increase RCS pressure during the 1 hr stabilization period (A incorrect)
- Perform the actions of other procedures in effect which do NOT decrease RCS temperature or increase RCS pressure until the 1 hr stabilization period is complete
- Maintain RCS pressure and cold leg temperature within the limits shown on Attachment A (B adverse containment)

- Maintain cooldown rate in RCS cold legs LESS THAN 50°F in any 60 minute period

"A" is incorrect. Additional heaters would increase pressure.
"C" is incorrect. Additional feedwater would result in additional cooldown.
"D" incorrect, placing RHR in service after recirc would result in a cooldown.

"B" is correct because lowering pressure is allowed.

OBJECTIVES: FP103C

K/A: W/E 08 EA1.1, Function/safety systems

New question

SRO 89

INITIAL CONDITIONS:

- The unit is at 100% power
- Master pressure controller set to control at its normal pressure
- Pressurizer backup heaters are "OFF"
- Pressurizer spray valves are "MODULATING"
- Pressuriz.r PORV tailpipe temperature is 120°F

Which of the following RCS pressures would cause this response?

A. 2235 psia.

B. 2250 psia.

C. 2280 psia.

D. 2350 psia.

ANSWER:

C. 2280 psia.

REFERENCE: Functional sheet 11

JUSTIFICATION: Pressurizer backup heaters turn off at 2233 psia. Spray valves begin to modulate at 2275 psia, (25 psia above normal setpoint), because spray valves are modulating, "A" and "B" incorrect.

Pressurizer PORVs lift at 2350 psia, (setpoint plus 100 psia), tailpipe temperatures are normal, therefore, pressure must be less than 2350, "D" incorrect.

"C" correct, spray valves would be opening, heaters off, PORVs closed.

OBJECTIVES: PPL04C

K/A: 002 A3.01, Pressure/temperature/flow

modified from question 12

SRO 90

With the plant operating at 25% power, one of the 4 operating RCPs trips.

Which of the following describes the status of the steam generator pressures in the OPERATING loops one minute after the RCP trips?

A. Increased due to increasing steam generator temperature.

B. Decreased due to reactor trip on low - low steam generator level.

C. No change due to a constant steam demand.

D. Decreased due to increased steam flow.

ANSWER:

D. Decreased due to increased steam flow.

REFERENCE: Decrease in Reactor Coolant System Flow Rate (MCORE04)

JUSTIFICATION: When the pump stops, the running pumps will reverse flow through the idle loop. The affected steam generator will do very little steaming due its pressure dropping. The unaffected loops will increase their steaming, this will cause Pstm in the steam generators to decrease and Tcold in the loops to decrease also.

OBJECTIVE: MC0402

K/A: 003 K5.04, RCP effects on secondary

modified 3181

SRO 91

A reactor trip has occurred due to a turbine trip from full power. Narrow range steam generator levels are off scale low.

Why does ES-0.1, Reactor Trip Response instruct the operator to feed the steam generators at greater than 530 GPM?

A. To enhance natural circulation.

B. To provide an adequate heat sink for decay heat removal.

C. To ensure the steam generator U-tubes remain "wet" preventing hot and dry steam generators.

D. To prevent the formation of steam in the steam generator feed ring.

ANSWER:

B. To provide an adequate heat sink for decay heat removal.

REFERENCE: E-0, Background

JUSTIFICATION: AFW flow is necessary for secondary heat sink. If SG level is in the narrow range in at least one SG, a heat sink is available. However, if narrow range level has not been established, feeding at greater than 530 GPM verifies the ensures a heat sink for decay heat removal. If adequate AFW flow for decay heat removal cannot be established, the transition to the FR-H.1, RESPONSE TO LOSS OF SECONDARY HEAT SINK, is necessary to establish an alternate source of feed flow or an alternate heat sink. (B correct)

"A" incorrect, RCPs could be running, and neither steam generator level or AFW flow is checked to verify natural circulation.

"C" incorrect, the "wet" U-tube concept is a concern in FR-H.1 after generators have dried out and flow is to be established to them.

"D" incorrect, the J-tubes in the feed ring are to prevent steam formation in the feed line.

OBJECTIVES: S0103C

K/A: 007 EK1.06, Stabilization/relationship to decay heat

modified from exam item number 560

SRO 92

Which of the following would prevent the operator from OPENING RHR loop suction isolation valve RHS*MV8701B?

A. RHR to CHG/SI valve (3SIL*MV8804A) CLOSED

B. RSS to RHR cross-connect valves (3RSS*MV8837A) CLOSED.

C. RWST suction isolation valve (3SIL*MV8812A) OPEN.

D. RCS pressure on PT405 380 psia.

ANSWER:

C. RWST suction isolation valve (3SIL*MV8812A) OPEN.

REFERENCES: P/ID 112A, LSK 27-7B

JUSTIFICATION: To open (MV8701B)

- 3SIL*MV8804A RHR P1A to CHG Pump Valve Closed ("A" incorrect)
- 3SIL*MV8812A RWST to RHR P1A Valve Closed ("C" correct)
- 3RSS*MV8837A RSS to RHR XCONN Valve Closed ("B" incorrect)
- 3RSS*MV8838A RSS to RHR XCONN Valve Closed
- Wide-range loop 1 hot leg pressure, as read by PT405/405A must be less than 390 psia ("D" is wrong)

OBJECTIVES: RHR04C (a)

K/A: 005 K4.07, System interlocks 3.2/3.5

modified from exam bank item 169

SRO 93

While operating at 90% power a control Bank "D" Group 1 rod becomes misaligned higher than the rest of its group.

The crew has entered AOP 3552, Malfunction of the Rod Drive System, and is currently aligning the affected rod to the rest of the bank.

During the rod alignment, the procedure requires the operators to insert the affected rod until the next lower DRPI LED just changes state. The operator then resets the affected group step counter to a value of two steps higher than the affected rod's indicated DRPI position.

Which of the following describes the basis for resetting the affected group step counter to a value of two steps higher than the affected rod's indicated DRPI position?

A. This action will reset the logic cabinet master cycler to ensure proper group stepping.

B. Having determined the rods position, this action will reset the P/A converter to the proper value.

C. DRPI accuracy is two steps above the actual position, therefore, when the next DRPI indicator lights, the rod is actually two steps higher.

D. DRPI indicators will light when the rod is actually within ±2 steps of actual, setting the group step counters at the high end is conservative.

ANSWER:

C. DRPI accuracy is two steps above the actual position, therefore, when the next DRPI indicator lights, the rod is actually two steps higher.

REFERENCE: AOP 3552 attachment A, Basis document for step 6. Align Rod, DRPI handout

JUSTIFICATION: The coils are placed 6 steps apart with the LED positioned 1/2 the distance between the coils. As the rod steps through the coil, the LED above the coil will light. The band is 2 steps above and 3 steps below the indicated position. With the position of the misaligned rod now located, the affected group step counter can be reset to that position and the rod then inserted to the same height as the rest of the rods in the group as recorded initially. (C correct)

OBJECTIVES: RPI04C

K/A: 014 A2.04, Misaligned rod impact

modification of 1539, (not used in previous exams)

SRO 94

Power to VIAC-1 has been lost and cannot be immediately restored.

Assume that the operators are taking prompt action per AOP 3564 "Loss of One Protective System Channel" and that no bistables are in the tripped condition prior to VIAC-1 deenergizing.

Which statement describes the immediate consequence, if any, of this event?

A. The reactor will trip if power is below 10%.

B. The reactor will trip if power is above 10%.

C. The reactor will trip, regardless of power level.

D. The reactor will not trip, regardless of power level.

ANSWER:

A. The reactor will trip if power is below 10%.

REFERENCE: AOP3564, Functional sheet 4

JUSTIFICATION: Sheet 4 shows the IR are powered from protective channel I and II. If VIAC-1 deenergizes, IR 35 will de-energize, causing a trip on High Flux if power is less than 10%.

OBJECTIVES: RPS04C (2)

K/A: 012 K1.01, RPS & 120v vital 3.4/3.7

New question

SRO 95

Channel 1 is selected as the primary channel or the only input to all controllers on the Main Boards.

The loop D Thot instrument has failed HIGH.

The control room team completed all of the actions in AOP 3571, Instrument Failure Response, for this instrument failure and all applicable controls are in automatic.

Subsequently, pressurizer pressure instrument PT-457 fails HIGH.

Which of the following describes the expected plant response, if any?

A. Reactor remains at power.

B. Reactor trip will occur due to OTΔT

C. Reactor trip will occur due to OPΔT

D. Reactor trip will occur due to high pressurizer pressure.

ANSWER:

A. Reactor remains at power.

REFERENCES: AOP 3571

JUSTIFICATION: The Thot failure corrective action trips the OT and OPΔT bistables for loop "D". When the pressure channel fails, the high pressure trip B/S for that channel will actuate, but it will be the only one, requires 2 of 4, "D" incorrect.
"A" is correct because the plant will not trip.
"B" is incorrect because the reactor trip will not occur on OTΔT because failing high will increase the setpoint for the "B" loop and will not energize the required second bistable, (one is tripped in AOP 3571).
"C" is incorrect because pressure does not impact the setpoint.

OBJECTIVES: RPS04C

K/A: 012 K6.03, Trip logic circuits 3.03

New question

SRO 96

A loss of All AC power has occurred. The crew is attempting to locally start the "A" EDG.

The operator at the diesel reports that the Service Water valve, 3SWP*AOV39A is closed.

According to ECA-0.0, what actions, if any, are required?

A. Restore instrument air and then open the valve.

B. Vent air from the valve operator

C. Vent air using 3SWP*HV39A in the EDG enclosure.

D. No action required, the valve should open when the diesel is started.

ANSWER:

C. Vent air using 3SWP*HV39A in the EDG enclosure.

REFERENCE: ECA-0.0, attachment E, step 1, P & ID 133D

JUSTIFICATION: Attachment 1, step 1 has the operator open the service water valve, if it is not open by venting air by using AOV39A. (C correct)

"D" incorrect, the LOP should have already opened the valve(by the stem of the question indicating the valve is closed, this failed to work).

"B" incorrect, this is the action required for 3SWP*AOV39B

"A" incorrect, the valve fails open on loss of air.

OBJECTIVES: A0003C

K/A: 055 EA2.03, Actions to restore power 3.7/4.1

New question

SRO 97

With the plant operating at 100%, a total loss of offsite power occurs. The crew has entered ECA-0.0, "Loss of All AC Power".

The SBO diesel has been started.

Which of the following will allow the operator to energize emergency bus 34C from the SBO diesel?

A. After 60 seconds have elapsed, the operator resets the sta') LOP signa' at the sequencer.

B. After 6 minutes have elapsed, the operator resets the station LOP signal at the sequencer.

C. After 60 seconds have elapsed, the operator resets the LOP signal at MB2 and the station LOP signal at the sequencer.

D. After 6 minutes have elapsed, the operator resets the LOP signal at MB2.

ANSWER:

D. After 6 minutes have elapsed, the operator resets the sequencer LOP signal at MB2.

REFERENCE: LSK 24-3K

JUSTIFICATION: Once the LOP occurs, if the RSST does not energize the bus afte. 1.8 seconds, the bus is locked out for 6 minutes to provide sufficient time to restore from the emergency diesel. If the diesel is not able to be placed on the bus, after 6 minutes, the LOP lockout can be reset and either off-site or the SBO could be placed on the bus. ("D" correct)

"A" incorrect, both signals do not need to be reset and 6 minutes must have elapsed.

"B" incorrect, the station LOP signal is the wrong signal to reset.

"C" incorrect, the station LOP is incorrect signal and don't need to do both.

OBJECTIVES:

K/A: 055 EA2.06, Lockouts to restore bus 3.7/4.1

modified from exam item number 2077

SRO 98

A plant shutdown due to a large steam generator tube leak is being performed. The crew believes the tube leak is in the "C" steam generator.

During the shutdown, the plant trips and Safety Injection actuates.

What is the earliest the operator can isolate steam flow from the "C" steam generator?

A. When narrow range level in the "C" steam generator is greater than 6%.

B. After completing the immediate actions of E-0.

C. Once the transition is made from E-0 to E-3.

D. When directed to do so in E-3.

ANSWER:

D. When directed to do so in E-3.

REFERENCE:	OP 3272 EOP User's Guide Attachment 3, "Special Considerations"
JUSTIFICATION:	An operator may not isolate the steam line to a ruptured steam generator until directed by the procedure. This is not an identified safe condition unless specifically directed by the SGTR procedure
OBJECTIVES:	E3003C
K/A:	037 EA2.11, Isolate one/more SG

modified from 2277

SRO 99

A leak develops in the reference leg for the wide range level indication for one of the steam generators.

Which of the following describes the indication the operator would see at the main boards for the affected steam generator?

A. Wide range level would INCREASE, narrow range levels would not change.

B. Wide range level would DECREASE, narrow range levels would not change.

C. Wide range level and one narrow range level would INCREASE.

D. Wide range level and one narrow range level would DECREASE.

ANSWER:

C. Wide range level and one narrow range level would INCREASE.

REFERENCE: Non-Nuclear Instrumentation Text (NNI016T), P&ID 130C and D

JUSTIFICATION: $\Delta P = P_{REF} - P_{VARIABLE} = 0$ at 100% indicated SG Level
P_{REF} decreases . As ΔP approaches 0 the level indicated is HIGHER than actual.

The Wide Range instruments share a reference leg with one narrow range steam generator level.(A and B incorrect)

Therefore, the level in both the wide range and one narrow range level will each be INCREASING. (C correct, D incorrect)

OBJECTIVES: MC0202

K/A: 035 K6.03, Loss SG level detectors 2.6/3.0

taken from exam bank items 2469, 2168 and 1519

SRO 100

PLANT CONDITIONS:

- Reactor trip from full power and ESF actuation 5 minutes ago
- RCS Pressure: 1500 psia and increasing.
- RCS Tav: 470°F and decreasing
- PZR Level: 10% and increasing.
- SG Narrow Range Levels: All offscale low.
- SG Pressure: All 600 psig
- AFW Flow: 300 GPM per steam generator
- Containment Pressure: normal
- ECCS Equipment: All operating at normal flows.

Which of the following describes the cause of the transient?

A. Loss of Coolant Accident.

B. Steam Generator Tube rupture.

C. Steam break inside containment.

D. Steam break outside containment.

ANSWER:

D. Steam break outside containment.

REFERENCE: Increase in Heat Removal by the Secondary System (MCORE02)

JUSTIFICATION: Containment pressure is normal, therefore, a steam break inside containment or a LOCA has not occurred, also RCS pressure is increasing. (A and C incorrect).

If a tube rupture had occurred, RCS pressure and pressurizer level would be not increasing and Temperature would not have dropped to 470 degrees. (B incorrect)

A steam break outside containment would result in lowered steam generator pressure until the MSIVs closed, then RCS pressure and pressurizer level would begin to increase. RCS temperature would decrease due to AFW flow to all steam generators.

OBJECTIVES: MC0202

K/A: 040 EA2.03, Difference between steam line break vs LOCA

New question

Senior Reactor Operator Answer Key

1. d	26. d	51. b	76. d
2. d	27. d	52. a	77. a
3. c	28. d	53. c	78. a
4. c	29. d	54. c	79. c
5. c	30. c	55. a	80. ~~a~~ C TYPOGRAPHICAL ERROR - SEE SRO Key.
6. a	31. b	56. a	81. b
7. d	32. a	57. c	82. b
8. b	33. c	58. b	83. c
9. b	34. d	59. d	84. a
10. b	35. d	60. a	85. a
11. d	36. a	61. d	86. b
12. d	37. c	62. c	87. c
13. d	38. b	63. a	88. b
14. b	39. c	64. c	89. c
15. d	40. d	65. c	90. d
16. c	41. d	66. c	91. b
17. c	42. b	67. c	92. c
18. c	43. b	68. a	93. c
19. a	44. a	69. c	94. a
20. a	45. a	70. b	95. a
21. a	46. b	71. d	96. c
22. b	47. c	72. d	97. d
23. c	48. d	73. c	98. d
24. d	49. c	74. b	99. c
25. c	50. c	75. c	100. d

Attachment 2

Millstone Unit 3 RO WRITTEN EXAM W/ANSWER KEY

RO 1 MASTER

Which of the following conditions would require the crew to initiate immediate boration of the RCS?

A. In MODE 5, with dilution paths not isolated and cold calibrated pressurizer level at 50% core burnup is 6,000 MWD/MTU and RCS boron concentration is 2050 PPM. (See attached curves.)

B. With the plant at 100% power, an unexplained event is causing Tave and reactor power to increase, and control rods to insert.

C. After a reactor trip, with the crew performing ES-0.1, "Reactor Trip Response", the RCS cools down uncontrollably to 540°F.

D. While performing a rapid downpower IAW OP 3275, the ROD CONTROL BANKS LIMIT LO Alarm is received.

ANSWER:

B. With the plant at 100% power, an unexplained event is causing Tave and reactor power to increase, and control rods to insert.

JUSTIFICATION: "B" is correct, since this is indication of an unexplained positive reactivity addition.

"A" is ~~wrong~~ correct since there is ~~adequate~~ inadequate boron per the "loops filled" curve, and PZR level is > 40%. Ruff

"C" is wrong since the setpoint is 530°F.

"D" is wrong since Rod LO-LO is the Immediate Borate setpoint.

