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 March 17-20, 1997, AT Indian Point 2.
DOCKET #50-247 (Written Retake Exam)

On March 17-20, 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.

PRS
LCP

April 15, 1997

Mr. Stephen E. Quinn
Vice President - Nuclear Power
Consolidated Edison Company of
New York, Inc.
Indian Point 2 Station
Broadway and Bleakley Avenues
Buchanan, NY 10511

SUBJECT: INDIAN POINT TWO RETAKE EXAMINATION REPORT NO. 50-247/97-02 (OL)

Dear Mr. Quinn:

During the week of March 17, 1997, the NRC and your training department administered retake examinations to two employees of your company who had applied for licenses to operate the Indian Point Unit 2 station, but had failed portions of their first examination administered in October 1996. These retake examinations were developed by the Indian Point Unit 2 training department, utilizing the guidance of NUREG-1021, "Operator Licensing Examiner Standards," Rev. 7 and Generic Letter 95-06.

The results of the examinations were that both the Senior Reactor Operator (SRO) and the Reactor Operator (RO) candidate passed their exam and were issued licenses.

Should you have any questions concerning this examination, please contact me at 610-337-5211.

Sincerely,

Original Signed By:

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

Docket No. 50-247

Enclosure: Examination Report No. 50-247/97-01 (OL) and w/Attachments 1 and 2

IE42

9704180198 29

cc w/encl:
C. Jackson, Manager, Nuclear Safety and Licensing
B. Brandenburg, Assistant General Counsel
C. Faison, Director, Nuclear Licensing, NYPA
C. Donaldson, Esquire, Assistant Attorney General, New York Department of Law
Director, Electric Division, Department of Public Service, State of New York
W. Stein, Secretary - NFSC
F. William Valentino, President, New York State Energy Research and Development Authority
J. Spath, Program Director, New York State Energy Research and Development Authority

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

April 15, 1997

Mr. Stephen E. Quinn
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Division of Reactor Safety

Docket No. 50-247

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w/Attachments 1 and 2

9704180198 2pp.

IE42

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and Development Authority
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and Development Authority

U.S. NUCLEAR REGULATORY COMMISSION
REGION I

Docket No.	50-247
Report No:	97-02 (OL)
Licensee:	Consolidated Edison
Facility:	Indian Point Unit 2
Dates:	March 17 - 20, 1997
Examiner:	Joseph D'Antonio, Operations Engineer
Approved I	Glenn Meyer, Chief Operator Licensing and Human Performance Branch Division of Reactor Safety

9704180200 107pp.

EXECUTIVE SUMMARY

Examination Report 50-247/97-02(OL)

On March 18, 1997, one examiner administered an initial retake simulator examination to one Senior Reactor Operator (SRO) candidate. On March 20, 1997, the training department administered an initial retake written examination to one reactor operator candidate. These examinations had been developed by the facility training department in accordance with NRC guidance and approved, administered, and graded by the NRC.

Operations

Both candidates passed their examinations.

The facility did a good job in developing the examination.

In the simulator, the SRO candidate was generally soft spoken to the point where the other crew members would have to walk away from their boards to hear what she was saying. She did better once scenario events led to EOP entry and had no difficulty directing crew actions.

05 Operator Training and Qualification

05.1 Examination Development

a. Scope

The facility staff developed the written and operating examinations and submitted these proposed examinations for NRC review and approval. The NRC reviewed the proposed examinations, provided comments, and approved the examination for administration. The examinations, as administered, reflected incorporation of the NRC comments.

b. Observations and Findings

The NRC had minor comments on approximately twenty percent of the written exam. Half of the comments concerned the substantial number of negatively worded questions. This was an indication of a generally well prepared exam with some time pressure to meet schedule deadlines. This was understandable since the decision to perform this retake as a facility-generated exam and the scheduled date left the facility with approximately 1 month less than the normally allotted time for these exams. Nonetheless, the NRC comments were satisfactorily resolved for the administered exam.

The content of the simulator scenarios was good. The only necessary modification was to start one scenario at a different power level to provide an up-power maneuver as a normal event. However, the facility will need to considerably expand the level of detail provided in the "operator actions" writeup for the evaluator in future examinations.

c. Conclusions

The facility did a generally acceptable job of developing the examinations.

05.2 Exam Results

a. Scope

Initial retake licensing examinations were administered to one reactor operator and one senior reactor operator applicant.

b. Observations and Findings

Both candidates passed their examinations.

The items missed by the RO applicant on the written examination did not indicate any one particular area of weakness.

The SRO candidate in the simulator was generally soft spoken to the point where the other crew members would have to walk away from their boards to hear what she was saying. This trait renders crew briefs an absolute necessity to ensure all crew members are aware of all relevant information. The candidate did a good job both with crew briefs and in involving crew members in decision-making. Once scenario events resulted in EOP entry, she raised her volume somewhat and had no difficulty directing crew actions.

c. Conclusions

Summary Results

	RO Pass/Fail	SRO Pass/Fail
Written	1/0	waived
Simulator	waived	1/0
Walk-through	waived	waived
Overall	1/0	1/0

X1 Exit Meeting

The NRC expressed its appreciation for the facility examination development and validation efforts and facility accommodation of the needs of the examination process. Strengths and weaknesses observed in the operating examinations and exam development were discussed.

PARTIAL LIST OF PERSONS CONTACTED

J. Ferrick, Manager - Operations Training

Attachments:
1. Written Examination and Answer Key
2. Simulation Facility Report

ATTACHMENT 1

WRITTEN EXAMINATIONS AND ANSWER KEYS

REACTOR OPERATOR EXAMINATION

Consider the following initial conditions when selecting your answer:

- Reactor Power 100%
- RCS T_{ave} 559°F
- Control Rods Automatic/Bank D @ 215 Steps
- RCS Boron 1070 PPM

- Chemistry has advised the control room that #21 CVCS Mixed Bed Demineralizer resin is exhausted. The Reactor Operator is directed to coordinate with the Nuclear NPO to place #22 CVCS Mixed Bed Demineralizer in service.

- #22 CVCS Mixed Bed Demineralizer resin was replaced last week. The demineralizer has remained isolated since resin replacement. No operations have been performed on this demineralizer since that time.

Which of the following statements describes the effect, if any, that placing #22 CVCS Mixed Bed Demineralizer in service without saturating at existing RCS boron concentration will have on the following RCS parameters:

- RCS T_{ave}
- Control Rod Position
- RCS Boron Concentration

A. RCS T_{ave} DECREASE
Control Rod Position INCREASE
RCS Boron Concentration INCREASE

B. RCS T_{ave} NO CHANGE
Control Rod Position NO CHANGE
RCS Boron Concentration NO CHANGE

C. RCS T_{ave} INCREASE
Control Rod Position DECREASE
RCS Boron Concentration DECREASE

D. RCS T_{ave} DECREASE
Control Rod Position DECREASE
RCS Boron Concentration INCREASE

Which of the following statements correctly describes the status of the Main and Low Flow Feedwater Regulating Valves after a reactor trip from 100% power and subsequent five (5) minute cooldown to 530°F due to a stuck open atmospheric steam dump valve?

A. The Main Feed Regulating Valves are CLOSED, and the Low Flow Feed Regulating Valves are 75% OPEN (100% power position).

B. The Main Feed Regulating Valves AND the Low Flow Feed Regulating Valves are CLOSED.

C. The Main Feed Regulating Valves are full OPEN due to large level error signal caused by "shrink". The Low Flow Feed Regulating Valves are 75% OPEN (100% power position)

D. The Main Feed Regulating Valves are full OPEN due to large level error signal caused by "shrink". The Low Flow Feed Regulating Valves are CLOSED.

Consider the following initial conditions when selecting your answer:

- Reactor Power 100%
- RCS T_{ave} 559°F
- Control Rods Automatic/Bank D @ 215 Steps

The narrow range hot leg RTD for Loop 23 fails instantaneously high due to an open circuit. Which of the following statements correctly describes the effect of this failure on the Rod Control System?

A. Rods automatically insert to restore indicated T_{ave} to T_{ref}. Rod withdrawal is blocked in AUTOMATIC and MANUAL.

B. No rod motion occurs. Rod insertion and withdrawal are blocked in AUTOMATIC only. MANUAL rod insertion and withdrawal are available.

C. No rod motion occurs. AUTOMATIC rod insertion and withdrawal are blocked. MANUAL rod insertion is available. MANUAL rod withdrawal is blocked.

D. Rods automatically insert to restore T_{ave} to T_{ref}. AUTOMATIC and MANUAL rod insertion and withdrawal are blocked.

Consider the following event when selecting your answer:

- A small break LOCA has occurred
- The reactor was tripped from 100% power
- A Safety Injection was initiated due to low Pressurizer level
- Emergency Operating Procedure ES-1.2, Post LOCA Cooldown and Depressurization is being used to reduce safety injection flow.

Which of the following statements correctly describes the anticipated response of Pressurizer level and pressure immediately after the first Safety Injection Pump is stopped?

A. Pressurizer level AND pressure will INCREASE due to voiding in the reactor head.

B. Pressurizer level will DECREASE, Pressurizer pressure will INCREASE as water in the Pressurizer flashes to steam.

C. Pressurizer level AND pressure will DECREASE due to reduced injection flow.

D. Pressurizer pressure will DECREASE and Pressurizer level will INCREASE due to voiding in the reactor head.

If a Main Steam Isolation Valve (MSIV) began to drift closed, which of the following will occur?

A. Safety Injection due to High Steamline ΔP

B. Safety Injection due to High Steam flow from the unaffected Steam Generators

C. Turbine Trip from the MSIV 86 relay

D. Reactor Trip due to level shrink on the affected Steam Generator

Consider the following event when selecting your answer:

With the plant at 100% Reactor Power, control rod H-8 (center of core), drops approximately 100 steps (62 inches) from the bank position of 215 steps. Which of the following statements correctly describes the effect, if any, that this event will have on axial core power distribution (Δ-I)?

Assume that a turbine runback DOES NOT occur.

A. Δ I will become more negative due to reduced power generation in the top of the core and increased power generation in the bottom of the core.

B. Δ I will become less negative due to increased power generation at the top of the and reduced power generation at the bottom of the core.

C. Δ I will not change since rod H-8 is at the center of the core and will affect all quadrants equally.

D. Δ I will initially become more negative then return to its original value when positive reactivity from the power coefficient returns reactor power to its original value.

During operation at 100% reactor power, 6.9 KV Bus 1 normal supply breaker (UT-1) trips open due to a relay failure and a reactor trip occurs. From the choices below, select the protection signal which initiated the reactor trip?