OBJECTIVE: CHS08C (a)

REFERENCE: AOP 3566 Entry Conditions

K/A: 024.EK3.01 Emergency Boration Requirements

Exam Item: 2373

Note: Credit should be given for both a and b. Ruff

RO 2

The following plant conditions exist:

- The plant has just completed a 428 day full power run.
- The plant is cooling down in MODE 3
- RCS temperature is 490°F
- RCS pressure is 1800 psia
- Excess letdown is in service

A leak develops in train "B" of the RPCCW system causing the train "B" surge tank level to INCREASE.

WHICH of the following could be the potential source of leakage into train "B"?

A. Excess letdown heat exchanger

B. "B" RHR heat exchanger

C. Letdown heat exchanger

D. Seal Water heat exchanger

ANSWER:

A. Excess letdown heat exchanger

REFERENCES: P&ID 1-CC-B2020A, CC
AOP 3555, RCS Leak, Attachment B

JUSTIFICATION: Only the letdown heat exchanger, the excess letdown heat exchanger and the seal water heat exchanger have reactor coolant moving through them in MODE 3 for the stated conditions. The letdown heat exchanger is on Train "A". The seal water return pressure is 30 - 50 psig, or less than RPCCW pressure. A leak into the "B" RPCCW surge tank can only come from excess letdown heat exchanger. The "B" heat exchanger is isolated from the RCS.

OBJECTIVE: CCP07C (e), CCP06C (e)

K/A: APE026 A2.01, 3.5

97 Exam 8

RO 3

Which of the following signals will cause the RPCCW to CDS cross-ties in containment (MOV226 - 229) to close?

A. High RPCCW flow rate

B. Containment Isolation - Phase A

C. Containment Isolation - Phase B

D. Low RPCCW surge tank level

ANSWER:

D. Low RPCCW surge tank level

REFERENCE: LSK 9-1C

JUSTIFICATION:

OBJECTIVE: CCP03C (b), CCP07C (e)

K/A: 026 EA 1.05 - RPCCW Surge Tank, include level control, alarms in RMS.

EXAM BANK ITEM: 2177

RO 4

The following conditions exist:

- The plant is currently in operational mode 2.
- Reactor power is 4%.
- Preparations to increase power to 100% are in progress.
- The Pressurizer Pressure Channel Selector Switch is in the PT-455/PT-456 position.
- The annunciator for HIGH PRESSURE ALARM, actuates.
- PORV PCV-456 opens.
- Actual pressurizer pressure is 2200 psia and decreasing rapidly.

Which one of the following describes the instrument malfunction which caused this transient and the response of the PORV (PCV-456)

A. PT-455 failed high and PORV will remain open in AUTO.

B. PT-456 failed high and PORV will close in AUTO when actual pressure decreases below 2200 psia

C. PT-456 failed high and PORV will remain open in AUTO

D. PT-455 failed high and PORV will close in AUTO when actual pressure decreases below 2200 psia

ANSWER:
B. PT-456 failed high and PORV will close in AUTO when actual pressure decreases below 2200 psia

REFERENCES: OP 3353.MB4A(3-4); AOP 3571; PPL system text

JUSTIFICATION: In the CHI/CHII position PORV456 is controlled by PT456. On high pressurizer pressure on channel 456 the PORV will open and will close when 2/4 pressurizer channel decrease to less than 2200 psia if the PORV is in AUTO. (B is correct)

A is incorrect because PT 456 failed and the PORV will close in AUTO.

C is incorrect because PORV will close if in AUTO.

D is incorrect because PT456 failed not PT455.

OBJECTIVES: PPL03D (a)

K/A: 027 A1.01 Ability to operate/monitor pressurizer heaters, spray, PORVs.

Modified question 2118

RO 5

Given the following situation:

- A safety injection actuation has occurred due to a steam line break upstream of the MSIV.
- The operators have completed the actions in E-0 and isolated the affected SG in accordance with E-2 "Faulted SG Isolation".

Which one of the following correctly describes the plant response, during the next hour, once the affected steam generator is empty? (Assume no further operator actions.)

A. Pressurizer pressure, level, and RCS hot leg temperature in the affected loop decrease.

B. Pressurizer pressure, level, and RCS hot leg temperatures continue to decrease.

C. Pressurizer pressure and level increase, and RCS hot leg temperatures continue to decrease.

D. Pressurizer pressure, level, and RCS hot leg temperatures increase.

ANSWER:

D. Pressurizer pressure, level, and RCS hot leg temperatures increase.

REFERENCE: MITCOR Text; FSAR Chapter 14, Increase in Heat Removal

JUSTIFICATION: Once the faulted SG has blown dry, ECCS injection flow will cause an increase in pressurizer pressure and level. Decay heat will cause RCS hot leg temperatures to increase. As a result, "D" is the correct answer.

OBJECTIVE: EO203C

K/A: 040K1.03, RCS Shrink/depressurization 3.8/4.2

95 LOIT NRC Exam

RO 6

While performing E-3, "Steam Generator Tube Rupture", prior to initiating the RCS COOLDOWN, the Unit Supervisor directs you to determine if the ruptured steam generator pressure is greater than 420 psig.

If pressure is less than 420 psig, E-3 directs you to transition to ECA - 3.1, "SGTR with Loss of Reactor Coolant - Subcooled Recovery Desired".

Which of the following describes a reason for ensuring the ruptured SG pressure is greater than 420 psig prior to performing the cooldown of the RCS?

A. To ensure subsequent cooldown does not cause a PTS concern.

B. To ensure the ruptured SG pressure is greater than the intact SG pressures.

C. To ensure that a low steam line pressure Safety Injection does not occur.

D. To ensure subsequent RCS cooldown does not cause a red path on Subcriticality due to a loss of shutdown margin.

ANSWER:

A. To ensure subsequent cooldown does not cause a PTS concern.

REFERENCE: E-3, Step 13 Basis, WOG Background Document

JUSTIFICATION: Although it is important that the ruptured SG pressure be greater than intact SG pressure, this verification is made at step 13 of E-3, and appropriate action taken if these conditions do not exist ("B" is incorrect). Since intact SG pressure must be less than ruptured SG pressure to maintain RCS subcooling, the RCS cooldown that would be required to maintain the necessary pressure differential between the ruptured and intact SG's would present an PTS concern in the RCS ("A" is correct). Distracter "C" cannot be correct because automatic SI actuation at this point in the recovery would have been defeated since SI is reset in Step 8 of E-3. "D" is incorrect because a Subcriticality red path only comes in for reactor power greater than 5% which could not occur based on the conditions provided.

OBJECTIVE: E3004C

K/A: WEST E08 EK2.2, PTS relationship to Heat Removal 4.0

Used 97 exam 8 (modified distracter)

RO 7

Given the following conditions:

- Reactor trip due to station blackout has occurred.
- Natural Circulation cooldown has been established.
- The crew is in ECA 0.0 dumping steam at the maximum rate.

The RO states that he is unsure if natural circulation is providing adequate heat removal.

Which one (1) of the following indications would indicate a potential problem with core heat removal by natural circulation. (Consider each condition independently).

A. SG pressures have decreased from 700 psia to 600 psia and are now starting to decrease at a slower rate.

B. Pressurizer level is 15% and decreasing while subcooling is 80°F and increasing.

C. T_{HOT} is 520°F and T_{COLD} is 460°F and the ΔT between the two is increasing.

D. T_{HOT} is 490°F and decreasing and CET's are decreasing slowly.

ANSWER:

C. T_{HOT} is 520°F and T_{COLD} is 460°F and the ΔT between the two is increasing.

REFERENCE: ECA-0.0; MITCOR Text

JUSTIFICATION: For adequate natural circulation the following should be occurring:

SG pressure stable or decreasing ↓ - This condition is satisfied. It is expected that the depressurization rate slows as pressure gradient decreases. (A is not a problem).

The pressurizer is expected to empty under these conditions. The RCS is adequately subcooled. The contraction cannot be compensated for due to the loss of power. (B is not a problem).

The core ΔT is greater than full power ΔT which is indicative of degrading natural circulation and cooldown. ΔT should be stable or decreasing during the cooldown, not increasing.

T_{HOT} and CET must be decreasing if a cooldown is in progress.

OBJECTIVE: A0003C

K/A: 055 AK1.02 Natural Circulation Cooling

New Question/Modified 2467

RO 8

The plant is at 100% power when the following alarm is received on Main Board 8A:

- Inverter 1 Trouble

Locally at Inverter 1, the operator observes that only the reverse transfer light is lit.

Which of the following events is the likely cause of these alarms/indications?

A. Inverter 1 is de-energized.

B. Inverter 1 DC feeder breaker has opened.

C. Inverter 1 is being supplied from it's DC supply.

D. Inverter 1 has transferred to it's alternate supply.

ANSWER:

D. Inverter 1 has transferred to it's alternate supply.

REFERENCE: OP 3353.MB8A, 1-5

JUSTIFICATION: "A" is incorrect, if the reverse power is lit, the UPS must have transferred to its alternate supply.

"C" is incorrect because the reverse transfer light would not light if Inverter 1 is being supplied by its backup DC power supply. Additionally, the backup DC would have to go through the same UPS as the normal AC supply.

"B" is incorrect because if the cause was only the DC feeder breaker opening the UPS would still be supplied by its normal AC power supply.

"D" is correct because the local indication is that the Inverter 1 is still powered otherwise there would be more alarms than just the reverse transfer.

OBJECTIVE: A64764D (1)

K/A Rating: APE057 A2.06, Inst. Bus Alarms for Alt Power

97 Exam 8

RO 9

Given the following conditions:

- The control room has been evacuated due to a fire.
- All remote shutdown system line ups have been completed.
- RCP #2 is running.
- The RCS is borated to cold shutdown conditions
- A plant cooldown is in progress

Which of the following describes the PREFERRED sequence to depressurize the RCS to 1950 psia in accordance with EOP 3504 over the next 4-8 hours.

A. De-energize all pressurizer heaters, use one PORV, use aux spray.

B. Use normal spray, use aux spray, use one PORV.

C. De-energize all pressurizer heaters, use aux spray, use one PORV.

D. Use aux spray, use normal spray, de-energize all pressurizer heaters.

ANSWER:

C. De-energize all pressurizer heaters, use aux spray, use one PORV.

REFERENCE: EOP 3504

JUSTIFICATION: Normal spray is not available from the auxiliary shutdown panel, thus b and d are incorrect.

The preferred way to depressurize the RCS is to deenergize heaters and allow the RCS to depressurize slowly due to ambient losses.

If need to depressurize further, procedure guides them to place letdown in service and use aux spray. If letdown not available, then use one PORV. Thus c is correct.

A is incorrect - sequence is reversed. Tested caution note in procedure.

OBJECTIVE: ASP03

K/A: 068 A1.12 Aux Shutdown Panels/controls

New Question

RO 10

Plant conditions are as follows:

- Reactor Trip & SI have occurred
- No high head SI pumps or charging pumps are running
- RCS pressure is 800 psia
- Containment pressure is 20 psia
- RVLMS 0% (plenum)
- CET's are 723°F and increasing

The crew has properly transitioned to the appropriate functional recovery procedure.

Why is the operator directed to check if one RCP should be stopped?

A. To minimize RCS inventory pumped or lost out of the break.

B. To reduce the heat input into the RCS and help maintain RCS inventory.

C. To reserve one RCP for future use and aid in the plant recovery.

D. To reduce the inventory lost when the seal packages fail in the running RCPs

ANSWER:

C. To reserve one RCP for future use and aid in the plant recovery.

REFERENCE: EOP Basis Document

JUSTIFICATION: EOP 35 F-02 Background document. One RCP is preserved to be able to be restarted in FR-C.1 if plant conditions degrade to an inadequate cooling situation.

OBJECTIVE: FC203C

K/A: 074K3.04 Tripping RCPs

New Question

RO 11

Plant Conditions

- The operators are performing ES-1.3, "Transfer to Cold Leg Recirculation."
- The RO is aligning the RHR and RSS Systems for cold leg recirculation using the recirc. array.

The STA informs the SM that a red path just came in on the Integrity CSF Status Tree.

Which of the following is the correct course of action?

A. Transition to FR-P.1 immediately because vessel integrity is in jeopardy.

B. The SM and BOP will perform FR-P.1 concurrent with the US and RO completing ES-1.3

C. Continue in ES-1.3 until directed to return to the procedure in effect, and then go to FR-P.1.

D. Continue in ES-1.3 until the cold leg recirculation alignment is completed, and then transition to FR-P.1

ANSWER:

D. Continue in ES-1.3 until the cold leg recirculation alignment is completed, and then transition to FR-P.1

REFERENCE: ES-1.3, NOTE prior to Step 1.

JUSTIFICATION: Must complete steps 1-3 to ensure long term cooling by cold leg recirculation is established prior to transitioning to other FRPs.

OBJECTIVE: EOU (1733)

K/A: 011 K3.12, Actions Contained in EOPs 4.4

Modified Exam Bank 2577

RO 12

The LOCA outside containment procedure (ECA-1.2) is entered based on:

A. Inadequate inventory of the containment sump to allow shiftover to cold leg recirculation.

B. SI actuation with indications that the RCS is intact, SG tubes are intact, and SG boundaries are intact.

C. Abnormal radiation level in the auxiliary building or ESF building with SI actuation.

D. Abnormal radiation levels in the spent fuel building with SI actuation.

ANSWER:

C. Abnormal radiation level in the auxiliary building or ESF building with SI actuation.

REFERENCES:

JUSTIFICATION: Symptom which requires entry into ECA-1.2 from E-1/E-0.

"A" is incorrect - there are symptoms of a Loss of Emergency Coolant Recirculation.

"B" is incorrect - they are possible indications for an inadvertent SI signal or extremely small break.

"D" is incorrect - because no safeguards release paths or components are located in the spent fuel building.

OBJECTIVES: A120 IC

K/A: W/E 04 EK 1.1 - 3.5 Components, Functions, and capacities.

Modified question 2579

RO 13

The following conditions exist:

- A steam generator tube rupture has occurred.
- The operators are presently at step 4 of E-3 "Steam Generator Tube Rupture" verifying ruptured SG level greater than 6% (42% adverse) level.

Which one of the following describes the basis for maintaining the ruptured SG level greater than 6% (42% adverse) level after the SG is isolated?

A. Provides filtering of the element iodine present in the ruptured steam generator.

B. Maintain a thermal stratification layer over the ruptured SG U-tubes, allowing for depressurization of the RCS to the ruptured SG pressure.

C. Maintain an adequate heat sink available in the ruptured SG.

D. Minimize stresses on the SG U-tube bends to prevent subsequent failures when the RCS is cooled down and depressurized.

ANSWER:

B. Maintains a thermal stratification layer over the ruptured SG U-tubes, allowing for depressurization of the RCS to the ruptured SG pressure.

REFERENCE:

JUSTIFICATION: WOG ERG background document for E-3 "Steam Generator Tube Rupture" states the reason for a minimum level in the SG is to maintain a thermal stratification layer over the top of the SG U-tubes. This will insulate the affected SG steam bubble and prevent the affected SG from depressurizing during the RCS cooldown. This makes selection 'B' correct.

K/A: 037K3.07 Actions in EOPs

OBJECTIVE: E0303C

COMMENTS: SRO #77, 95 NRC Exam (modified distracter D)

RO 14

The plant is at 100% power.

A loss of DC bus 5 occurs.

Which of the following describes the expected plant response?

A. Turbine Stop valves close due to low ETS pressure, the plant trips on Low-Low Steam Generator level or $OT\Delta T$.

B. ETS pressure switches in the Turbine Trip/Reactor Trip circuit deenergize causing a reactor trip.

C. ETS oil pressure is dumped causing the turbine to trip, which causes a reactor trip.

D. Turbine stop valves close due to low ETS pressure, causing a turbine trip on reverse power, which causes a reactor trip.

ANSWER:

B. ETS pressure switches in the Turbine Trip/Reactor Trip circuit deenergize causing a reactor trip.

REFERENCE: AOP 3563, Attachment E

JUSTIFICATION: ETS pressure is powered from 301C-1A6, from battery bus 5

OBJECTIVES: A6302C

K/A: 058 A2.03 Impact on Indications

97 Exam 7(modified)

RO 15

A loss of off-site power has occurred.

While the Emergency Generator Loading Sequencer (EGLS) is in the process of completing the stepping sequence, an automatic safety injection occurs.

Which of the following describes what occurs upon initiation of the Safety Injection?

A. Sequencer resets to step 0, Emergency Diesels remain running, Emergency Diesel output breakers remain closed, all previously running loads remain running unless not required for the SI and the sequencers sequence the SI loads.

B. Emergency Diesel output breakers open to strip the bus, sequencer reset to step 0, "SI" signal causes the Emergency Diesel output breakers to close and the sequencers to restart all the loads in the SI/LOP stepping sequence.

C. Sequencers stop at the step in progress, Emergency Diesels remain running, Emergency Diesel output breakers remain closed, the EGLS' start any SI loads that should be running and then resumes sequencing at the steps in progress in the SI/LOP mode.

D. Emergency Diesel output breakers open to strip the bus, Emergency Diesels remain running but the breaker will not close for at least 6.8 seconds. The sequencers reset to step 0 in the SI/LOP mode and will restart the entire sequence when 6.8 seconds have elapsed and the "second" SI/LOP is sensed by the sequencers.

ANSWER:

A. Sequencer resets to step 0, Emergency Diesels remain running, Emergency Diesel output breakers remain closed, all previously running loads remain running unless not required for the SI and the sequencers sequence the SI loads.