A. Two loop loss of flow

B. 6.9 KV bus undervoltage

C. Reactor Coolant Pump under frequency.

D. Single loop loss of flow

Consider the following event when selecting your answer:

Following a reactor trip from 100% power due to a loss of both Main Boiler Feed Pumps, the Motor-Driven Auxiliary Feedwater Pumps are being used to control Steam Generator (SG) Levels. Shortly after the AFW system is placed in service the piping downstream of the Auxiliary Feed Regulating valve to #24 SG ruptures. Which of the following statements correctly describes the automatic response of the AFW system to this failure?

A. The #24 SG auxiliary feed regulating valve will automatically close. Feed flow to #21, #22 and #23 SG will not be affected.

B. The #22 SG and #24 SG auxiliary feed regulating valves will automatically modulate to prevent pump runout and maintain sufficient AFW discharge pressure to maintain auxiliary feed flow to #22 SG. Feed flow to #21 and #23 SG will not be affected.

C. The #23 SG and #24 SG auxiliary feed regulating valves will automatically modulate to prevent pump runout and maintain sufficient AFW discharge pressure to maintain auxiliary feed flow to #23 SG. Feed flow to #21 and #22 SG will not be affected.

D. Since #23 and #24 SGs are supplied by the same AFW pump, feed flow to both SGs will be completely lost. Feed flow to #21 and #22 SG will not be affected.

Due to failure of a pre-amplifier in the circuit for NIS Power Range Channel N42, the lower detector output has failed to zero. Total indicated power from NIS Power Range Channel N42 indicates 51% with the Reactor at 100% power. In order to repair the pre-amplifier the channel must be removed from service by removing the instrument power fuses.

Which of the following statements correctly states the effect that this operation will have on the Power Range Nuclear Instrumentation System and Reactor Protection System?

A. The high flux bistables for N42 will be inhibited from tripping. A minimum of two of the remaining three power range channels must sense a high flux condition to trip the reactor.

B. The high flux bistables associated with N42 will trip. A minimum of one of the three remaining power range channels must sense a high flux condition to trip the reactor.

C. The high flux bistables associated with N42 will trip. A minimum of two of the three remaining power range channels must sense a high flux condition to trip the reactor.

D. The high flux bistables for N42 will be inhibited from tripping. A minimum of one of the three remaining power range channels must sense a high flux condition to trip the reactor.

Which of the following correctly identifies the instruments that provide the pressure signal for the determination of the subcooling value displayed on Flight Panel (FD)?

A. Pressurizer Pressure (Channels I and II)

B. RCS OPS System Pressure Instruments (PT-413/433/443)

C. Pressurizer Pressure (Channels III and IV)

D. RCS Wide Range Pressure Instruments (PT-402/403)

Consider the following indications when selecting your answer:

• Reactor Power	100%
• T_{AVE}	559°F
• Control Bank D	215 steps/automatic
• Containment Area Radiation Monitor (R2)	Alarming
• Containment Atmosphere Radiation Monitors (R4, 42)	Normal
• Containment Area Radiation Monitor (R7)	Alarming
• Charging Pump Cell Area Radiation Monitor (R4)	Alarming
• Sample Cell Area Radiation Monitor (R6)	Alarming
• Plant Vent Radiation Monitors (R43/R44)	Normal

From the list below select the event which would explain the indications described above?

A. Reactor core fuel element failure

B. RCS leak in containment

C. RCS/CVCS leak in charging cell

D. RCS/CVCS leak in sample cell

While discharging #13 Waste Distillate Storage Tank (WDST) to the river using #14 Waste Distillate Transfer Pump (WDTP), an alarm occurs on Panel SAF-1, "R-54 LIQUID WASTE DISTILLATE HI RAD/TROUBLE" due to a high radiation signal.

Which of the following correctly identifies the anticipated response of the following Liquid Radwaste System components:

- #13 and 14 WDTPs
- #13 and 14 WDTP Discharge Valves (SOV CT-965-MCV and SOV CT-982-MCV)
- Common WDTP Discharge Valve (SOV CT-971-FCV)

A
#13 WDTP	RUNNING
#14 WDTP	STOPPED
#13 WDTP Discharge Valve (SOV CT-965-MCV)	CLOSED
#14 WDTP Discharge Valve (SOV CT-982-MCV)	CLOSED
Common WDTP Discharge Valve (SOV CT-971-FCV)	OPEN

B.
#13 WDTP	STOPPED
#14 WDTP	STOPPED
#13 WDTP Discharge Valve (SOV CT-965-MCV)	CLOSED
#14 WDTP Discharge Valve (SOV CT-982-MCV)	CLOSED
Common WDTP Discharge Valve (SOV CT-971-FCV)	OPEN

C.
#13 WDTP	RUNNING
#14 WDTP	STOPPED
#13 WDTP Discharge Valve (SOV CT-965-MCV)	OPEN
#14 WDTP Discharge Valve (SOV CT-982-MCV)	CLOSED
Common WDTP Discharge Valve (SOV CT-971-FCV)	CLOSED

D.
#13 WDTP	STOPPED
#14 WDTP	STOPPED
#13 WDTP Discharge Valve (SOV CT-965-MCV)	CLOSED
#14 WDTP Discharge Valve (SOV CT-982-MCV)	CLOSED
Common WDTP Discharge Valve (SOV CT-971-FCV)	CLOSED

REACTOR OPERATOR EXAMINATION

The Senior Reactor Operator has directed you to initiate a Containment Building Pressure Relief. Approximately 15 minutes after the release has been initiated the S-133B, Meteorological Data Display on the Accident Assessment Panel stops functioning. Which of the following actions should you take to compensate for this failure?

A. Immediately stop the release and have no further releases until the display is repaired.

B. Record the Plant Vent Radiation Monitor (R44) reading every hour until the release is terminated.

C. Verify that meteorological data is available and record meteorological data every hour throughout the remainder of the release.

D. Stop the release and prepare a new release permit using the most adverse meteorological conditions.

During power operation the CVCS Automatic VCT Makeup System initiates blended makeup due to the VCT level reaching the low level setpoint. An air leak on the supply line to Boric Acid Flow Control Valve, FCV-110A, causes air pressure to the valve diaphragm to decrease to 0 PSIG and prevents the valve from responding to the flight panel controller (FIC-110A). The air leak is small enough that it does not have any significant effect on instrument air header pressure.

Which of the following statements correctly describes the effect, if any, that this failure will have on the boric acid concentration of the Reactor Coolant System?

A. RCS Boric Acid Concentration will DECREASE because FCV-110A fails CLOSED on a loss of air pressure, causing blended makeup to have a lower than desired boric acid concentration.

B. RCS Boric Acid Concentration will INCREASE because FCV-110A fails OPEN on a loss of air pressure, causing blended makeup to have a higher than desired boric acid concentration.

C. RCS Boric Acid Concentration will NOT CHANGE because FCV-110A is only used when the Makeup Selector Switch is in the BORATE position.

D. RCS Boric Acid Concentration will NOT CHANGE because FCV-110A is only used when the Makeup Selector Switch is in the MANUAL position.

Consider the following conditions when selecting your answer:

- A small break LOCA has occurred
- RCS pressure is at 1900 PSIG
- The Safety Injection System has automatically actuated due to High Containment Pressure
- ALL ESF equipment has responded as designed
- RCS break flow is approximately 800 GPM
- No Charging Pumps are operating

Select the response which correctly describes the status of the following ESF equipment, and RCS parameters which will exist when stable conditions are reached in the Reactor Coolant System? Assume that a cooldown HAS NOT been initiated.

- SI Pump Status
- RHR Pump Status
- RCS Pressure
- RCS Break Flow

A. SI Pump Status — Running - Recirc to RWST
RHR Pump Status — Running - Flow to RCS
RCS Pressure — LESS THAN 1500 PSIG and GREATER THAN 150 PSIG
RCS Break Flow — LESS THAN 800 GPM

B. SI Pump Status — Running - Flow to RCS
RHR Pump Status — Running - On Recirculation
RCS Pressure — LESS THAN 1500 PSIG and GREATER THAN 150 PSIG
RCS Break Flow — LESS THAN 800 GPM

C. SI Pump Status — Running - Recirc to RWST
RHR Pump Status — Running - Flow to RCS
RCS Pressure — LESS THAN 150 PSIG
RCS Break Flow — GREATER THAN 800 GPM

D. SI Pump Status — Running - Flow to RCS
RHR Pump Status — Running - Flow to RCS
RCS Pressure — LESS THAN 150 PSIG
RCS Break Flow — LESS THAN 800 GPM

Consider the following initial conditions when selecting your answer:

- Reactor Power — 1×10^{-6} IR amps
- T_{ave} — 547°F
- Plant startup in progress

During a plant startup, after the Main Steam Isolation Valves have been opened and Main Condenser Vacuum has been established, a fault is detected on the Station Auxiliary Transformer causing a loss of 138 KV electrical power to the station. Assuming no operator action, which of the following describes the immediate impact that this failure will have on the operation of the following:

- Condensate Pumps
- Condenser vacuum
- Reactor

A. Condensate pumps — Stopped
Condenser vacuum — Stable
Reactor — Critical

B. Condensate pumps — Running
Condenser vacuum — Decreasing
Reactor — Tripped

C. Condensate pumps — Stopped
Condenser vacuum — Decreasing
Reactor — Tripped

D. Condensate pumps — Running
Condenser vacuum — Stable
Reactor — Critical

During power operation a catastrophic failure of the service water supply line to the #23 Containment Fan Cooler Unit results in flooding of the containment building including the filling of the reactor cavity sump to the 46 ft. elevation of the containment building. How and why could this failure impact the ability of the Emergency Safeguards System (ESF) to perform its design function if a design bases large break LOCA were to occur?

Assume all other ESF equipment operates as designed when the accident occurs.

A. Containment pressure may exceed design limits due to loss of cooling water to the Containment Fan Cooler Units which are required to meet the minimum safeguards equipment requirements for containment pressure control.

B. Accumulator isolation valves will be submerged and may fail to open when signaled resulting in failure of the accumulators to inject into the RCS.

C. Cold leg recirculation will be impossible since the recirculation pumps will be submerged preventing the establishment of any recirculation path.

D. Containment pressure and water level may exceed design limits due to reduction in containment free volume before the accident occurred.