REFERENCES: LSK 24-9A

JUSTIFICATION: "B" & "D" are incorrect because the diesel output breaker does not open. "C" is incorrect because the sequencer resets to time 0. "A" is correct, the sequencer resets, any loads not needed for the m ode, such as RPCCW pumps for CDA would be tripped, any SI loads would be started.

OBJECTIVES: EDS05D (f)

K/A: APE056 A2.49 Component Capacity, Functions

97 Exam 8

RO 16

The control room is responding to an overpressure condition in "B" SG. SG pressure is currently 1250 psig and no safety valves or atmospheric relief valves are open.

In order to depressurize the SG as fast as possible, in accordance with FR-H.2, the operators should:

A. Maximize blowdown flow from "B" SG.

B. Feed "B" SG to cooldown the SG and lower the pressure.

C. Dump steam from the "B" SG to lower the pressure.

D. Dump steam from the other SG's to cool down the RCS to depressurize the "B" SG.

ANSWER:

C. Dump steam from the "B" SG to lower the pressure.

REFERENCE:

JUSTIFICATION: The quickest way to remove heat from the SG is to dump steam. (C is correct)

A cannot remove much heat. Increasing blowdown is a possible action for a high SG level not pressure.

Feed will cool the SG but procedure caution against feeding SG until steam heat removal path is available. (B is incorrect).

D cannot remove as much heat as C. D is the second alternative in accordance with FR-H.2 to depressurize the SG.

OBJECTIVE:

K/A: W/E 13 K 1.1 Components, capacity and functions.

New Question

RO 17

Given the following conditions:

- MP3 is operating at 80% power.
- Rod control is in automatic.
- Rods are at 225 on Bank D.
- The controlling <u>Tref</u> signal fails to a value of 575°F.

Control rod speed will indicate a speed of ________ steps per minute, and the ________ indication will be lit.

A. 48, out

B. 72, out

C. 48, in

D. 72, in

ANSWER:

D. 72, in

REFERENCE:

JUSTIFICATION:

Tav (80%) = 557 + 24 = 581
Tref = 575
Tav - Tref = 6°F
Max rod speed at 72 spm
(d is correct)

A is incorrect - wrong speed / direction
B is incorrect - wrong direction
C is incorrect - wrong speed

OBJECTIVE: A5202C; ROD02C (f.6)

K/A: 001 K6.02 Sensor feed to RCS

Modified 382

RO 18

The analysis for Reactor Coolant Pump locked rotor loss of flow accident predicts the initial trend over the first few seconds is:

A. Both DNBR and Pressurizer pressure increase.

B. Both DNBR and Pressurizer pressure decrease.

C. DNBR increases, Pressurizer pressure decreases.

D. DNBR decreases, Pressurizer pressure increases.

ANSWER:

D. DNBR decreases, Pressurizer pressure increases.

REFERENCE: FSAR Chapter 14, Decrease in Reactor Coolant System Flowrate (MCORE04)

JUSTIFICATION: Loss of flow causes the DNBR to DECREASE (closer to DNB), the loss of flow also causes the RCS to heat up as less heat is removed, and pressure will increase ("D" is correct).

OBJECTIVE: RCS08C; MCORE04

K/A: 003 K 5.01 - RCS Flow/Core & RCS pressure

97 EOP Exam
Question 2509

RO 19

Which one of the following is the reason that RCP #1 seal leakoff is isolated at RCS pressures below 125 psia?

A. Leakoff flow decreases at low pressures so #1 seal leakoff is isolated to force more flow through #2 seal.

B. Backflow from the VCT through the seal leakoff line could flush contaminants into the seals.

C. Controlled leakage limits may be exceeded due to excessive seal injection flow at low pressures.

D. Leakoff flow instruments are not accurate at low pressures and excessive leakoff could go undetected.

ANSWER:

B. Backflow from the VCT through the seal leakoff line could flush contaminants into the seals.

REFERENCE: Reactor Coolant Pump Text

JUSTIFICATION: The VCT may be at higher pressure than the RCS and backflow through the seal leakoff line may result. This flow is not subject to filtration (no seal injection filters on this line) so VCT contaminants may be introduced into the seals and result in mechanical damage when the pump is restarted.

K/A: 003 K1.03, RCP Seals 3.3/3.6

OBJECTIVE: RCP06C

95 NRC exam

RO 20

What is the purpose of the interlock requiring the Letdown isolations (3CHS*LCV459/460) to be open prior to opening the orifice isolations (3CHS*AV8149A-C)?

A. To prevent flashing downstream of the Letdown pressure control valve (3CHS-PCV131).

B. To prevent damage to the Letdown Regenerative Heat Exchanger shell due to pressure spikes.

C. To prevent excessive Letdown Heat Exchanger temperatures overloading the RPCCW system.

D. To prevent the letdown relief valve from lifting due to the heatup of the water between 3CHS*LCV459/460 and 3CHS*AV8149A-C.

ANSWER:

B. To prevent damage to the Letdown Regenerative Heat Exchanger shell due to pressure spikes.

REFERENCE: NSSS TEXT, VOL 1; JPM28

JUSTIFICATION: The interlock which prevents opening or closing the letdown isolation valves unless the orifice isolation valves are closed <u>prevents damage to the shell side of the regenerative heat exchanger due to pressure spikes.</u>

If LCV459 and LCV460 were opened with the orifice isolation valves open, the hot letdown flow would depressurize to VCT pressure as LCV459 and LCV460 left their closed seats and acted like throttle valves, and the water would flash to steam. This would cause a pressure spike.

If LCV459 and LCV460 closed with the orifice isolations open, the hot letdown water between the letdown isolation valves and the regenerative heat exchanger would again depressurize and flash.

OBJECTIVES: CHS04C (a1,a2)

K/A: 004K4.15 - Interlocks with letdown orifice isolation valves

Modified Bank 818

RO 21

Which of the following is a reason for placing the control switches for the charging pumps in pull-to-lock during a Loss of all AC Power?

A. To prevent the injection of cold seal injection water into the RCP seal packages when an emergency bus is restored, possibly damaging the seal packages or bowing the RCP pump shaft.

B. To prevent PTS by thermal shocking the reactor vessel downcomer with cold, high pressure Safety Injection water when an emergency bus is restored after the SGs are depressurized.

C. To prevent thermal stressing the charging loop penetrations by injecting cold, safety injection water into the RCS possibly creating an unisolable LOCA.

D. To prevent a pressurizer overfill situation when power is restored, since FCV-121 will be full open and the pressurizer will be empty due to the cooldown when the SGs were depressurized.

ANSWER:

A. To prevent the injection of cold seal injection water into the RCP seal packages when an emergency bus is restored, possibly damaging the seal packages or bowing the RCP pump shaft.

REFERENCE: ECA-0.0 Step 6 basis

JUSTIFICATION: Defeating automatic loading of Charging/SI pumps functions to protect RCP from damage when AC power is restored. This action prevents the automatic delivery of relatively cold seal injection flow into the RCP number 1 seal chamber and shaft are which has the potential for thermal shock and subsequent damage to the RCP seals and shaft.

OBJECTIVES: CHS06C (c.1); A0003C

K/A: 004 K2.02, Charging Pump Power Supplies

Test item 690 modified

RO 22

The plant is operating normally at 100% power.

Which of the following would be a result of the Letdown Pressure Transmitter (3CHS*PT131) failing LOW?

A. The Regenerative Heat Exchanger Outlet Temperature increases above its original value.

B. The Letdown Heat Exchanger Outlet temperature increases above its original value.

C. The CDTT level increases.

D. The PRT level increases.

ANSWER:

D. The PRT level increases.

REFERENCE: P&ID 104A, 102F; CVCS Text

JUSTIFICATION: Indicated pressure decrease = PCV131 close = actual pressure increases = relief lifts to the PRT.

A is incorrect. When PT 131 fails low, letdown pressure control valve PCV*131 will close. Regenerative heat exchanger outlet temperature will decrease because more heat is being removed by the charging flow. When the letdown relief opens, the temperature will increase slightly, but it will not increase above the original value.

B is incorrect. Letdown flow will decrease. The initial heat exchanger outlet temperature will decrease and the RPCCW valve will close.

C is incorrect. The letdown relief lifts to the PRT not the CDTT.

OBJECTIVE: CHS04C (a.6, a.7)

K/A: 004 A4.05, Letdown Pressure/Temp. Control Valves 3.6

96 NSSS Exam ID 2213

RO 23

PLANT CONDITIONS:

- Plant is at 100% power
- Pressurizer pressure channel, PT-455, failed high three (3) hours ago.
- The operating crew carried out the actions of AOP 3571, Instrument Failure Response and removed the channel from service.
- Channel PT-457 is now the controlling channel.
- All systems have been returned to automatic control

A loss of 120 VAC vital instrument panel VIAC 3 has just occurred.

Which of the following describes the impact on the plant of the loss of VIAC 3?

A. The plant will remain at 100% power. However the automatic actuation of the PORVs has been lost.

B. A safety injection will have occurred due to low pressurizer pressure logic coincidence being met.

C. One PORV will open due to a high pressure signal and this will eventually lead to a safety injection on low pressurizer pressure.

D. The master pressure controller will cause the pressurizer control heaters to go to minimum output and close the spray valves.

ANSWER:

B. A safety injection will have occurred due to low pressurizer pressure logic coincidence being met.

REFERENCES: AOP 3571, Functional sheets 12 and 12a

JUSTIFICATION: The operators will have tripped the bistables associated with the failure of the pressurizer pressure channel. This will include the low pressure SI bistable. When the vital AC bus is lost a second low pressurizer pressure bistable will be received and an SI will occur. The

OBJECTIVES: RPS07C

K/A: 013 A2.04 Loss of instrument bus

97 Exam 8

RO 24

A Safety Injection Signal occurs while VIAC 2 is deenergized.

How will the Train "B" ESF equipment respond?

A. Diesel Generator "B" starts, Bus 34D is stripped and Train B ESF loads are immediately loaded onto the bus.

B. Diesel Generator "B" starts, Bus 34D is not stripped, and the Train B ESF loads do not start.

C. Diesel Generator "B" does not start, Bus 34D is not stripped, and the Train B ESF loads do not start.

D. Diesel Generator "B" does not start, Bus 34D is stripped, and the Train B ESF loads are immediately loaded onto the bus.

ANSWER:

C. Diesel Generator "B" does not start, Bus 34D is not stripped, and the Train B ESF loads do not start.

REFERENCE: AOP 3564, Caution prior to Step 6

JUSTIFICATION: The caution prior to Step 6 of AOP 3564 states that:

> "Loss of VIAC 1(2) results in diesel generator A(B) being de-energized. If an ESF actuation takes place during this condition, the following items will *not* occur automatically.
>
> - The A(B) diesel will not start (except LOP)
> - Loads will not be stripped from the emergency busses
> - A(B) train loads will not start"

Distracter "C" is therefore correct. Distracters "A" & "B" are incorrect because they state the "B" EDG will start even though the question does not assume an LOP is present. Distracter "D" is incorrect because loads will not be stripped from Bus 34D.

OBJECTIVES: A64764D (1); A64764D (3)

K/A: 013 A3.02 Actuation of safeguards equipment

97 Mitcore Exam
2370

RO 25

Given the following conditions:

- The crew is performing a reactor startup.
- The RO has just pulled the control rods several steps and is waiting for source range counts to stabilize.

Assuming the reactor is very close, but not yet critical, source range counts should:

A. Stop increasing and stabilize immediately, with SUR decreasing to zero.

B. Increase at a constant rate with a constant, positive SUR.

C. Continue to increase, but at a slower rate, with SUR stabilizing at a lower positive value.

D. Continue to increase for a short period of time, then plateau, with SUR decreasing to zero.

ANSWER:

D. Continue to increase for a short period of time, then plateau, with SUR decreasing to zero.

REFERENCE: Reactor startup procedure (op 3202) and fundamentals

JUSTIFICATION: A is incorrect. Doesn't consider the effects of subcritical multiplication and its impact as criticality is approached.

B is incorrect. The reactor is still subcritical.

C is incorrect because SUR will decrease to zero, and count will stabilize at higher volume.

D is correct.

OBJECTIVE: 60202C

K/A: 015 K5.05, Criticality and its indications 4.1

New Question

RO 26

The plant is operating at 95% power. The Balance of Plant (BOP) operator inadvertently partially opens the Low Pressure Feedwater Heater Bypass Valve (CNM-MOV88).

How will **opening the valve** affect plant cycle efficiency?

A. Decrease since more energy from the reactor is required due to feedwater entering the steam generator at a lower temperature.

B. Increase due to less work required to pump the water through the low pressure feedwater heaters.

C. Decrease because mass flowrate entering the steam generators has increased.

D. Increase due to an increase in feedwater temperature resulting in more efficient high pressure feedwater heater performance.

ANSWER:

A. Decrease since more energy from the reactor is required due to feedwater entering the steam generator at a lower temperature.

REFERENCE: Westinghouse heat transfer fundamentals chapter 7, Mollier diagram.

JUSTIFICATION: Shutting the bypass valve will result in an increase in feedwater temperature due to more flow thru the feedwater heaters to be preheated an no "cold" feedwater bypassing the heater strings. As a result, feedwater will enter the steam generators at a higher temperature. The enthalpy change of the working fluid across the steam generator is smaller therefore, the heat transferred into the system (Q_s) is smaller. A smaller heat addition results in increased cycle efficiency.

OBJECTIVES: FWH06C; FWH07C

K/A: 056 A212, Heater String Bypass

SOURCE: Modified exam item 127

RO 27

Event Sequence:

0900 A turbine trip/reactor trip occurs

0905 A loss of offsite power occurs
A loss of all main feedwater occurs
Auxiliary feedwater initiates with approximately 1200 GPM total flow

0915 Narrow range level in the "C" SG is 6% and increasing

0918 Auxiliary feedwater total flow continues at approximately 1200 GPM

What are the consequences of not throttling back on AFW flow to all of the SGs at this time?

A. SG levels will increase until SG overfill protection isolates AFW flow to all SGs.

B. Full AFW flow will cause the RCS to cooldown and depressurize. Letdown will isolate and an SI will occur.

C. Full AFW flow will cause a plant cooldown until low-low Tav is reached and the feedwater isolation signal stops AFW flow to all SGs.

D. The SGs will cooldown and depressurize. This will result in a safety injection on the high steam pressure rate.

ANSWER:

B. Full AFW flow will cause the RCS to cooldown and depressurize. Letdown will isolate and an SI will occur.

JUSTIFICATION: A is incorrect because the high-high signal trips the turbine, which is already tripped and provides no isolation of AFW.

C is incorrect because a feedwater isolation signal has no affect on AFW flow.

D is incorrect because the high steam pressure rate will result in a main steamline isolation, not an SI.

OBJECTIVE: EO003C

K/A: 059 K3.04, Effects on RCS Temperature

New Question

RO 28

Given the following conditions:

- MP3 was operating at 100% power.
- The station experienced a total loss of all AC power.
- The crew is currently depressurizing the SGs to 260 psig in ECA-0.0.

As SG pressures are lowered, available AFW flow will:

A. Decrease to zero (0) gallons total flow

B. Drop to 100 GPM per steam generator

C. Remain above a total of 530 GPM

D. Remain near the original flow of approximately 1150 GPM

ANSWER:

C. Remain above 530 GPM

REFERENCE:

JUSTIFICATION: The Terry Driven AFP is sized to remove the decay heat and cooldown the plant to go on RHR at 250 psig in RCS (SG Press ~100 psig)

The Terry Turbine will provide adequate heat sink for reactor and be above red path value. (C is correct)

A & B are incorrect. B is the value of flow due to the flow restricting venturi for feedline break protection (incorrect). A is from number for SG pressure 150 when RHR is put in service, (A is incorrect).

D is incorrect because flow decreases when pressure drops especially when SG pressure drops < 600 psig (FSAR).

OBJECTIVES: FWA03D (6)

K/A: 059 A1.02 Steam generator pressure effects on terry turbine 3.6

New Question

RO 29

Given the following conditions:

- Unit is operating at 100% power.
- A Hi alarm is received on the containment gaseous radiation monitor.
- Charging flow is 135 GPM.
- Seal return flow is 3.2, 2.8, 3.3, 3.4 GPM (A, B, C, D)
- Seal injection flow is 9.8, 9.0, 9.2, 9.0 GPM (A, B, C, D)
- Letdown flow is 75 GPM.
- RCS Tave is on program.
- Containment sump level is increasing.
- PRZR level is on program.
- VCT level is 30% and decreasing.
- PDTT level is stable at 25%.

Based on the above information, the current leak rate is ________ GPM.

A. 47.3

B. 60.0

C. 84.3

D. 94.0

ANSWER:

C. 84.3

REFERENCE:

JUSTIFICATION: Leak Rate = Charging + Seal Injection - (Letdown + Seal Return)
Leak Rate = 135 + 37 - (75 + 12.7)
Leak Rate = 172 - 87.7
Leak Rate = 84.3

OBJECTIVES: A5502C

K/A: 002 A2.01 RCS inventory balance/loss of inventory 3.9

Modified Exam Item 659

RO 30

INITIAL CONDITIONS:

- Reactor power is 60%
- Loop 1 Delta-T indicates LOW
- Loop 1 Tave indicates HIGH

Which of the following Loop 1 RTD failures coulo cause these indications?

A. T-hot failed HIGH.

B. T-cold failed LOW.

C. T-cold failed HIGH.

D. T-hot failed LOW.

ANSWER:

C. T-cold failed HIGH.

REFEREN

JUSTIFICA N: $Tave = \frac{T_H + T_C}{2}$ $\Delta T = T_H - T_C$

If T_C fails high, Tav indicates high ΔT indicates low, (c is correct)

A is incorrect because ΔT would indicate high.

D is incorrect because ΔT would indicate high.

D is incorrect Tav could indicate low.

OBJECTIVES: RCS02C (i)

K/A: 002K 5.12 Relation of temperature indications

Modified item 1468

RO 31

Given the following conditions:

A small break LOCA has occurred.
Fuel failure occurred during rapid downpower prior to trip. Crew is currently in ES-1.2 "Post LOCA Cooldown and Depressurization."

RCS pressure is 1600 psia and decreasing
Wide range Tc's are 550°F and decreasing
Wide range Th's are 560°F and decreasing
CET's are 565°F and appear stable.
Containment pressure is 20 psia and decreasing.
Containment temperature is 185°F and decreasing.
SG levels are being maintained at 35%.
Pressurizer level is 25% and stable

Based upon the above indications, SI flow:

A. Can be reduced because CET's are stable.

B. Cannot be reduced since subcooling is inadequate.

C. Can be reduced because the SG's are an adequate heat sink.

D. Cannot be reduced since containment pressure is still greater than the SI actuation setpoint.

ANSWER:

B. Cannot be reduced since subcooling is inadequate.

JUSTIFICATION: B is correct. Subcooling is less than 115°F. Thus SI flow cannot be reduced.

A is incorrect because SI reduction criteria is a function of subcooling and pressurizer level not CET's alone.

C is incorrect. SG's levels aren't greater than those required for heat removal under adverse containment conditions.

D is incorrect. SI flow can be reduced under circumstances when containment pressure has not been restored to less than the actuation setpoint (18 psia)

OBJECTIVES: S1202C; S1203C

K/A: 006 A1.19 Effects of subcooling 4.0

New Question

RO 32

The plant is in MODE 5 with Tave approximately 185°F. The crew is in the process of drawing a bubble in the pressurizer. Pressurizer level has just started to come on scale.

A complete loss of instrument air occurs.

Which of the following describes plant response with NO operator action?

A. The plant slowly depressurizes.

B. RCS pressure will increase to the COPS setpoint.

C. RCS pressure will increase to safety valve setpoint.

D. The plant will maintain pressure until pressurizer heaters trip.

ANSWER:

B. RCS pressure will increase to the COPS setpoint.

REFERENCE: OP 3201 Attachment 1

JUSTIFICATION: Letdown isolates, spray valves fail closed. COPs is armed. Setpoint for COPS at the current temperature is ≈ 400/450 psia. Pressure will increase until COPs open

OBJECTIVES: RCS08C

K/A: 010 K4.03 Overpressure Control

97 Exam 9
2184

RO 33

Given the following conditions:

- MP3 is holing at 9% to complete turbine shell warming.
- PZR pressure channel II bistables are in trip because channel previously drifted high.
- SG level is being controlled with the FRV bypass valves, which are in automatic.
- FRVs are isolated.
- RCS temperature is being controlled by the steam dumps.

WHICH ONE of the following events would, if no corrective actions were taken, result in a reactor trip?

A. Running charging pump trips.

B. A second PZR pressure channel fails to 1900 psia.

C. Loss of power to the FRV bypass solenoids.

D. EHC piping rupture.

ANSWER:

C. Loss of power to the FRV bypass solenoids.

REFERENCE: Tech Specs, Functional sheets 6 & 7

JUSTIFICATION: The loss of power to the FRV bypass solenoids will result in them failing closed. This will result in a reactor trip on SG LO-LO level, a trip that is not bypassed by P-7.

Power < P-7 (10%) blocks or bypasses the following trip:

Turbine trip - Rx trip bypassed by P-9 (D is incorrect)
Pressurizer low press Rx trip (B is incorrect)
High Pressurizer level

Running charging pump trip will cause auto start of standby pump and will not cause a trip. RPS trip is off high pressurizer level (A is incorrect)

An SI signal, causing a reactor trip, will not be generated because the setpoint is 1892 psia.

OBJECTIVES: FWS04C (p)

K/A: 012 K6.10 Permissive circuits 3.3

New Question

RO 34

Pressurizer pressure instrument PT-455 has failed high. The crew has tripped the following bistables:

- High pressurizer pressure reactor trip
- Low Pressurizer Pressure
- P-11
- Pressurizer pressure low pressure SI
- Low pressurizer pressure PORV logic

What additional bistables, if any, must still be tripped to complete AOP 3571, Instrument Failure Response?

A. No additional bistables must be tripped, all the bistables listed in AOP 3571 for the failed channel have been tripped.

B. OTΔT
C-3

C. OTΔT
OPΔT

D. OPΔT
C-3

ANSWER:

B. OTΔT
C-3

REFERENCE: AOP3571, attachment B

JUSTIFICATION: There are 7 bistables listed in the attachement, the crew has tripped 5. The remaining 2 are OTΔT and C-3, (Low Pressure Relief interlock)

K/A: 012 A3.05, RPS Channel Trips

OBJECTIVE: PPL08C

RO 35

Which of the following situations could cause a General Warning on the DRPI display?

A. One central control card differs from the other two.

B. One dropped rod at 100% power with normal rod alignment prior to the rod dropping.

C. Either Data "A" or Data "B" card inputs are invalid.

D. Two rods within the same bank deviate from one another by more than 12 steps.

ANSWER:

C. Either Data "A" or Data "B" card inputs are invalid.

REFERENCES: DRPI Technical Manual; AOP 3552; ARP MB4C

JUSTIFICATION: A is incorrect. A central control card failure will cause the control card failure alarm on the DRPI display and an RPI Non Urgent Alarm

B is incorrect. A dropped rod will result in "ONE ROD BOTTOM" annunciator, "ROD DEVIATION" alarm, and a rod bottom light on DRPI.

D is incorrect. For this situation a "ROD DEVIATION" alarm and "ROD POSITION DEVIATION" alarm would be generated.

C is correct. The general warning DRPI display LEDs are illuminated for Data "A" or Data "B" input being rejected or any cause for an DRPI urgent failure alarm which are:

- Loss of both A and B data or error in BOTH; or
- > 1 bit difference in A and B gray codes: or
- Combined data sum is > 38

OBJECTIVES: RPI07C

K/A: 014 A1.02, Control Board Indications

Exam Item: 48

RO 36

What will be the effect on Feed Regulating Valve 3FWS*FCV510 due to the controlling channel for 'A' Steam Generator level instantaneously failing high?

A. The feed regulating valve will start to close after a predetermined time due to the built in lag circuit of the controller.

B. The feed regulating valve will immediately start to close and then close at a speed determined by the magnitude of the error signal and the time the error signal is present.

C. The feed regulating valve will immediately start to close, then change its closing speed based on the time the error signal is present only.

D. The feed regulating valve will immediately start closing at a constant rate for a predetermined time, then begin opening at a rate determined by the actual level error.

ANSWER:

B. The feed regulating valve will immediately start to close and then close at a speed determined by the magnitude of the error signal and the time the error signal is present.

REFERENCE: Functional drawing 108D685, Instrumentational and Operational Analysis, Nuclear Energy TRNG. Module 7

JUSTIFICATION: The valve will close immediately without any lag. (A is incorrect)

The valve will close immediately at a rapid rate then close at rate dependent on the integral time constant. (C is incorrect).

Because the proportional and integral signals are additive, the rate is never constant and the valve is always closing due to the system being level dominant. (D is incorrect).

OBJECTIVES: FWS07C (a1)

K/A: 016 A3.01 Auto select/signals to control currently.

RO 37

Following a CDA signal, what is the purpose of the time delay associated with the start of the Recirculation Spray System (RSS) Pumps?

A. To allow the operators time to override the starting of the RSS pumps, on an inadvertent CDA signal, preventing an undesired spray down of Containment.

B. To allow time for the RHR pumps to trip off on low-low RWST level and prevent exceeding the heat removal capabilities of the service water system.

C. To be consistent with the sequencer time delays for a CDA/LOP situation.

D. To allow time for proper NPSH to the RSS pumps to be established.

ANSWER:

D. To allow time for proper NPSH to the RSS pumps to be established.

REFERENCE: FSAR Section 6.2.2.3

JUSTIFICATION: The 11 minutes allows CTMT sump level to increase with QSS water, and condensed RCS or MSS steam, preventing RSS pump starts prior to achieving and adequate net positive suction source.

OBJECTIVES: CDA06C (g.4); CDA03C (e)

K/A: 026 K1.01 Relationship with ECCS

Exam Item 2473

RO 38

Given the following conditions:

- The Unit has just synched on line and is ramping past 15% power.
- "B" reactor coolant pump trips.

Assuming no operator actions, "B" steam flow will ______ and "B" SG level will _______ (consider only the immediate effects).

A. Increase, increase

B. Increase, decrease

C. Decrease, increase

D. Decrease, decrease

ANSWER:

D. Decrease, decrease

REFERENCE:

JUSTIFICATION:	Steam flow will decrease as pressure drops and level will decrease, due to colder RCS when flow reverses.
OBJECTIVES:	RCS08C
K/A:	035 K5.03 Shrink and Swell

Modified Exam Item 1818

RO 39

The plant is at full power in a normal electric plant lineup.

The NSST breaker for bus 34A opens.

Which of the following would result in a slow transfer rather than a fast transfer?

A. The 34A-34C bus tie breaker opens at the same time as the NSST breaker.

B. The NSST "early actuation" signal is present.

C. An RSST output voltage of 3600 volts.

D. Bus 34A normal supply lockout relay failed to actuate.

ANSWER:

A. The 34A-34C bus tie breaker opens at the same time as the NSST breaker.

REFERENCE: LSK 24-3D

JUSTIFICATION: A is correct. The tie breaker being closed is a condition for fast transfer and the breaker being open is a condition for slow transfer.

B is incorrect. The early actuation signal is a requirement for fast transfer.

C is incorrect because low voltage on the RSST would prevent either transfer.

D is incorrect because the supply lockout relay will impact both transfer schemes.

OBJECTIVES: 4KV07C; 4KV06C

K/A: 062 K4.03 Interlocks, auto transfer

Modified bank item 2059

RO 40

In accordance with the Millstone 3 FSAR, which one of the following describes the expected battery capacities for a loss of DC on battery bus (301A-1/301B-1)?

A. 1 hour

B. 2 hours

C. 4 hours

D. 8 hours

ANSWER:

B. 2 hours

REFERENCES: Technical Specification 3/4.8.2 and FSAR

JUSTIFICATION: The battery capacity is assumed to be 2 hours. (TS 3/4.8.2) "B" is correct.

OBJECTIVES: 12501C; 12502C

K/A: 063 K4.03, Battery Charger Capacities

New Question

RO 41

The crew is in E-0, "Reactor Trip or Safety Injection", due to SI actuation

Prior to resetting the SI signal the operator depresses both Emergency STOP push-buttons on MB8 for the "B" Emergency Diesel Generator.

Which of the following describes the response to the "B" Emergency diesel Generator? (No further operator action occurs).

A. The diesel continues to run.

B. The diesel would stop and remain shutdown.

C. The diesel would stop but automatically restart after 140 seconds, and restart EGLS in the mode SI/LOP.

D. The diesel would stop and then immediately restart and restart EGLS in the mode SI/LOP.

ANSWER:

B. The diesel would stop and remain shutdown.

REFERENCE:	LSK 24-09.3A,B and J
JUSTIFICATION:	The emergency stop push-buttons energizes the shutdown relay which energizes the solenoid to cut fuel to the diesel, preventing a start. The diesel will remain shutdown. (B correct)
OBJECTIVES:	EDG06C; EDG04C
K/A:	064 K3.01, Impact of Auto Loading

Modified from 443

RO 42

The plant is in:

- Mode 3
- RCS is solid
- COP's is in auto and armed
- RCS is at 150°F and 340 psia
- SG temperatures are 200°F after natural circulation

Which of the following will cause an RCS overpressure transient in the RCS?

A. Loop A T_H fails low.

B. RCP B is restarted.

C. Letdown Pressure Control valve fails open.

D. Starting an RHR pump.

ANSWER:

B. RCP B is restarted.

REFERENCE:

JUSTIFICATION: "A" is incorrect because COPs valves are armed and won't open unless RCS pressure exceed setpoint. This failure will not cause any transient in RCS.

"B" is correct - starting "B" RCP is solid plant with COPS armed and a 50ΔT will cause a pressure increase and the COPs valves will open.

"C" is incorrect - letdown pressure control valve failing open will cause RCS pressure to decrease.

"D" is incorrect - starting the RHR pump should cause the RCS pressure to decrease slightly due to increase flow and heat removal in the RHR heat exchanger. PCV 113 will reposition in auto to maintain RCS pressure.

OBJECTIVES: RHR05C; RHR08C

K/A: 005 A2.02, Cold Overpressure Protection

New Question

RO 43

Maximum allowed RHR flow and the bases for this maximum during reduced inventory conditions are:

A. RHR flow limited to 3000 GPM to prevent exceeding the heat removal capabilities of the RPCCW system.

B. RHR flow limited to 1000 GPM to prevent loss of flow due to vortex entrainment of air at pump suction.

C. RHR flow limited to 1000 GPM to minimize potential inventory loss in the event of a leak.

D. RHR flow limited to 3000 GPM to prevent overloading the pump motor from operating near runout conditions.

ANSWER:

B. RHR flow limited to 1000 GPM to prevent loss of flow due to vortex entrainment of air at pump suction.

REFERENCES:

JUSTIFICATION: "B" is correct. EOP 3505 limits flow to 1000 GPM to minimize effects of vortexing.

"D" is incorrect. RHR flow is released from 3000 GPM to 1000 GPM as reduced inventory procedures are implemented.

"B" is incorrect because pressure limits flow from leaks not RHR flowrate.

"A" is incorrect because flow is limited to 1000 GPM and the RPCCW system is cabable of removing heat at higher then 3000 GPM flow rates.

OBJECTIVES: RHR05C

K/A: 005 A1.02 RHR flow rates.

New Question

RO 44

The plant has tripped and experienced a safety injection and "CIA".

Which of the following describes how the reactor plant chilled water system is affected?

A. Chilled water containment isolation valves close, the RPCCW to CDS cross-ties open, and all other loads outside containment isolate.

B. All loads outside containment isolate, the chilled water loads in containment are maintained unless a "CIB" actuates.

C. Chilled water containment isolation valves isolate, the RPCCW to CDS cross-ties open, flow to all other loads is maintained.

D. All loads outside containment isolate, the chilled water loads in containment are maintained unless a "CDA" actuates.

ANSWER:

C. Chilled water containment isolation valves isolate, RPCCW to CDS cross-ties open, flow to all other loads is maintained.

REFERENCE: P&ID 122B and 121B

JUSTIFICATION: CIA closes the containment valves and opens the RPCCW cross ties (C Correct)
CDA will not directly cause any valves to reposition. (D Incorrect)
CIB will cause the RPCCW supplies to CDS to close. (B Incorrect)
CIA will not cause a loss of all loads outside containment. (A Incorrect)

OBJECTIVES: CDS03C; CDS06C

K/A: 008 A3.05, Control/auto isolation valves

Bank item 1504

RO 45

The following plant conditions exist:

- A plant cooldown from 557°F has been started.
- The steam dumps are being utilized in the steam pressure mode.
- At 553°F all the steam dumps close.

Select the action which will allow continuation of the cooldown on steam dumps.

A. Take the "mode selector" switch to "reset".

B. Take bypass interlock selector switches to "bypass".

C. Take bypass interlock selector switches to "off/reset" position.

D. Take the steam pressure controller to manual to open the steam dump valves.

ANSWER:

B. Take bypass interlock selector switches to "bypass".

JUSTIFICATION: To bypass low low Tav interlock you must take both bypass interlock switches to bypass. ("B" is correct.)

Taking the mode switch to reset - resets the load rejection arming memory (C-7) only. ("A" is incorrect.)

Taking the bypass interlock select switches to off/reset position will block steam dump operation. ("C" is incorrect.)

Shifting steam pressure controller to manual doesn't override/bypass the block signal ("D" is incorrect.)

OBJECTIVES: SDS05C (c); SDS02C (g)

K/A: 041 A4.02, Use of cooldown valves

Modified from 353 (Confidence Weighted Question)

RO 46

Plant Conditions:

A & B SW pumps selected as lead pumps

C & D SW pumps selected as follow pumps

The following plant conditions exist:

- Reactor trip has occurred due to a loss of off-site power.
- The "A" EDG starts and immediately closes in to supply bus 34C
- The "B" EDG starts but its output breaker does not close automatically. The operator closes the output breaker after the diesel has been running for 30 secs.

On a board walkdown the expected following service water pump combinations should be running:

A. All 4 service water pumps should be running.

B. The A and D SW pumps should be running.

C. The C and D SW pumps should be running.

D. The A and B pumps should be running off their respective emergency diesels.

ANSWER:

D. The A and B pumps should be running off their respective emergency diesels.

REFERENCES:

JUSTIFICATION: The lead pumps should be running. The follow pumps only start if a lead pump fails to start. (A, B, & C are incorrect.)