Consider the following initial conditions when selecting your answer:

- Reactor Power 100%
- T_{ave} 559°F
- Control rods Bank D @ 215 Steps/Automatic

During power operation with the above conditions Power Range Channel N41 fails to maximum detector output (120%). Which of the following statements correctly describes the effect that this failure will have on the Rod Control System?

A. Control rods will automatically insert to reduce indicated reactor power to equal turbine power.

B. Control rods will automatically insert until the rate of change of indicated reactor power versus turbine power decays to zero.

C. Control rod motion will be inhibited due to the overpower rod stop.

D. Control rods will automatically insert due to a turbine runback initiated by the change in indicated reactor power exceeding 5% in 5 seconds.

While releasing #24 Large Gas Decay Tank (LGDT) the Nuclear NPO inadvertently opens RCV-014 (LGDT Release Hand Control Valve) to the full open position causing a the R-43/R-44 PLANT VENT HI RAD/TROUBLE alarm to actuate on panel SAF-1 in the control room, due to R-44, Plant Vent Gas Monitor, exceeding the High Radiation Gas alarm setpoint.

Which of the following statements describes the impact, if any, that this event will have on the following equipment:

- RCV-014
- PAB Supply Fan
- PAB Exhaust Fan

(A)	RCV-014	Closes
	PAB Supply Fan	No change
	PAB Exhaust Fan	No Change
B.	RCV-014	Closes
	PAB Supply Fan	Trips
	PAB Exhaust Fan	No Change
C.	RCV-014	No change
	PAB Supply Fan	Trips
	PAB Exhaust Fan	Trips
D.	RCV-014	Closes
	PAB Supply Fan	No Change
	PAB Exhaust Fan	Trips

During power operation the APPROACHING ROD INSERTION LIMIT 12.5" and the ROD INSERTION LIMIT 0" alarms are actuated on Panel SAF in the control room. Investigation reveals that all control and shutdown rods are fully withdrawn and that the rod insertion limit is not being violated, indicating that the alarms have been actuated due to a malfunction.

Which of the following components, if malfunctioning, could cause the alarms to actuate?

A. Individual Rod Position Indicator voltage decreases from 3.45 VDC to 0 VDC.

B. Rod Bottom Rod Stop Bistable

C. Pulse to Analog Converter (P/A Converter)

D. Rod Bottom Rod Stop Bypass Bistable

Consider the following initial conditions when selecting your answer:

- Reactor Power 60%
- T_{ave} 554°F
- Control rods Bank D @ 200 Steps/Automatic

During surveillance testing it is discovered that the #22 Reactor Coolant Pump is not performing at design conditions. The flow rate in loop #22 is 7% less than design due to degradation of the diffuser ring in the pump casing. Flow in the remaining loops are at or above design and total core flow is reduced by 5%. Which of the following statements describes how this condition will affect the following:

- Core Δ -T
- Departure From Nucleate Boiling Ratio (DNBR)

A. Core Δ - T — Lower than design
DNBR — Lower than design

B. Core Δ - T — Lower than design
DNBR — Higher than design

C. Core Δ - T — Higher than design
DNBR — Higher than design

D. Core Δ - T — Higher than design
DNBR — Lower than design

DNBR = Bnl / Area 2/1 = 2

2/2 = 1

Which of the following correctly identifies the signals that are used to provide input to the MBFP Speed Control System for determining the following parameters:

- Actual Feed Regulating Valve Differential Pressure (ΔP)
- Power
- % Startup Signal

A.	ΔP	Main Feed Header Pressure - Main Steam Header Pressure
	Power	Turbine First Stage Pressure (PT-412A)
	% Startup Signal	MBFP Governor Control Switch
B.	ΔP	Main Feed Header Pressure - Main Steam Header Pressure
	Power	Total Stm Flow (sum of all controlling steam flow channels)
	% Startup Signal	MBFP Governor Control Switch
C.	ΔP	Main Feed Header Pressure - Main Steam Header Pressure
	Power	Total Stm Flow (sum of all controlling steam flow channels)
	% Startup Signal	MBFP Foxboro Speed Controller Manual Setting
D.	ΔP	Main Feed Header Pressure - Steam Generator Pressure
	Power	Total Stm Flow (sum of all controlling steam flow channels)
	% Startup Signal	MBFP Governor Control Switch

While retrieving a dropped control in Control Bank D Group 1, with the unit at 80% power, a ROD CONTROL URGENT FAILURE alarm is received on panel SBF-1 in the control room. Which of the following statements correctly describes the cause of this alarm?

A. Remaining rods in Control Bank D Group 1 are not moving when demanded.

B. Alarm is caused by the affected rod position deviating by more than 12 steps from bank demand position.

C. Alarm is caused by control bank rods moving with no bank overlap.

D. All rods in Control Bank D Group 2 are not moving when demanded

Consider the following event when selecting your answer:

- Reactor Power 0%
- T_{ave} 530°F
- Total Steam Flow 0.75×10^6 lbm/hr
- Pressurizer Level 32%
- Pressurizer Pressure 2235 PSIG

The plant is at hot shutdown due to a tube leak in #21 Steam Generator(SG). The Main Steam Isolation Valve for #21 SG is closed and the Reactor Operator is performing a cooldown to Cold Shutdown using the condenser steam dumps.