OBJECTIVES: SWP04C (a); SWP06C (d)

K/A: 076 K4.02, Auto Start Features

Modified Exam Item 1997

RO 47

Initial Conditions:

- Unit is at 100% power
- All circulating water pumps are running
- It is August 20th and Long Island Sound average temperature is 71°F
- All heater drain pumps are running
- A & C TPCCW pumps are running
- B Stator liquid cooling pump is running

A "BUS 34A BUS DIFF" alarm occurs on main Board 8. The operators should:

A. Go to E-0 because the unit will trip on OTΔT due to the MSIVs closing due to low air pressure.

B. Reduce load in an attempt to maintain vacuum in the condenser.

C. Trip the reactor and go to E-0 because the main feedwater pumps tripped on low suction pressure.

D. Stabilize the unit after the turbine runback due to high stator water cooling temperatures.

ANSWER:

B. Reduce load in an attempt to maintain vacuum in the condenser.

JUSTIFICATION: A is incorrect. the MSIVs would close on loss of DC power to the solenoids or the loss of instrument air. DC power is not affected by the given AC power loss. The "B" TPCCW pump will auto start on low TPCCW pressure. Even if TPCCW was lost the air compressors would shift to the domestic water system as a backup.

C is incorrect because even though heater drain flow will be reduced and main feed pump suction will start to decrease, the condensate pumps will make up for the lost flow. As feed flow decreases, the feed regulating valves open, causing feed pressure to decrease which in turn causes main feed pump speed to increase. The unit will not trip on loss of the main feed pumps or loss of suction pressure.

D is incorrect because the stator water cooling pump has not been lost (B pump not powered by 34A). Additionally the unit can operate for up to one hour without stator water cooling or cooling to the stator cooling heat exchangers by TPCCW. The "B" TPCCW pump will start in auto on low TPCCW pressure.

B is correct. With only one circulating water pump running for each bay of the condenser, vacuum will start to decrease if the plant remains at 100% power.

OBJECTIVES: 4KV06C; 4KV07C

K/A: 062 K2.01 Loss of power to major loads

New Question

RO 48

The plant is in MODE 6. Core offload is in progress.

The alarm circuitry for one of the Spent Fuel Pool area monitors fails. I & C is investigating. No other operator actions have been taken.

Which of the following describes the required ACTION, if any, to be taken regarding the fuel movement in progress?

A. No ACTION required, all LCOs are satisfied, fuel movement may continue.

B. Fuel movement may continue for up to 4 hours while adjusting the setpoint to within the limit.

C. Fuel movement may continue for up to 30 days.

D. Fuel movement must be suspended until an appropriate portable continuous monitor with the same Alarm Setpoint is provided in the fuel storage pool area.

ANSWER:

D. Fuel movement must be suspended until an appropriate portable continuous monitor with the same Alarm Setpoint is provided in the fuel storage pool area.

REFERENCE: Technical Specification 3.3.3.1, ACTION 28

JUSTIFICATION: A incorrect, Tech Spec Minimum channels required is 2, currently only 1.

B incorrect, the setpoint is not the problem, the 4 hour limit does not apply.

C incorrect, ACTION required is to provide the backup or suspend fuel movement.

D correct, until the backup is provided, no fuel movement can occur, then the monitor must be returned to OPERABLE status in 30 days or suspend fuel movement.

K/A: 072 K3.02 Effects on fuel handling operations

OBJECTIVE: RMS05C, RMS07C, RMS08C

New Question

RO 49

Containment purge and exhaust are in operation.

Fuel Drop monitor, 3RMS*RE41goes into high alarm.

Which of the following describes the automatic response of the system to the alarm?

A. All four containment purge supply and exhaust valves close only.

B. All four containment purge supply and exhaust valves close and the running exhaust fan, 3HVR-FN4A or 4B stops.

C. One supply and one exhaust valve closes only.

D. One supply and one exhaust valve closes and the running exhaust fan, 3HVR-FN4A or 4B stops.

ANSWER:

C. One supply and exhaust valve closes only.

REFERENCE: LSK 22.27B

JUSTIFICATION: Purge isolation takes place if EITHER fuel drop monitor goes into alarm. (One monitor isolates the inside containment valves, the other monitor isolates the outside containment valves.) The fans do not get tripped. (C correct only)

OBJECTIVES: RMS05C; RMS07C; RMS08C

K/A: APE 061 A1.01, Automatic Actuation of ARM 3.6/3.6

modified from 398 and 2117

RO 50

Plant Conditions:

- An ATWS has occurred due to a fire in the switchgear room
- The reactor could not be tripped locally because the breakers are fused together
- Reactor power is 40% and decreasing
- Tav is increasing
- Pressure is being maintained by the PORVs cycling around 2350 psia

The PREFERRED sequence to add negative reactivity to shutdown the reactor is:

A. Perform an immediate boration, drive rods in auto or manual, initiate a safety injection

B. Drive rods in auto or manual, shift charging pump suction to the RWST, perform an immediate boration.

C. Perform an immediate boration, maximize charging flow, drive rods in auto or manual.

D. Shift charging pump suction to the RWST, drive rods in auto or manual, perform an immediate boration.

ANSWER:

C. Perform an immediate boration, maximize charging flow, drive rods in auto or manual.

REFERENCE: FR-S.1

JUSTIFICATION: The first 4 steps of FR-S.1 contain the preferred sequence for adding negative reactivity in an ATWS situation.

OBJECTIVES: FS102C

K/A: 029 EK3.12 Actions in EOP

New question

RO 51

The plant is in Mode 1. The following personnel are in the control room The CO who is the "Operator at the Controls", the US and the SM. The other CO is on a plant tour.

When, if at all, may the US be considered the "Operator at the Controls?"

A. If the "Operator at the Controls" must leave the red carpeted area (Operations Area).

B. If the "Operator at the Controls" is around back of the main control board acknowledging an annunciator.

C. If the Shift Manager is in the surveillance area and a proper turnover occurs.

D. The US cannot be considered the "Operator at the Controls".

ANSWER:

C. If the Shift Manager is in the surveillance area and a proper turnover occurs.

REFERENCE: COP-200.1 Section 1.7 & 1.6
COP-200.1 Attachments 5 & 6
Tech. Specs 6.2.2

JUSTIFICATION: "A" incorrect due to CO not restricted to Operations area (COP 200.1, Section 1.6)
"B" incorrect because CO allowed behind boards to acknowledge main control board annunciators and is still the "Operator at the Controls"(COP 200.1, Section 1.6)
"C" correct. With the SM in the surveillance area and with a proper relief the US can become the "Operator at the controls"
"D" incorrect. Section 1.7 of COP200.1 states 1 licensed operator in the surveillance area at all times in addition to the SM or the US in the control room. If other personnel are to be considered in the surveillance area, they shall meet the normal relief requirements.

OBJECTIVES: NAD407

K/A: 2.1.2, Operator Responsibilities

Modified question 2189

RO 52

Safety-related plant equipment is known to have been operated in a manner which had the potential to damage the equipment.

Which of the following describes an action which must be taken in accordance with COP 200.1 CONDUCT OF OPERATIONS?

A. Shift Manager shall notify the Duty Officer and initiate a CR.

B. A notification must be made to the NRC within twenty-four hours of the event.

C. The Unit Supervisor shall initiate a priority 1 AWO for maintenance to investigate.

D. The Operations Manager shall notify the Technical Services Engineering Manager and cognizant system engineer.

ANSWER:

A. Shift Manager shall notify the Duty Officer and initiate a CR.

REFERENCE: C OP 200.1

JUSTIFICATION: If plant equipment is known to have been operated in a manner which had the potential to damage the equipment, the following actions shall be taken:

Operator shall inform the US of the problem.
US shall direct equipment or plant be placed in a safe condition.
SM shall describe the problem in the SM log.

If potentially damaged equipment is safety-related, the following additional actions shall be taken:

SM/US shall check applicability of Technical Specifications or Technical Requirements.
SM shall inform the Duty Officer.
SM shall initiate an CR.
SM shall initiate appropriate investigative action to determine status of potentially damaged equipment.

OBJECTIVES: NAD413

K/A: 2.1.1
2.1.7, Evaluate performance/judgment 3.7/4.4

Question #3093

RO 53

Which of the following is a difference between a dual verification and independent verification?

A. An Independent Verification can be performed by the same individual performing the initial verification. Dual Verification requires two individuals.

B. An Independent Verification is performed prior to the task being performed. Dual Verification is performed upon completion of the task.

C. Dual Verification can be performed using the "time and distance" method Independent Verification is usually performed concurrently with the initial verifier.

D. Dual Verification is performed concurrently with the task. Independent Verification is performed after the task or evolution has been completed.

ANSWER:

D. Dual Verification is performed concurrently with the task. Independent Verification is performed after the task or evolution has been completed.

REFERENCE: WC-6, Attachment 2

JUSTIFICATION: DUAL VERIFICATION: Dual verification is performed concurrently with the task. In general, **dual verification is performed when an action could result in an immediate threat to safe and reliable plant operation.** For dual verification, concurrently means the "performer" and the "verifier" together determine and agree the work location and component are correct for the specified action. And, the task, to the best of their knowledge, will result in the desired outcome. The person performing the task and the person performing the verification both must positively identify the component, determine the actual and required position or state, and agree on the method to be used prior to the action taking place. The verifier essentially completes the verification of the intended action, before the action is undertaken and then witnesses the task performed. It should be noted that if a specific task requires a dual verification during its performance, the need for a separate independent verification of the same evolution is not required. [v Comm 3.1]

INDEPENDENT VERIFICATION: **Independent verification is performed after a task or evolution has been completed to ensure it has been performed in the correct location, on the correct equipment, or the desired results have been obtained.** An independent verification is also performed periodically to ensure systems, equipment, components, etc. are still in the condition they were left following their last manipulation or verification. To satisfy the requirements for an independent verification, when plant conditions or the situation allows, a good rule of thumb to ensure real independence, is to apply the "time and distance" method. This requires the independent verifier to not visually observe the person who initially perform the task. Conducting the verification in this manner will alleviate the possibility that the performer goes to the wrong item or place and the independent verifier, watching the performer, simply goes to the same (wrong) location and verifies the performer's actions were completed correctly (at the wrong location!). In lieu of using the time and distance technique, as with all verifications, using attentiveness and attention to detail during their performance will prevent inadvertent problems. It is the responsibility of the verifier to ensure he or she is in the correct location, checking the required equipment or components, and determining if they meet the specified acceptance criteria.

OBJECTIVES: NAD613

K/A: 2.1.29, Conduct/verify valve lineup

Question #3268

RO 54

When is a second Main Condensate Pump required to be operating?

A. When pumping the Main Condenser Hotwell.

B. When a Main Feed Pump is operating.

C. When the Main Circulating Water Pumps are operating.

D. Anytime steam is being released from the Steam Generators.

ANSWER:

B. When a Main Feed Pump is operating.

REFERENCE: OP 3319A section 4.1

JUSTIFICATION: Pumping the hotwell requires only one condensate pump and is only performed when the plant is in a shutdown condition and in long recycle (no MFPs operating) - ('A' incorrect)

Operation of the Circ pumps requires that tube sheet seal water be applied. This can be supplied from the CNS system during plant shutdown conditions and does not require use of the condensate pumps ('C' incorrect)

Steam may be released from the S/Gs to atmosphere with feed being supplied.

OP 3319A section 4.1 states "A minimum of two condensate pumps must be running while feeding steam generators with main feed pumps." This requires that a 2nd condensate pump be operating any time a main feed pump is operating ('B' correct)

OBJECTIVES: CNM06C (a)

K/A: 2.1.32, System limitations/precautions

Question #2964

RO 55

The following plant conditions exist:

- The plant was operating at 100% power when a large LOCA occurs.
- A Containment Depressurization Actuation Signal (CDA) is generated due to high containment pressure.
- All safeguards equipment operates as designed except the B EDG fails to auto start and cannot be started.

15 minutes later, while performing E-0, Reactor Trip or Safety Injection, all offsite power is lost.

Approximately 2 minutes later, the RO checks power availability to the Safeguards equipment.

What should be the status of the RSS pumps?

A. All four pumps should be running.

B. Only RSS pumps "A" and "C" should be running.

C. Only RSS pump "A" should be running.

D. None of the pumps should be running.

ANSWER:

B. Only RSS pumps "A" and "C" should be running.

REFERENCE: LSKs 24-9.4A, 24-9.4B, 24-9.4Q, and 27-11J

JUSTIFICATION: The CDA signal starts the 11 minute timer to start the pumps. Once started, if an LOP occurs, the pumps are restarted by the sequencer after 60 seconds.

A is incorrect, because the "B" EDG is not running

C is incorrect. Both "A" train pumps will start. Only the "A" pump would start if in the SI/Recirc mode.

D is incorrect because the "A" train pumps will start.

OBJECTIVES: CDA06C (1); CDA07C

K/A: 103 K1.08 SIS/CDA including reset

Bank item

RO 56

A Safety Injection has occurred and the crew was conducting a brief at the end of Step 14 of E-0 when a complete loss of Off-Site power occurs.

The Emergency Diesel Generators start, their output breakers close to restore power to the vital busses, and the sequencers complete their sequencing on of loads.

What is the status of Containment Air Recirculation (CAR) fans?

A. All the CAR fans will be running.

B. The "A" and "B" CAR fans will be running.

C. Only the "C" CAR fan will be running.

D. None of the CAR fans will be running.

ANSWER:

B. The "A" and "B" CAR fans will be running.

REFERENCES: LSK 22-27.C, LSK 24-9.4a

JUSTIFICATION: The crew has stopped the "C" CAR fan at Step 12 of E-0 ("A" and "C" wrong).

The "A" and "B" fans will start on the SI signal, and when the LOP occurs, the "A" and "B" fans will trip and will sequence back on at 39 seconds. ("B" correct, "C" and "D" wrong.).

OBJECTIVE: CVS03C (b.4)

K/A: 022 A3.01 Initiation of Safeguards

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RO 57

During an uncontrolled rod withdraw from 175 steps on "D" bank, the final steady state actual reactor power will ________, and RCS Tave will ________. (Assume no operator action/reactor does not trip)

A. Increase, Increase

B. Increase, remain the same

C. Remain the same, decrease

D. Remain the same, increase

ANSWER:

D. Remain the same, increase

REFERENCE:

JUSTIFICATION: The p reactivity addition of the control rods will cause power and Tave to initially increase. The increasing Tav will drive power down to approximately the initial level. The end result approximately at the same power level with an elevated temperature. The temperature increase will feedback negative reactivity to offset the positive reactivity originally added by rod motion.

OBJECTIVES: ROD06C (a)

K/A: 001 K1.03, Relationship of reactivity and Rx power to rod movement 3.9

New Question

RO 58

Given the following conditions:

- MP3 is holding at 75% power following a refueling outage.
- Rod control is in automatic.
- Rod height is 220 on Bank D.

Power range channel N-44 fails high.

WHICH ONE of the following describes the response on the rod control system?

A. Rods will drive in until a Tav-Tref error develops which will result in the rods driving out.

B. Rods will not move.

C. Rods will continuously drive in unless stopped by operator intervention.

D. Rods will drive in until the power mis-match circuit output decays away.

ANSWER:

D. Rods will drive in until the power mis-match circuit output decays away.

REFERENCES:

JUSTIFICATION: The over power rod stop will prevent outward motion (part a is incorrect).

Rod stops don't prevent inward auto motion (B is incorrect).

Temperature error and power mismatch circuit output decaying away will stop rod motion. (D is correct)

C is incorrect because the rods will eventually stop driving in without operator intervention.

OBJECTIVES: NIS07C (g)

K/A: 001 K1.05, Cause/effect between CRDS and NIS and RPS 4.5

Modified item 2207

RO 59

During a Reactor Startup, you must ensure the reactor goes critical above the rod insertion limit. The reason that this is safety significant at this time is to ensure:

A. Peak centerline fuel melt temperatures are not exceeded if you ejected a rod during startup.

B. Sufficient positive reactivity is available by the control rods to offset power defect on the power escalation.

C. No power tilt is introduced in the core because Bank C is partially withdrawn with Bank D still on the bottom.

D. The reactor will have adequate shutdown margin for a steam line break accident.

ANSWER:

D. The reactor will have adequate shutdown margin for a steam line break accident.

REFERENCE:

JUSTIFICATION: Tech Spec Bases. The hot zero power Rod Insertion Limit is to ensure reactor DNB limits are not exceeded on large Steam Line Break and the reactor doesn't return to critical on small steam line break. (D is correct)

A is incorrect because HCP rod ejection do not cause fuel melt concerns, rod ejection is limiting at HFP.

Power defect must be offset by rods and dilution on power escalations. Normal rod height for criticality is about Bank D at 150 steps. (B is incorrect).

QPTR is not a concern less than 50%. QPTR insures FQ (LOCA) calculations are within limits. This is not a concern at hot zero power. (C is incorrect).

OBJECTIVES: ROD03C (c); ROD08C (b)

K/A: 001 K5.08, RIL Setpoint

New Question

RO 60

PLANT CONDITIONS:

- A small break LOCA has occurred
- The crew is in ES-1.2, "Post-LOCA Cooldown and Depressurization"
- An RCS cooldown has been initiated by dumping steam to the atmosphere.

Which of the following statements describes the optimum reactor coolant pump configuration, and the basis for this configuration?

A. All RCPs should be stopped to minimize RCS inventory loss following break uncovery, and prevent steam voiding in the reactor vessel on subsequent RCS depressurization.

B. One RCP should be run to produce effective heat transfer and RCS pressure control, yet minimize RCS heat input.

C. One RCP should be run to produce effective heat transfer and RCS pressure control, yet minimize RCS inventory loss.

D. Two RCPs should be run to ensure symmetric heat transfer to the intact SGs, to enhance RCS pressure control, and to prevent steam voiding in the reactor vessel head on the subsequent RCS depressurization.

ANSWER:

B. One RCP should be run to produce effective heat transfer and RCS pressure control, yet minimize RCS heat input.

REFERENCE: ES 1.2 Background

JUSTIFICATION: Forced coolant flow is the preferred mode of operation to allow for normal RCS cooldown and provide PZR spray. All but one should be stopped to minimize heat input into the RCS. "A" is incorrect because if all RCPs are stopped voiding could occur in the vessel head during the depressurization. "C" is incorrect because the RCS inventory loss is based on the existing differential pressure and not on the forced flow through the RCS. While the reasons in distracter "D" are correct, the procedure does not address running two (2) RCPs.

OBJECTIVES: S1203C

K/A: WEST 03 EA1.3, Desired results 4.1

97 Exam 8

RO 61

Initial Plant Conditions:

- The plant is at 100% power at EOL
- Bank D rods are at 225 steps
- All controllers are in automatic

The following MB annunciators are received:

- TAVE/TREF DEVIATION
- ROD POSITION DEVIATION
- POWER RNG CHANNEL DEVIATION
- POWER RNG FLUX RATE HI

Based on the above indications, which of the following events has occurred?

A. The contolling first stage pressure transmitter has failed high.

B. An NIS power range upper detector has failed low.

C. A rod position indication channel has failed low.

D. A control rod has dropped into the core.

ANSWER:

D A control rod has dropped into the core.

REFERENCE: AOP 3552

JUSTIFICATION: The dropped control rod will generate the negative rate signal and the NIS power channel deviation because of the power tilt in the core. The dropped rod will cause the plant to cooldown causing the Tave/Tref deviation alarm. (D is correct)

OBJECTIVES: A5203C; ROD07C (c)

K/A: 003A2.03, Dropped rod using in-core/ex-core inst., loop temp.

New Question

RO 62

Which of the following represents the safeguards signal that actuates the listed CVCS system valves?

	RCP Seal Return	Charging Flow Control Isolation	Letdown Isolation
A.	CIB	CIA	SI
B.	CIA	SI	SI
C.	CIA	SI	CIA
D.	CIB	CIA	CIB

ANSWER:

C. CIA SI CIA

REFERENCES:

JUSTIFICATION: The seal return MOVs close on CIA and CVCS letdown and charging flow control isolation valves close on an SI.

OBJECTIVES: ECC03C (b.1)

K/A: 006 K4.09, Valve position on an SI

modified test item 2382

RO 63

Which of the following describes how the Emergency Diesel Generators are placed in the "Speed Droop" mode of operation?

A. Automatically selected when started Manually from Main Board 8 and no safeguards signals are present.

B. Taking the mode selector switch on Main Board 8 to "Unit".

C. Taking the mode selector switch on Main Board 8 to "Parallel".

D. Automatically selected for all Emergency Diesel Starts.

ANSWER:

C. Taking the mode selector switch on Main Board 8 to "Parallel".

REFERENCE: LSK 24-9.3C

JUSTIFICATION: Speed droop is selected when the mode selector switch is in the "Parallel" position, providing no safeguard signals are present. Speed Droop allows the diesel speed to change allowing load sharing. When the diesel is the sole source of power, (i.e. in Unit) no speed droop is allowed to ensure equipment is running at full speed.

OBJECTIVES: EDG02C 9v); EDG06C (f)

K/A: 064 A4.06, Manual start/stop of EDG

New question

RO 64

A large break LOCA occurred on unit 3. All systems responded as designed. When transitioning from E-0, the US directs you to perform an evaluation of the CSFs. You identify an orange path on integrity, a red path on containment and a yellow path on core cooling, a yellow path on heat sink, and a yellow path on inventory.

Based on this information, the operating crew should:

A. Transition to E-1.

B. Transition to FR-C.3.

C. Transition to FR-Z.1.

D. Transition to FR-P.1.

ANSWER:

C. Transition to FR-Z.1.

REFERENCES:

JUSTIFICATION: First transition to the highest priority critical safety function procedure which is FR-Z.1 - the only red path (correct), other answer are incorrect in accordance with EOP rules of usage.

OBJECTIVES: E0004C

K/A: 2.4.1, Ability to recognize entry conditions for EOPs 4.3

Modified Question 533

RO 65

The following plant conditions exist:

* Plant is in Mode 5 with all loops FULL
* Train "A" of RHR is in service providing shutdown cooling
* Train "B" electrical outage is in progress and is expected to last another 6 hrs.
* Due to a plane crash in the switchyard, all offsite power is lost
* The "A" EDG starts but fails to load onto Bus 34C
* The Unit Supervisor enters the appropriate procedure for loss of shutdown cooling

How should the crew re-establish shutdown cooling?

A. Transition to EOP 3501, Loss Of All AC Power (Mode 5, 6 and Zero), and restore power to be able to restart cooling.

B. Satrt and align the SBO Diesel onto Busses 34A/C and then perform Attachment B, Loss of Shutdown Coolong And/Or RCS Inventory Mode 5, of EOP 3505.

C. Initiate decay heat removal as the RCS heats up by dumping steam from at least one available SG.

D. Inject available SI Accumulators into the RCS.

ANSWER:

A. Transition to EOP 3501, Loss Of All AC Power (Mode 5, 6 and Zero), and restore power to be able to restart cooling.

REFERENCE: EOP 3505 Step 1

JUSTIFICATION: The appropriate procedure for loss of shutdown cooling in Mode 5 is EOP 3505

"A" is correct. Step 1 RNO will direct a transition to EOP 3501 if neither Bus 34C or 34D is energized.

"B" is incorrect because the SBO diesel should only be started using the guidance of EOP 3501. EOP 3501 directs the operators to attempt manual loading of the operating EDG before directing action to start/load the SBO diesel.

"C" and "D" are incorrect because they are strategies used in EOP 3501 after all attempts at energizing operable and degraded busses have been unsuccessful.

OBJECTIVES: E05805C (3)

K/A: 2.4.9, Low power operations in EOP

Exam Bank item 2935

RO 66

The plant is operating normally at 100% power.

Which of the following conditions require a reactor trip followed by a trip of ALL RCPs?

A. VCT temperature increases to 140°F.

B. Isolation of a single train of RPCCW to containment.

C. Seal injection flow to each pump decreases to 5.5 GPM.

D. The "A" RCP bearing oil temperature increases to 200°F.

ANSWER:

A. VCT temperature increases to 140°F.

REFERENCE: AOP 3561 Foldout Page

JUSTIFICATION: (A correct) If VCT temperature is greater than 135°F AND RCS temperature is above 400°F, then all RCPs must be stopped (RCPs are also stopped at any time if VCT is above 150°F)

(B incorrect)Both trains of RPCCW to containment must be lost to trip RCPs
(C incorrect) A reactor and pump trip is required if seal injection flow is less than 6 GPM AND thermal barrier cooling is also lost,
(D incorrect)High oil temperature only requires a trip of the affected pump

OBJECTIVES: A61761D (1)

K/A: 2.4.11, Knowledge of abnormal operating procedures

Question # 275

RO 67

The plant tripped from 100% power. A large break LOCA has occurred. The control room has carried out E-O and is now entered E-1. A computer failure has occurred and the US is performing a manual status tree check.

The following conditions exist:

SR energized with a negative SUR
Core Exit TC's 640 degrees
Subcooling 28 degrees
RVLMS Plenum 100%
S/G NR's 4%
Total AFW flow 500 GPM
Cold leg temp decrease in last hour 20 degrees
RCS temperature 480 degrees
Containment Pressure 60 psia
Pzr level is at 0%
RVLMS upper head is at 100%

What answer best describes the sequence for dealing with the Critical Safety Functions

A. Heat Sink
Containment

B. Containment
Heat Sink

C. Heat Sink
Core Cooling
Containment

D. Containment
Heat Sink
Core Cooling

ANSWER:

A. Heat Sink
Containment

REFERENCES: CSF status Trees. OP3272 EOP Users Guide Section 1.6

JUSTIFICATION:

SR energized with a negative SUR- **Green Path Subcriticality**
Core Exit TC's 640 degrees
Subcooling 28 degrees

RVLMS Plenum 100%- **Yellow Path Core Cooling**
S/G NR's 4%
Total AFW flow 500 GPM **Red Path Heat Sink**
Cold leg temp decrease last hour 20 degrees
RCS temperature 480 degrees **Green Path Integrity**
Containment Pressure 60 psia **Red Path Containment**
Pzr level is at 0%
RVLMS upper head is at 100% **Green Path Inventory**

(A correct) Red paths are first. Heat sink is before Containment
(B incorrect) Containment is after Heat sink
(C is incorrect) Core cooling is a yellow path and is before Containment which is a Red path
(D incorrect) Heat Sink and Containment are out of order of priority

OBJECTIVES: E0004C

K/A: 2.4.21, Knowledge of Status Trees/Logics for CSF

New question

RO 68

PLANT CONDITIONS:

Plant is in MODE 3
T_{ave} is 557 F
Main steam pressure is 1092 psig, contrc lled via turbine bypass valves
Atmospheric steam dump valves (ASD'/s), 3MSS*PV20A, B, C and D are in AUTO, set at 1150 psig

What is the proper method for changing ASDV set points?

A. Place turbine bypass valve controller (MSS-PK507) in MANUAL. Place all ASDV controllers in MANUAL. Adjust each ASDV set point to the desired value, then place the ASDV in AUTO. Return MSS-PK507 to AUTO.

B. Place one ASDV controller in MANUAL. Adjust the set point to the desired value, then place the controller in AUTO. Repeat this process for each remaining ASDV.

C. Slowly dial the ASDV thumbwheels to the desired value, ensuring the set point tracks properly and the valves do not open.

D. Place MSS-PK507 in MANUAL. Ensure all ASDVs are closed, then place each ASDV in MANUAL, one at a time, and adjust set points to the desired value.

ANSWER:

B. Place one ASDV controller in MANUAL. Adjust the set point to the desired value, then place the controller in AUTO. Repeat this process for each remaining ASDV.

REFERENCES: OP 3203

JUSTIFICATION: 4.26 Place the Atmospheric Steam Dump controllers in MANUAL one valve at a time prior to making any setpoint changes, then, Return the controller to AUTO. This prevents the Atmospheric Steam Dumps from opening rapidly causing steam pressure transients

OBJECTIVES: NAD106; NAD108

K/A: 2.2.2, Local/manual operations of controllers

Question: 2131

RO 69

While performing a new Surveillance on the Safety Injection pumps, the CO performs a step as written and notices that the Safety Injection pump does not have adequate recirc flow.

What is the first action(s) the CO should take?

A. Determine the cause of the problem.

B. Continue with the surveillance, but consult with the Unit Supervisor.

C. Stop the surveillance and place the Safety Injection pump in a stable or safe condition.

D. Initiate a procedure modification in accordance with DC-1, Administration of Millstone Procedures and Forms.

ANSWER

C. Stop the surveillance and place the Safety Injection pump in a stable or safe condition.

REFERENCE: DC-4, Procedural Compliance, 1.9

JUSTIFICATION: 1.9 Inadequate or Unexpected Results

1.9.1 IF procedure appears to be inadequate, OR yields unexpected results while executing work activity, PERFORM the following:

a. STOP work activity.

b. IF applicable, PLACE equipment or system in stable or safe condition. (C correct)

c. CONSULT First Line Supervisor for direction. (B incorrect)

d. DETERMINE cause of problem. (A incorrect)

e. IF necessary, Refer To DC 1, "Administration of Millstone Procedures and Forms" and INITIATE modification to procedure to rectify problem

(Stephen E. Scace memo "Procedure Compliance - Management Expectations" to Millstone Station Personnel, number MP-91-801, dated October 10, 1991, states: "If you think you can't follow the written procedure, consult First Line Supervision to determine what actions (i.e. procedure changes) are necessary before proceeding. If a procedure cannot be followed as written: a. Stop the task and place the equipment or system in a safe condition. b. Change the procedure using the procedure change process. c. Proceed with the task.")

OBJECTIVES: RAD544

K/A: 2.2.12, Knowledge of Surveillance Procedures

Question : 3260 modified

RO 70

When preparing a clearance, which of the following system/equipment conditions should be isolated from the work area by two closed valves in series?

A. A fluid system which operates at 170°F.

B. A gas system which operates at 50 psig.

C. Caustic or acid systems at any temperature or pressure.

D. Systems from confined work spaces.

ANSWER:

D. Systems from confined work spaces.

REFERENCE: WC-2

JUSTIFICATION: If practical, isolate fluid or gas systems that operate at greater than 200°F or 100 psig from the work area with two closed valves in series. Isolate systems from confined work space with two closed valves in series.

OBJECTIVES: NAD318

K/A: Generic 2.2.13 Tagging/Clearances

97 LOIT Remediation Exam

RO 71

Given the following conditions:

- MP3 was operating at 100% power when a spurious SI occurs.
- All systems respond as designed with the exception of the "B" reactor trip breaker, which did not open and remains closed.
- The crew responds IAW EOPs, eventually transitioning to ES-1.1, "SI TERMINATION".
- The crew takes both SI reset switches to RESET.

WHICH ONE of the following describes the status of SI?

A. Both trains of SI are reset and automatic initiation blocked.

B. Neither train of SI is reset, nor is automatic initiation blocked.

C. Both trains of SI are reset, only the "A" train automatic initiation is blocked.

D. Both trains of SI are reset, only the "B" train automatic initiation is blocked.

ANSWER:

C. Both trains of SI are reset, only the "A" train automatic initiation is blocked.

REFERENCES:

JUSTIFICATION: Both trains of SI can be reset but only Train "A" is blocked because P-4 is not enabled because B reactor trip breaker is closed. ("C" is correct.)

SI cannot be blocked in Train "B" because P-4 is not present. ("A" and "D" are incorrect.)

SI can be reset in both trains. ("B" is incorrect.)

OBJECTIVES: RPS012C

K/A: 013 K4.01, SIS Reset

New Question

RO 72

Following a Loss of Coolant Accident, adverse containment conditions exist. The following values have been recorded by the STA.

TIME	CONTAINMENT TEMP	CONTAINMENT RADIATION LEVELS
0800	185°F	5×10^4 R/HR
0815	190 °F	2×10^5 R/HR
0830	180 °F	2×10^5 R/HR
0845	175 °F	1×10^5 R/HR
0900	170 °F	9×10^4 R/HR

When, if ever, may the crew suspend the use of adverse containment values?

A. At 0845 because containment temperature has decreased below it's adverse setpoint.

B. At 0900 because both containment temperature and radiation levels are below their adverse values.

C. Adverse values can never be relaxed once they are entered if containment temperature limits were exceeded.

D. Adverse values can never be relaxed once they are entered if containment radiation limits were exceeded.

ANSWER:

D. Adverse values can never be relaxed once they are entered if containment radiation limits were exceeded.

REFERENCE: OP 3272

JUSTIFICATION: "A" is incorrect because containment radiation levels are still adverse and adverse values will always apply.

"B" is incorrect because 10^5 R/HR limits are/were exceeded.

"C" is incorrect because of containment radiation levels haven't exceeded 10^5 R/HR - containment temperature regarding adverse values can be relaxed when temperature drops < 180°F.

OBJECTIVES: E0003C

K/A: W 16 EK 1.3, Hi Rad Alarms and Actions

Modified Exam Item 3213

RO 73

FR C-1 is entered when core exit thermocouples are greater than 1200°F. Implementation of FR-C.1 is safety significant at this time because additional operator action is required to:

A. Prevent core uncovery.

B. Provide core cooling to stop the hydrogen generation due to zircaloy water reaction.

C. Limit containment pressure to less than design pressure.

D. Provide core cooling to prevent exceeding peak clad temperature limits.

ANSWER:

D. Provide core cooling to prevent exceeding peak clad temperature limits.

JUSTIFICATION: When FR-C.1 is entered the core is already uncovered. ("A" is incorrect.)

H_2 generation from zircaloy water reaction starts ~ 1800 - 2200F. ("B" is incorrect)

FR-C.1 is written for a small break LOCA with no high head injection. Car fans and spray will limit pressure. FR-C.1 established injection flow to cool core. ("C" is incorrect.)

OBJECTIVES: MC1004

K/A: 017 A2.02, Mitigating Core Damage

New Question

RO 74

Given the following conditions:

- The unit is at 45% power with all loops in operation.
- Control Rods M12 and D4 in Bank "D" , group 2, are stuck and misaligned and will not move.
- The affected rods are trippable.
- The Bank D stuck rods indicate 156 and 162 steps respectively on DRPI while the other Bank D rods and step counters indicate 180 steps.

Assuming the rods cannot be repaired within the next week, which one of the following correctly describes the actions required by Technical Specifications for the misaligned rods?

A. Verify shutdown margin requirements are met within 2 hour.

B. Verify QPTR within 1 hour and apply LCO 3.2.4.

C. Align the remaining Bank "D" rods to ± 12 steps of the inoperable rods.

D. Be in HOT STANDBY within the next 6 hours.

ANSWER:

D. Be in HOT STANDBY within the next 6 hours.

REFERENCE: Technical Specification 3.1.3.1, AOP 3552 "Malfunction of the Rod Drive System"

JUSTIFICATION: Technical specification 3.1.3.1 outlines the actions to be taken for a single stuck but trippable control rod.
A is incorrect because shutdown margin must be satisfied in one hour not 2.
B is incorrect because QPTR requirements are not applicable when less than 50% power.
C is incorrect because multiple rods in the same group are misaligned by greater than 12 steps. Alignment is not allowed and the unit must shutdown and be in hot standby because of the multiple misaligned rods in the same group (D is correct).