While positioning the condenser steam dump valves, a controller failure causes ALL of the steam dumps to OPEN fully. A Safety Injection Signal is received. Directly afterwards the Reactor Operator notes the following parameters:

- Reactor Power 0%
- T_{ave} 523°F
- Total Steam Flow 0 lbm/hr
- Pressurizer Level 25%
- Pressurizer Pressure 2110 PSIG

Which of the following ESF actuation signals initiated the Safety Injection signal?

A. High Containment Pressure

B. High Steam Flow coincident with Low T_{ave}

C. Low Pressurizer Pressure

D. Main Steam Line ΔP (21 SG GREATER THAN 22,23,24 SG)

Select the answer which correctly identifies the automatic reactor trip signals that are blocked when the POWER BELOW P-7 permissive is enabled?

A. Pressurizer Low Pressure Reactor Trip
Pressurizer High Level Reactor Trip
Two Loop Loss of Flow
6.9 KV Bus Undervoltage

B. Pressurizer Low Pressure Reactor Trip
Pressurizer High Level Reactor Trip
Steam Generator Lo-Lo Level Trip
6.9 KV Bus Undervoltage

C. Pressurizer Low Pressure Reactor Trip
Pressurizer Low Level Reactor Trip
Two Loop Loss of Flow
6.9 KV Bus Undervoltage

D. Pressurizer High Pressure Reactor Trip
Pressurizer High Level Reactor Trip
Steam Generator Lo-Lo Level Trip
6.9 KV Bus Undervoltage

While performing a natural circulation cooldown using ES-0.2, Natural Circulation Cooldown, you are directed to depressurize the Reactor Coolant System to 1890 PSIG after verifying that the RCS Hot Leg temperatures are less than 550°F. Which of the following statements correctly describes the reason for the maximum limit on hot leg temperature before depressurization can commence?

A. Ensure that the ΔT limit between auxiliary spray fluid temperature and the RCS is not violated.

B. Ensure that wide range hot leg temperatures are approximately saturation temperature for SG pressure.

C. Ensure that RCS subcooling is above the RCP Termination Criteria for the E-0 series of procedures.

D. Ensure that adequate subcooling exists to prevent void formation in the reactor head when pressure is reduced to 1890 PSIG.

REACTOR OPERATOR EXAMINATION

During power operation an increase in RCS leakage is noted during a routine RCS leak rate surveillance test. The subsequent RCS leakage safety evaluation determines that Reactor Coolant Drain Tank in-leakage has increased by the same amount that RCS leakage has increased. Which of the following leakage sources is the most probable cause?

A. CVCS Letdown Line Relief valve leakage

B. Reactor Vessel Flange O-Ring

C. Pressurizer Power Operated Relief Valve (PORV) Leakage

D. Reactor Coolant Pump #3 Seal Leakage

REACTOR OPERATOR EXAMINATION

During power operation a Pressurizer insurge results from a step decrease in power from 100% to 90%. As Reactor Operator you observe both Pressurizer spray valves throttling open, and all Backup heaters energized simultaneously. As Reactor Operator you should:

A. Immediately TRIP the Backup heaters

B. Immediately CLOSE the Spray valves

C. Ensure Heaters and Spray continue to operate as designed

D. Take the appropriate action for the controlling Pressurizer Pressure channel failing

Which one of the following signals are used to automatically **ARM** the RCS Overpressurization System (OPS) during a plant cooldown to Cold Shutdown?

A. Two out of three (2/3) RCS Wide Range Cold Leg Temperature RTDs LESS THAN setpoint

B. Two out of three (2/3) RCS Wide Range Hot Leg Temperature RTDs LESS THAN setpoint

C. Two out of three (2/3) RCS Wide Cold Leg Pressures GREATER THAN setpoint

D. Two out of three (2/3) RCS Wide Cold Leg Pressures LESS THAN setpoint

While performing Emergency Operating Procedure FR H-1, Loss of Secondary Heat Sink, following a Loss of Coolant Accident (LOCA), you are directed to discontinue use of the procedure if RCS Pressure is LESS THAN non-faulted Steam Generator(SG) pressure(s). Which of the following statements correctly describes why it is not necessary to continue with heat sink restoration if RCS Pressure is LESS THAN non-faulted SG pressure(s)?

A. FR H-1, will initiate RCS Bleed and Feed. A bleed path is not necessary if RCS pressure is low.

B. Since RCS pressure is less than non-faulted SG pressure, RHR flow may be established to remove core decay heat.

C. Secondary heat sink is not required in this condition since core decay heat is removed by break flow.

D. Since RCS pressure is less than non-faulted SG pressure, an Inadequate Core Cooling condition probably exists, which is a higher priority procedure.

REACTOR OPERATOR EXAMINATION

During power operation a Main Steam Line Break occurs inside containment on the steam line from #24 Steam Generator(SG). A Safety Injection signal is received from the Main Steam Line ΔP circuit. During recovery operations the control room operators are unable to close the Main Steam Line Isolation Valve for #24 SG.

Which of the following statements is correct regarding the ability of the operators to control Reactor Coolant System (RCS) temperature during this event?

A. RCS temperature will continue to decrease until ALL SGs have dried out. Subsequent temperature control will be performed by limiting Auxiliary Feedwater flow.

B. RCS temperature will continue to decrease until 24 SG has dried out. Subsequent temperature control will be performed using the remaining SGs, and auxiliary feed water flow.

C. RCS temperature will continue to decrease until 24 SG has dried out. Subsequent temperature control will be performed using the Atmospheric Steam Dump Valve(s) on the intact SGs since an automatic Main Steam Line Isolation was actuated by the Main Steam Line ΔP Safety Injection signal.

D. RCS temperature will continue to decrease until ALL SGs have dried out. Subsequent temperature control will be performed by using RCS Bleed and Feed.

Which of the following statements correctly describes the purpose of the POWER ABOVE P-10 permissive?

A. Automatically blocks the Source Range High Flux and Intermediate Range High Flux reactor trips. Prevents re-instatement of the Source Range instruments.

B. Allow operator to manually block the Power Range High Flux Low Setpoint and Source Range High Flux reactor trips. Prevents re-instatement of the Source Range instruments.

C. Automatically blocks the Power Range High Flux Low Setpoint and Intermediate Range High Flux reactor trips. Prevents re-instatement of the Source Range instruments.

D. Allow operator to manually block the Power Range High Flux Low Setpoint and Intermediate Range High Flux reactor trips. Prevents re-instatement of the Source Range instruments.

REACTOR OPERATOR EXAMINATION

During power operation Channel I Pressurizer Pressure Instrument fails high to maximum output (100%). While performing the subsequent actions of the Abnormal Operating Instruction (AOI) the Reactor Operator is directed to trip the following bistables:

- Pressurizer High Pressure Reactor Trip EX
- Pressurizer Low Pressure Reactor Trip ILL
- Pressurizer Low Pressure Safety Injection ILL
- Pressurizer Low Pressure SI Unblock

Which of the following correctly states the expected status (illuminated/extinguished) of the associated bistable proving lamp as each bistable trip switch is placed in the trip position?

A. Pressurizer High Pressure Reactor Trip — Extinguished
Pressurizer Low Pressure Reactor Trip — Extinguished
Pressurizer Low Pressure Safety Injection — Extinguished
Pressurizer Low Pressure SI Unblock — Illuminated

B. Pressurizer High Pressure Reactor Trip — Extinguished
Pressurizer Low Pressure Reactor Trip — Illuminated
Pressurizer Low Pressure Safety Injection — Illuminated
Pressurizer Low Pressure SI Unblock — Extinguished

C. Pressurizer High Pressure Reactor Trip — Illuminated
Pressurizer Low Pressure Reactor Trip — Extinguished
Pressurizer Low Pressure Safety Injection — Extinguished
Pressurizer Low Pressure SI Unblock — Illuminated

D. Pressurizer High Pressure Reactor Trip — Extinguished
Pressurizer Low Pressure Reactor Trip — Illuminated
Pressurizer Low Pressure Safety Injection — Extinguished
Pressurizer Low Pressure SI Unblock — Extinguished

REACTOR OPERATOR EXAMINATION

Consider the following event when selecting your answer:

While the plant was at 100% Reactor Power a loss of the 138 KV offsite power source occurred, followed by a Turbine Trip on low vacuum and subsequent reactor trip. The Reactor Operator is performing Step 1 of E-0, Reactor Trip or Safety Injection.

Which of the following statements correctly lists the indications that the Reactor Operator will use to verify that the Reactor trip has occurred?

A. Rod Bottom Lights
Reactor Trip Breaker Position

B. Neutron Flux
Individual Rod Position Indication

C. Reactor Trip Breaker Position
Neutron Flux

D. Reactor Trip Breaker Position
Bank Step Counters

While performing actions directed by the Emergency Operating Procedures, the Reactor Operator resets the Containment Spray signal. After depressing the reset push-buttons, the Reactor notes that the white indicating lights above the buttons illuminate and remain illuminated after the buttons are released.

Which of the following statements correctly describes the reason the lights illuminated when the reset push-buttons were depressed and remained illuminated when the buttons were released?

A. The lights illuminated when the buttons were depressed to indicate that the spray signal was reset. The lights remained illuminated indicating that an automatic Containment Spray actuation signal was present.

B. The lights illuminated when the buttons were depressed to indicate that the spray signal was reset. The lights remained illuminated indicating that both containment spray pumps were running.

C. The lights illuminated when the buttons were depressed to indicate that the spray signal could NOT be reset. The lights remained illuminated indicating that an automatic Containment Spray actuation signal was present.

D. The lights illuminated when the buttons were depressed to indicate that the spray signal was reset. The lights remained illuminated indicating that both containment spray pumps were NOT running.

Which of the following correctly identifies the Process Radiation Monitors which are capable of AUTOMATICALLY terminating a Containment Purge if they sense a high radiation condition?

A. R-44, Plant Vent Gas Monitor
R-43, Plant Vent Particulate Monitor
R-42, Containment Gas Monitor

B. R-43, Plant Vent Particulate Monitor
R-41, Containment Particulate Monitor
R-42, Containment Gas Monitor

C. R-43, Plant Vent Particulate Monitor
R-44, Plant Vent Gas Monitor
R-41, Containment Particulate Monitor

(D) R-44, Plant Vent Gas Monitor
R-41, Containment Particulate Monitor
R-42, Containment Gas Monitor

While performing a plant cooldown using the Condenser Steam Dumps, a 6.9 KV MOTOR TRIP (COMMON) alarm is actuated. The Reactor Operator notes a step decrease in steam flow at the same time the alarm occurs. No other annunciators are actuated.

Which one of the following 6.9 KV motors is the most likely cause of the alarm?