K/A: 005 K3.06, Actions in EOP

OBJECTIVE: ROD08C

Modified NRC Exam Item 95 LOIT Exam

RO 75

INITIAL CONDITIONS:

- MODE 5 with the RCS solid.
- Temperature being maintained 140°F to 150°F by the "A" RHR train.
- The plant is currently 4 days into a scheduled 8 day "B" train electrical outage, with 34B and 34D deenergized. "B" train load centers are NOT cross-tied to the "A" train.

Assuming NO operator action, which of the below statements describes the plant response to a loss of the 'A' Instrument Air Compressor?

A. RCP Thermal Barrier cooling flow from RPCCP will decrease.

B. Letdown flow will increase resulting in the RCS depressurizing.

C. RCS temperature will increase due to increased RPCCP flow through the RHR Heat Exchanger.

D. RCS temperature will decrease due to increased RHR flow through the RHR Heat Exchanger.

ANSWER:

D. RCS temperature will decrease due to increased RHR flow through the RHR Heat Exchanger.

REFERENCES: P&ID 104A, 121A, 112A, 121B

JUSTIFICATION: 'B' is incorrect because a loss of IAS will cause HCV128 to fail closed which will result in a loss of letdown and a resultant increase in RCS pressure.

'C' is incorrect because a loss of IAS will cause FV66A to fail AS IS resulting in NO change in RCS temperature from CCP flow. 'D' is correct because FCV 618 fail closed and HCV 606 fails open on a loss of IAS which will result in maximum flow through the RHR HX.

'A' is incorrect because the CCP return valves from the thermal barriers have a LOCK UP feature to prevent them from being affected by a loss if IAS. The CCP CTMT isolation valves are MOVs and therefore are not affected by a loss of IAS.

OBJECTIVES: PAS07C; RHR07C

K/A: 078 K3.02, Pneumatic control valves

96 AOP Exam

RO 76

Given the following:

- A loss of offsite power has occurred.
- Tave is 552°F
- "Turbine Bypass Tav Interlock Bypassed" is illuminated.
- Steam dumps are in steam pressure mode of control.
- Steam dumps demand is manually INCREASED to begin a cooldown.
- The steam dumps failed to open.

Which ONE (1) of the following explains why the steam dumps will NOT open?

A. P-12, LO-LO Tave, has disarmed the steam dumps.

B. P-4, Reactor trip, has locked out the steam header pressure signal.

C. C-9, Condenser available, interlock is not met.

D. The plant trip controller has not reset.

ANSWER:

C. C-9, Condenser available, interlock is not met.

JUSTIFICATION: "C" is correct because power is not available to circ pumps on loss of offsite power. Blocking signals override arming signals.

"A" is incorrect because P-12 has been bypassed.

"B" is incorrect because P-4 locks out the load rejection controller in the Tave mode of control.

"D" is incorrect because plant trip controller was reset when you shifted to pressure mode.

OBJECTIVES: SDS06C

K/A: 051 K3.01 Steam Dump Operation - Loss of Vacuum

Exam Item: 2422

RO 77

Which of the following conditions occurring concurrently with a large LOCA will required entry into ECA-1.1 - Loss of Emergency Coolant Recirculation?

A. Off-site power is lost and the "B" EDG did not start.

B. The power lockout relays fail to operate and all the white power lockout indicator lights are dim.

C. The A charging pump tripped on overcurrent and the B SI pump is tagged out for maintenance.

D. The "B" & "D" Recirc spray pumps are damaged and cannot be started.

ANSWER:

B. The power lockout relays fail to operate and all the white power lockout indicator lights are dim.

REFERENCE: ES-1.3 step 2, notes prior to step 2 & 4.

JUSTIFICATION: A is incorrect. The A EDG is still available to supply A train components for cold leg recirc.

C is incorrect - One charging pump and one SI pump are still available for cold leg recirculation.

D is incorrect because the A train of RSS is still operable.

B is correct because without the power lockout operating power not available to operate some of the recirculation valves, ES-1.3 directs the operator to ECA-1.1.

OBJECTIVES: A1101C

K/A: W/E 11 K2.1, Control, function, system

Modified 2853

RO 78

Which of the following plant conditions will cause the TD AFW pump to auto start?

A. 2/4 SG level detectors at low - low level in two SGs.

B. Safety Injection Signal.

C. Main Feedwater Isolation Signal.

D. Loss of Battery Bus 5

ANSWER:

A. 2/4 SG level detectors at low - low level in two SGs.

REFERENCE:

JUSTIFICATION: 2/4 low low signals in at least 2 SG will start TD AFW pump.

Only the motor driven AFW pumps start on SI (D is incorrect).

Main feedwater isolation will only isolate MFW but does not start any AFW pumps.

TD AFW pump will start on loss of batt Bus 1/2 not Bus 5 (D is incorrect).

OBJECTIVES: FWA04C

K/A: 059 A4.11, Auto start AFW 4.2

Modified 349

RO 79

Assume that prior to a startup, work on the Intermediate Range nuclear instrumentation resulted in BOTH channels being OVER COMPENSATED.

Which of the following describes the expected system response to this condition?

A. During startup the Intermediate Range indication will be less than actual, and during shutdown the Source Range may not be automatically reinstated.

B. During startup the Intermediate Range indication will be greater than actual, and during shutdown the Source Range may be reinstated prematurely causing an unwanted reactor trip.

C. During startup the Intermediate Range indication will be less than actual, and during shutdown the Source Range may be reinstated prematurely causing an unwanted reactor trip.

D. During startup the intermediate range indication will be greater than actual, and during shutdown the source range may not be automatically reinstated due to the P-10 permissive being active.

ANSWER:

C. During startup the Intermediate Range indication will be less than actual, and during shutdown the Source Range may be reinstated prematurely causing an unwanted reactor trip.

REFERENCE: Funct. Diag. Sht. 3 & 4

JUSTIFICATION: A is incorrect because on shutdown the Intermediate range detectors will read lower than actual and be automatically reinstated prematurely.

B & D are incorrect because readings on startup will be lower than actual not higher.

OBJECTIVES: NIS06C (a); NIS05C

K/A: 032 A2.04, SR/IR overlap

Bank Item 2217

RO 80

PLANT CONDITIONS:

- 100% power
- All systems in AUTOMATIC
- LT-459 selected for control of Pressurizer level control selected to LT-459
- PT-456 selected for control of Pressurizer Pressure
- Instrument for "A" and "C" steam generators selected to Channel I
- Instruments for "B" and "D" steam generators selected to Channel II

A loss of VIAC-2 occurs.

Which of the following lists controllers which should be taken to MANUAL as a result of the VIAC-2 failure?

A. Rod control
Pressurizer Pressure
Pressurizer Level

B. Rod Control
Pressurizer Pressure
Master main feed pump controller

C. Pressurizer Pressure
Pressurizer Level
Feed Regulating Valves for "A" & "C" SGs

D. Pressurizer Pressure
Master main feed pump controller
Feed Regulating Valves for "B" & "D" SGs

ANSWER:

D. Pressurizer Pressure
Master main feed pump controller
Feed Regulating Valves for "B" & "D" SGs

REFERENCE: AOP 3564, Process sheets 10 and 11, 25

JUSTIFICATION:

Pressurizer level controller will be affected because the backup channel will result in a letdown isolation
Pressurizer pressure is affected because its controlling channel is channel II
Main feed pump speed control is affected due to loss of two steam flow channels, low.
Rod Control is not affected
The Feed Regulating Valves on only the "B" and "D" SGs will be affected.

Only D correct.

OBJECTIVES: 12005C

K/A: ape 057 A1.06, Manual control of components

New question

RO 81

The plant is operating at 100% power. The steam dumps are in the Tave mode of control and rods are in AUTO.

3MSS*PT507, Main Steam Header Pressure, fails high. This will:

A. Open the steam dump cooldown valves, and the TDFWPs speed will decrease.

B. Block the load rejection controller from arming the steam dumps, and the TDFWPs speed will increase.

C. Arm the steam dumps but they won't open, and the TDFWPs speed will decrease.

D. Have no effect on steam dumps, and the TDFWPs speed will increase.

ANSWER:

D. Have no effect on steam dumps, and the TDFWPs speed will increase.

REFERENCES:

JUSTIFICATION: PT 506, not PT 507, feeds the load rejection controller and arms the steam dumps (B & C are incorrect).

The PT 507 effect on the cooldown valves function is only available in the pressure mode of control (A is incorrect).

PT 507 is only in effect when in the pressure mode of control (D is correct)

PT 507 feeds the TDFWP speed control circuitry. An increase in pressure will cause the pumps speed to increase.

OBJECTIVES: SDS07C

K/A: 039 A2.04, Manual control of components

Modified 2426

RO 82

INITIAL CONDITIONS:

- The Unit is operating at 48% power with the "NIS POWER RANGE P-9 PERMISSIVE" Blue Light NOT Lit.
- Due to an instrument failure, actual level in the "B" S/G level has increased to 80% resulting in an automatic FWI actuation.
- All equipment operated as designed.

Assuming no actions are taken in the instrument rack room, which of the following must occur to allow resetting the FWI signal from the main boards?

A. Clear P-14 and Reset P-4

B. Clear P-14 and P-9

C. Clear P-14 only

D. Reset P-4 only

ANSWER:

A. Clear P-14 and Reset P-4

REFERENCE: Functional Sheet 13

JUSTIFICATION: Permissive is lit below P-9, if level reaches the turbine trip setpoint, the reactor will trip, therefore to reset the FWI, both P-4 and P-14 will have to clear.

OBJECTIVES: NIS04C (b.4)

K/A: 059 A 4.11 Permissives

Question ID 1520
96 Quiz 6

RO 83

Initial Plant Conditions

- Plant is at 60% power
- Rod control is in manual
- Steam Dumps are in Tave mode of control

A turbine trip occurs. The turbine trip fails to cause a reactor trip and actuate the steam dumps on the turbine trip controller.

The next automatic reactor trip signal to be generated for this transient would be:

A. High Pressurizer Level Trip

B. OTΔT

C. High Pressurizer Pressure Trip

D. OPΔT

ANSWER:

C. High Pressurizer Pressure Trip

REFERENCES:

JUSTIFICATION: The power mismatch will cause a rapid increase in pressurizer pressure causing the reactor to trip (C is correct).

Pressurizer level will backup the high pressure trip (A is incorrect)

B is incorrect. The temperature increase will drive power down and pressure up. Both of these factors are benefits with regard to OTΔT.

D is incorrect. Power will decrease during the transient. The margin to the OPΔT trip will be increasing.

OBJECTIVES: A5002C; MC0302

K/A: 045 A1.05, RCS following turbine trip

New Question

RO 84

Plant History:

- A loss of off-site power occu...ed 10 minutes ago.
- The crew has stabilized the plant and just completed ES-0.1.
- RCS temperature has stabilized at 557°F
- Steam Generator levels are between 25 - 30% in the narrow range.
- Total AFV/ flow to the steam generators is 535 GPM.

The Operations Manager has directed the plant be maintained at the current plant conditions for RCS temperature and steam generator levels.

One day from now the total AFW flow should be approximately:?

A. The same.

B. 110 - 140 GPM.

C. 250 - 275 GPM.

D. 400 - 425 GPM.

ANSWER:

B. 100 - 140 GPM.

REFERENCE:

JUSTIFICATION: One minute after a reactor trip the decay heat level is approximately 3-4%. After one hour it is approximately 1.5-2% and after one day it is 0.7-1%. Consequently after one day the existing decay heat is approximately one quarter of its value following the trip. Therefore since the AFW system is maintaining SG level, the flow will reduce by one quarter to approximately 134 GPM.

OBJECTIVES: S0103C

K/A: W/E 9 EK2.2 Relationship between emergency feedwater flow to S/G and decay heat removal for facility heat removal following a trip.

New Question

RO 85

PLANT CONDITIONS:

- Reactor is operating at 100% rated thermal power
- Annunciator 5-3 on MB4C "PR UP DET HI FLUX DEV/AUTO DEFEAT" has alarmed
- All control rods are positioned within 12 steps of their group demand counters
- Maximum QPTR based on plant computer program 3R5 is 1.04

Assuming QPTR is not reduced, within two hours reactor power must be reduced to ________, and NIS overpower trips reduced to _______ within the next four hours.

A. 50%, 59%

B. 50%, 55%

C. 88%, 97%

D. 94%, 103%

ANSWER:

C. 88%, 97%

REFERENCE: Tech. Spec. 3.2.4 Action Statement c.2; OP 3273 Modified Bank.

JUSTIFICATION: If QPTR is greater than 1.02 but less than 1.09 then within 2 hrs reduce thermal power 3% of rated power for every 1% greater than 1.0 and similarly reduce the overpower trip setpoints within the next four hours.

OBJECTIVES: NIS08C (b)

K/A: 015 A1.04, NIS/QPTR

Modified 1058

RO 86

Given the following conditions:

- The unit is critical
- The crew is holding power at 1.0 x 10-8 amps.
- SG levels are being controlled on the bypasses, in automatic.

N-36 control power fuse blows.

WHICH ONE of the following describes the plant's response?

A. An IR high flux rod stop will be received and the reactor will remain critical.

B. The reactor will remain critical with no rod stops.

C. The reactor will trip on SR high flux when they automatically energize.

D. The reactor will trip on IR high flux.

ANSWER:

D. The reactor will trip on IR high flux.

REFERENCES:

JUSTIFICATION: Loss of control power fuses causes a trip signal to be sent to RPS through the Reactor Protection System and will also generate a rod stop and reactor trip on 1/2 coincidence (d is correct)

A is incorrect because the trip will cause the reactor to go subcritical. (This also makes b incorrect).

C is incorrect as source ranges will not automatically energize.

OBJECTIVES: NIS07 (c)

K/A: 015 K2.01, NIS channels, power supplies 3.3

Modified item 2269

RO 87

The following conditions exist:

- Crew is in EOP 3503, Shutdown Outside The Control Room
- Control room is filling with dense smoke
- Control room is ordered evacuated
- The reactor is tripped from 100% power
- The turbine is tripped

SI occurs after trip due to the steam dumps malfunctioning

Which of the following describes the procedural flow path under these conditions:

A. Complete EOP 3503 and then enter E-0.

B. Exit EOP 3503 and enter E-0.

C. Perform E-0 in parallel with EOP 3503.

D. Complete EOP 3503, then perform cooldown in accordance with EOP 3504.

ANSWER:

D. Complete EOP 3503, then perform cooldown in accordance with EOP 3504.

REFERENCES:

JUSTIFICATION: EOP rules of usage - if SI or Rx Trip occurs in EOP 3503, you should remain in EOP 3503.

OBJECTIVES: EOU (1733)

K/A: 067 K3.04, Actions in EOPs

New Question

RO 88

Given the following conditions:

- Unit is at 100% power
- Pressurizer level control is selected to 459/461
- VCT makeup control is in Automa'ic

A reference leg leak occurs in pressurizer level transmitter LT-459.

Assume no other operator action, which of the following occurs:

A. Letdown isolation occurs
3CH-FCV-121 ramps open
Auto makeup occurs
Unit will eventually trip on high pressurizer level.

B. 3CH-FCV-121 ramps closed
VCT diverts
Letdown isolation occurs
Pressurizer level will begin to increase
Unit will eventually trip on high pressurizer level.

C. Letdown isolation occurs
3CH-FCV-121 ramp open
Auto make-up occurs
VCT swaps over to RWST
Plant cools down
Unit trips on low pressurizer pressure because heaters are de-energized.

D. 3CH-FCV-121 ramps closed
VCT diverts
Letdown isolation occurs
Pressurizer level will continue to decrease due to seal leakoff.
Unit trips on low pressurizer pressure

ANSWER:

B. 3CH-FCV-121 ramps closed
VCT diverts
Letdown isolation occurs
Pressurizer level will begin to increase
Unit will eventually trip on high pressurizer level.

REFERENCES:

JUSTIFICATION: Reference leg failure will cause indicated level to fall high this causes FCV-121 to ramp close. Thus A & C are incorrect).

Act pressurizer level will decrease and the renaming channels will cause letdown isolation. Seal injection will fill pressurizer and cause eventually a high level trip (B is correct).

D is incorrect because seal injection still occurs even if the FCV-121 is closed and level will begin to increase.

OBJECTIVES: A5503C; PPL06C; PPL07C

K/A: 011 K3.01, Loss of PZR Level effect on CVCS

Modified 375

RO 89

Given the following conditions:

- The unit is at 8% power.
- Plant startup is in progress
- Pzr level instrument LT-459 has failed LOW.
- All actions of AOP 3571 "Instrument Failure" Attachment C are complete.

Which of the following describes the course of action the crew should take if a subsequent failure of Pzr level instrument LT-460 HIGH?

A. Verify reactor trip.

B. Stop the startup, and restore one of the failed channels of pressurizer level to OPERABLE status prior to increasing power above 10%.

C. Stop the startup, and restore both of the failed channels of pressurizer level to OPERABLE status prior to increasing power above 10%.

D. Within one hour initiate ACTION to be in at least HOT STANDBY within the next 6 hours.

ANSWER:

B. Stop the startup, and restore ONE of the failed channels of pressurizer level to OPERABLE status prior to increasing power above 10%.

REFERENCE: AOP 3571 "Instrument Failure" Attachment C, Pzr Level and Pressure Control Lesson Plan, Technical specification 3.3.1 and functional sheet 11

JUSTIFICATION: With all actions of the AOP complete, the bistable associated with the high Pzr level Rx. trip has been placed in a tripped condition

When the second channel fails high, the coincidence for a high pressurizer level reactor trip is met, however, the trip is blocked less than 10%. (A incorrect)

Technical specifications require 2 channels to be OPERABLE, however, this is required below P-7 (10%), and to increase above 10%, the bistables must be tripped within 6 hours. B correct, D incorrect.

It is not required to have both channels OPERABLE to increase above 10%, (C incorrect)

OBJECTIVES: PPL07C

K/A: 028 A1.01, Pressurizer level bistables

modified from 1995 MP3 NRC exam

RO 90

Given the following conditions:

- The reactor is tripped.
- A Loss of Offsite Power has occurred
- Safety Injection is actuated from a small LOCA.
- All ECCS equipment is operating as expected.
- Pressurizer level is 48% and increasing on all channels.
- RCS pressure is 1700 psia and decreasing slowly on all channels.

Which one of the following describes a leak location that is consistent with the indications given?

A. A leaking pressurizer safety valve.

B. The letdown line relief valve lifting

C. A reference leg break on pressurizer level instrumentation.

D. A failed open spray valve.

ANSWER:

A. A leaking pressurizer safety valve.

REFERENCES:

JUSTIFICATION: A PORV or safety valve failing open will cause pressurizer level to increase and pressure to decrease on all channels. ("A" is correct.)

"B" is incorrect because this break location will be isolated by the SI/CIA signal.

"C" is incorrect. On the effected reference leg the indicated level channel would increase and pressure would decrease. However, on the non-affected channel level will decrease as well as pressurizer pressure.

"D" is incorrect - a failed open spray valve will only cause pressurizer pressure to decrease.

OBJECTIVES: A5503C

K/A: 008 A1.01, Operation Monitoring Instrumentation from for PORV, sprays

New question

RO 91

During performance of the shift control room rounds, the Control Operator discovers that 3HVQ-RE49, ESF Building Normal Ventilation Monitor, indicates OFF-LINE for both Data-A and Data-B at the RMS Console.

Which of the below statements describes the operating status of the radiation monitor?

A. The radiation monitor may be considered operational once its data and operation is verified at the Local Indicating Control panel (LIC).

B. The radiation monitor must be considered inoperable.

C. The radiation monitor continues to indicate properly at the RMS Console but all radiation monitor control functions must be performed manually at the RMS Console.

D. The radiation monitor may still be considered operational because it will still notify control room staff of high radiation conditions by actuating the "RADIATION ALERT" and "RAD HI" annunciators on MB2.

ANSWER:

A. The radiation monitor may be considered operational once its data and operation is verified at the Local Indicating Control panel (LIC).

REFERENCES: P&ID 152A, RMS073T and RMS073C Handouts, PIR 391-043, MP3 Memo MP-3-0-385 dated 3/25/91, Kaman Instrumentation Operation - Maintenance Manual volumes 1 thru 3 and RMS Console Help display.

JUSTIFICATION: 'B' is incorrect because each local unit is completely self contained, requiring the computer room computer ONLY for transmitting data to the control room.

'C' is incorrect because if off-line from both data-A and data-B, all communications between control and the RMU is terminated.

'D' is incorrect because the alarms at MB2 are a function of the computer. If the RMU computer is not communicating with the computer room computer, it can not cause the MB2 alarms to actuate.

'A' is correct because although not communicating with the control room, each RMU is completely a stand alone unit and is designed to function without the control room computer. Once the data and operation has been verified correct for the RMU at its LIC, the unit may be considered operational per the SS (Memo MP-3-0-385 and PIR 391-043 and RMS073 handouts)

OBJECTIVES: RMS08C

K/A: 073 A4.02, RMS Control Panels/Indications

Exam Item 2407

RO 92

Given the following:

- A twenty five (25) year old Maintenance Contractor with complete exposure records has the following exposure record for the current calendar year:
  - Shallow Dose Equivalent - 2.55 REM
  - Committed Dose Equivalent - 0.75 REM
  - Deep Dose Equivalent - 2.13 REM
  - Lens Dose Equivalent - 3.08 REM
  - Committed Effective Dose Equivalent - 1.95 REM

WHICH ONE (1) of the following is this individuals Total Effective Dose Equivalent (TEDE) for the current calendar year?

A. 2.88 REM

B. 4.08 REM

C. 5.21 REM

D. 5.43 REM

ANSWER:

B. 4.08 REM

REFERENCE: RPM 1.3.1
Get RAD Worker Training

JUSTIFICATION: TEDE = CEDE + DDE = 1.95 REM + 2.13 REM = 4.08

OBJECTIVE: GET Radworker Training

K/A: 2.3.1 / 3.0 10CFR20 Radiation Limits

New Question

RO 93

The rad waste PEO is dispatched to change LWS-FLT3. This PEO has not performed this task before. The HP technician informs the PEO that the dose rate on the outside of the filter housing is 1 R/hr.

Which one of the following is not an example of ALARA techniques for reducing exposure for filter replacement.

A Long handled tools to remove the old filter

B Place the filter in a shielded drum upon removal to reduce exposure

C. Have the rad waste PEO be assisted by the Turbine Building PEO who has done the task several times before.

D. Have the new PEO perform the filter replacement on a mockup first.

ANSWER:

C. Have the rad waste PEO be assisted by the Turbine Building PEO who has done the task several times before.

REFERENCE: RPM 5.2.4 Section 1.2,1.3,1.4

JUSTIFICATION: RPM 5.4.2 list 3 main areas to reduce Radiation exposure. Time Distance and Shielding
A is Distance
B is Shielding
D is Reduced Time by practice on a mockup prior to the job.

RPM 5.2.3 Section 1.1 states that individual exposures within a work group are balanced consistent with experience.
C will not balance the exposure if the experienced person always does the job.

OBJECTIVES: NAD721; NAD722; NAD723

K/A: 2.3.10, ALARA procedures to reduce radiation exposure

New question

RO 94

Given the following conditions:

- MP3 is at 35% power.
- "RCP B STANDPIPE HI LEVEL" has lit.
- "B" seal injection flow is 8.2 GPM.
- "B" seal leak-off flow is 0.2 GPM.
- Seal return temperature is 150°F and rising steadily.
- Pump radial bearing - rising slowly @ 145°F

Based on the above indications, the operating crew should:

A. Trip the unit, secure "B" RCP and close its No. 1 seal leakoff valve within 2 minutes.

B. Trip the "B" RCP and close its No. 1 seal leakoff valve after the pump has been stopped for five minutes.

C. Close the "B" RCP's No. 1 seal leakoff valve within 5 minutes and shutdown the unit within the next 30 minutes then secure the "B" RCP.

D. Trip the "B" RCP and close its No 1 seal leakoff after the pump has been tripped for two minutes.

ANSWER:

D. Trip the "B" RCP and close its No 1 seal leakoff after the pump has been tripped for two minutes.

REFERENCE: OP 3554

JUSTIFICATION: Since power less than P-8, the RCP can be stopped without tripping the unit (a is incorrect)

0P 3554 requires tripping RCP and closing the seal leakoff valve within two minutes (D is correct) The RCP must be removed from service within 5 minutes of failure (not within 5 minutes of closing seal leakoff valve). (B and C are incorrect)

OBJECTIVES: A5403C

K/A: 015 A2.01, Cause of RCP failure

Modified question 1104

RO 95

WHICH ONE of the following interlocks must be satisfied to start an RCP?

A. RCP #1 seal ΔP must be greater than 200 psid.

B. The overcurrent trip selector switches must be in the cold position.

C. Cold leg and hot leg isolation valves must be open.

D. Cold leg isolation valve must be open and the loop bypass and hot leg isolation valve must be closed.

ANSWER:

C. Cold leg and hot leg isolation valves must be open.

REFERENCES:	RCP text; OP 3301B
JUSTIFICATION:	A and B are incorrect because they are procedure administrative requirements but are not part of the interlock circuitry C is correct. D is incorrect. The cold leg stop valve must be closed with the bypass fully open to satisfy the RCP interlock.
OBJECTIVES:	RCS04C
K/A:	003 K6.14, RCP starting requirements 2.9

Modified question 2159

RO 96

A small break Loss of Coolant Accident has occurred.

The current plant conditions exist at the completion of E-0 step 14:

- SI has occurred.
- All SI equipment started.
- Containment Temperature is 185°F.
- RCS pressure is 1800 psia and stable.
- CET's are 520°F.
- Pressurizer level is 50% and slowly increasing.

Assuming conditions do not significantly change, you would expect to stop one charging pump in:

A. E-0 Reactor Trip on Safety Injection

B. ES-1.1 - SI termination.

C. ES-1.2 - Post LOCA Cooldown and Depressurization

D. ES - 1.3 - Transfer to Cold Leg Recirculation

ANSWER:

C. ES-1.2 - Post LOCA Cooldown and Depressurization

REFERENCES:

JUSTIFICATION: A is incorrect because adequate subcooling doesn't exist and RCS pressure is less than 1950 psia (adverse containment) to stop a charging pump in E-0.

B is incorrect because adequate subcooling doesn't exist to make the transition to ES-1.1.

C is correct because a 100°F/hr cooldown will be started which will increase subcooling major to allow stopping a charging pump in ES-1.2.

D is incorrect because charging pump will be stopped in ES-1.2, and cooldown will place plant on RHR, ES-1.3 will not be entered for a small break LOCA.

OBJECTIVES: S1203C

K/A: E02 EK2.1, SI Termination

Modified Exam Item 1533

RO 97

Given the following conditions:

- The Unit was operating at 75% power.
- A small break LOCA occurred in coincidence with a loss of off-site power.
- RVLIS indicates that a void exists in the reactor vessel head.
- The cooldown was stopped and RCS pressure raised to regain subcooling margin. Increasing RCS pressure will ________ the size of the void and _____ the leakage from the RCS.

A. Increase; increase

B. Decrease; increase

C. Increase; decrease

D. Decrease; decrease

ANSWER:

B. Decrease; increase

REFERENCE:

JUSTIFICATION: Raising the pressure will decrease the size of the void but increase the leakage for the RCS. ("B" is correct)

OBJECTIVES: MC0703 (c)

K/A: 009 K3.06, Inventory Balance During Small Break Loss.

New Question

RO 98

PLANT CONDITIONS:

- Plant is in Mode 6
- No fuel movements are in progress
- "A" Train Electrical outage is in progress
- Computer is available
- "B" Spent Fuel Pool cooling pump caught fire and tripped 1 hour ago
- Spent Fuel temperature - 115°F and slowly increasing
- Spent Fuel Pool level - 37% and decreasing slowly
- Reactor Cavity seal - intact
- Fuel Building Monitoring Group Histogram - NORMAL
- RWST level - 1,000,000 gallons

For the existing plant conditions, which one of the following corrective actions should be taken?

A. Align RWST to gravity feed the spent fuel pool.

B. Establish emergency makeup using the fire water system.

C. Supply makeup to the Spent Fuel Pool from the Primary Grade Water System

D. Establish emergency makeup to the Spent Fuel Pool from Service Water.

ANSWER:

A. Align RWST to gravity feed the spent fuel pool.

REFERENCE: EOP 3505A, Att. A, Step 3

JUSTIFICATION: Gravity feed is the preferred method to the spent fuel pool. (A is correct)

Emergency makeup using the fire water system is only used if the additional attempt of emergency makeup from the RWST is attempted after the gravity feed method does not work (B incorrect).

C and D are least preferred and are only done if RWST is not available (C & D are incorrect).

OBJECTIVES: E0503C; E05A3C

K/A: 033 A2.02, Loss of Spent Fuel Cooling EOP actions

Exam Item 1295

RO 99

The following events have occurred:

- A SGTR has occurred subsequent to a steam break inside containment.
- The crew has transitioned to E-3, "SGTR", from E-2, "Faulted Steam Generator Isolation."
- They have identified and isolated the ruptured Steam Generator, which is not faulted, and are preparing to initiate RCS cooldown.

Current Plant Conditions:

- Containment temperature is 185°F
- RCS pressure is 1420 psia
- Ruptured Steam Generator pressure is 895 psig
- Intact Steam Generator pressures are 850 psig
- Faulted Steam Generator is 600 psig
- Core exit temperature is 485°F

Using the provided reference, determine the required core exit temperature to be achieved by the RCS cooldown, if necessary.

A. A cooldown is not necessary, core exit temperature is already less than the required temperature.

B. 413°F

C. 434°F

D. 476°F

ANSWER:

C. 434°F

<u>**PROVIDE ATTACHMENT TO STUDENTS**</u>

REFERENCE: E-3 Step 14a graph (Adverse CTMT parameters to be used)

JUSTIFICATION: CTMT 185°F Adverse parameters. Step states not to interpolate, therefore, 885 psig on graph to be used. A incorrect since RCS is above the required temperature.
476°F = non adverse number for 850 psig (D incorrect).

OBJECTIVES: E3003C

K/A: 038 A1.34, Cooldown to specific temperature

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RO 100

PLANT CONDITIONS

The unit is running down in power from 100% to take the unit off-line.

Loop 3 Tave fails unobserved to a constant output of 572°F.

Which one of the following describes where pressurizer level will stabilize under these plant conditions? Assume no operator action taken.

Pressurizer level will stabilize at:

A. 22%

B. 28%

C. 45%

D. 89%

ANSWER:

C. 45%

REFERENCES:

JUSTIFICATION: Pressurizer level is programmed with auctioneered high Tav. Pressurizer level will decrease, then control at program level for 572°F which is 45%. (C is correct).

B is incorrect because pressurizer level will not be decreased no load value.

A is incorrect because pressurizer level will not decrease to cause let down isolation.

D is incorrect as pressurizer level will not increase above program value.

OBJECTIVES: PPL07C

K/A: 004 A1.02, CVCS/Tav/Pressurizer level

New question

Reactor Examination Answer Key

1. b and a
2. a
3. d
4. b
5. d
6. a
7. c
8. d
9. c
10. c
11. d
12. c
13. b
14. b
15. a
16. c
17. d
18. d
19. b
20. b
21. a
22. d
23. b
24. c
25. d

51. c
52. a
53. d
54. b
55. b
56. b
57. d
58. d
59. d
60. b
61. d
62. c
63. c
64. c
65. a
66. a
67. a
68. b
69. c
70. d
71. c
72. d
73. d
74. d
75. d

Note: See justification attached to credit both A/B as correct answers for RO question #1.

26. a
27. b
28. c
29. c
30. c
31. b
32. b
33. c
34. b
35. c
36. b
37. d
38. d
39. a
40. b
41. b
42. b
43. b
44. c
45. b
46. d
47. b
48. d
49. c
50. c

76. c
77. b
78. a
79. c
80. d
81. d
82. a
83. c
84. b
85. c
86. d
87. d
88. b
89. b
90. a
91. a
92. b
93. c
94. d
95. c
96. c
97. b
98. a
99. c
100.c

Attachment 3

Millstone Unit 3 WRITTEN EXAM COMMENT AND NRC RESOLUTION

RO Question #1

Facility Comment: "RO Exam Test Item -- Question #1: The question asks for conditions that would require immediate boration by the Reactor Operator. The facility recommends acceptance of an additional answer (a) that an immediate boration is required in Mode 5 if shutdown margin requirements are not satisfied in accordance with the attached curves. From the curves, the required boron under the given plant conditions is 2125 ppm. Additionally note the entry conditions from the attached copy of AOP 3566 and OP 3209B."

NRC Resolution: Following review of the referenced procedures and curves the examiner agreed with the facility comment. There were two correct answers to RO question No. 1. In accordance with Interim Revision 8 of the Examiner Standard (ES) 403, Paragraph D.1.b, two correct answers were allowed for this question. The answer key was changed accordingly.

Attachment 4

SIMULATION FACILITY REPORT

Facility Licensee: Millstone Unit 3

Facility Docket Nos: 50-423

Operating Tests Administered from: July 7 through 10, 1997

This form is used only to report simulator observations. These observations do not constitute audit or inspection findings and are not, without further verification and review, indicative of noncompliance with 10 CFR 55.45(b). These observations do not affect NRC certification or approval of the simulation facility other than to provide information that may be used in future evaluations. No licensee action is required in response to these observations.

During the performance of simulator scenarios and JPMs there were several deficiencies that caused confusion or apprehension of the candidates. The following is a listing:

- During the performance of JPM 109, when charging flow was being re-established, the flow controller (3CHS*FCV121) went to maximum flow when its outlet isolation valve (3CHS*MV8106) was opened. The controller was previously set to minimum flow by the operator, per procedure. This problem was observed four times during the performance of this JPM.

- During one performance of JPM 109, AOV 197A/194A, the non-safety header isolation valves for the running reactor plant closed cooling water would not open although the safety injection (SI) signal had been reset (SI annunciator was not lit). The operator reset the SI a second time and the valves functioned properly.

- During performance of JPM 50, the spray valve controller was supposed to fail in the open position and remain there when shifted to manual to conduct an alternate path JPM. On two occasions when shifted to manual the valve responded normally and allowed the operator to control reactor pressure normally. The incorrect response of the simulator removed the dynamic actions required by this JPM. Proper operator knowledge on actions for the anticipated failure were verified by subsequent questions by the examiners. The chief examiner also questioned the simulator operator to verify that the simulator had been properly setup for this JPM and was satisfied with the simulator operator's answers.

- During several power maneuvers the main turbine was slow to settle at load set causing the operating crew to believe that there was a possible problem with the main turbine control system. This simulator response did not adversely impact the simulator scenarios other than extending the duration of the scenarios and causing operating crew concern.