A. Reactor Coolant Pump

B. Circulating Water Pump

C. Heater Drain Pump

D. Condensate Pump

Which of the following lists correctly identifies the plant equipment protected by concentrated foam fire suppression systems?

A. Hydrogen Seal Oil Unit
Main and Auxiliary Transformer Oils Systems
Clean and Dirty Oil Storage Tanks
Turbine Oil Reservoir

B. Hydrogen Seal Oil Unit
Boiler Feed Pump Console
Clean and Dirty Oil Storage Tanks
Turbine Oil Reservoir

C. Hydrogen Seal Oil Unit
Boiler Feed Pump Console
Support Facility Ignition Oil Tanks
Turbine Oil Reservoir

D. Main Generator
Boiler Feed Pump Console
Clean and Dirty Oil Storage Tanks
Turbine Oil Reservoir

Due a fire in the transformer yard, fire main pressure is decreasing because of heavy demand. Which of the following selections correctly identifies the sequence in which the standby fire pumps will automatically start to maintain fire header pressure?

A. Fire Main Booster Pumps
Fire Diesel Pump
Standby Pressure Maintenance Pump

B. Fire Diesel Pump
Standby Pressure Maintenance Pump
Fire Main Booster Pumps

C. Fire Main Booster Pumps
Standby Pressure Maintenance Pump
Fire Diesel Pump

(D) Standby Pressure Maintenance Pump
Fire Main Booster Pump
Fire Diesel Pump

REACTOR OPERATOR EXAMINATION

Following a unit trip from 50% power, a station blackout occurs due to a fault on the 138 KV offsite power supply. Which one of the following selections correctly identifies the 480 VAC equipment that will automatically start WHEN the 480 VAC safeguards busses breakers are re-energized?

NOTE:NO SI SIGNAL(s) EXIST

A. Two Non-Essential Service Water Pumps
Three Essential Service Water Pumps
21 and 23 Motor Driven Auxiliary Feed Water Pumps

B. Two Essential Service Water Pumps
21 AND 23 Auxiliary Feed Water Pumps
21, 22, AND 23 Component Cooling Water Pumps

C. Three Essential Service Water Pumps
21 AND 23 Auxiliary Feed Water Pumps
21, 22, AND 23 Component Cooling Water Pumps

D. Three Essential Service Water Pumps
21 AND 23 Auxiliary Feed Water Pumps
21 AND 23 Component Cooling Water Pumps

When loading electrical equipment on the 480 VAC busses while recovering from a unit trip and station blackout you are directed to limit the load on Transformers 5,2,3 and 6, to less than 200 Amps. Which of the following selections correctly identifies the indication (parameter) that will be used to verify compliance with this direction?

A. Station Service Transformer High Side (6.9 KV) Ammeter(s)

B. 480 VAC Bus Ammeters

C. 6.9 KV Station Auxiliary Ammeters

D. Sum of individual equipment ammeters

REACTOR OPERATOR EXAMINATION

Following a Safety Injection you receive indication that 125 VDC Control Power has been lost to all equipment powered from 125 VDC bus #21. Which of the following selections describe HOW and WHY this failure WILL or WILL NOT impact your ability to satisfy the following potential EOP requirements:

- RCP Trip Criteria
- SI Reduction
- Close Accumulator Isolation Valves

A. RCP Trip Criteria - No impact, RCPs can always be tripped from CCR
SI Reduction - SI pumps must be tripped locally at 480 VAC Switchgear
Accumulator Isolation Valves - No impact, control power supplied from individual breaker AC feed

B. RCP Trip Criteria - RCPs that have lost control power must be tripped locally
SI Reduction - No impact, control power automatically transferred to 23 DC Bus
Accumulator Isolation Valves - No impact, control power supplied from individual breaker AC feed

C. RCP Trip Criteria - RCPs that have lost control power must be tripped locally
SI Reduction - SI pumps must be tripped locally at 480 VAC Switchgear
Accumulator Isolation Valves - No impact, control power supplied from individual breaker AC feed

D. RCP Trip Criteria - RCPs that have lost control power must be tripped locally
SI Reduction - No impact, control power automatically transferred to 23 DC Bus
Accumulator Isolation Valves - Valves that receive control power from 21 DC bus cannot be closed

21

REACTOR OPERATOR EXAMINATION

With the unit at Hot Shutdown (HSD) a full load test is being performed on #21 Emergency Diesel Generator (EDG). #22 and #23 EDGs are operable and in AUTOMATIC. After the EDG is fully loaded, a fire in the Station Auxiliary Transformer causes a loss of 6.9 KV power. Which of the following statements describe the expected response of the #21 EDG and 480 VAC bus 5A?

A. #21 EDG trips and restarts when returned to AUTO
Bus 5A Normal Feeder Breaker opens
All loads strip
Blackout loads sequence start

B. #21 EDG trips and restarts when returned to AUTO
Bus 5A Normal Feeder Breaker opens
All loads except running blackout loads strip
Non-running blackout loads sequence start when bus is re-energized

C. #21 EDG continues to run
Bus 5A Normal AND Emergency Feeder Breakers open
Bus 5A Emergency Feed Breaker closes
Blackout loads sequence start

D. #21 EDG continues to run
Bus 5A Normal Feeder Breaker opens
All loads except running blackout loads strip
Non-running blackout loads sequence start

Consider the following initial conditions when selecting your answer:

- Reactor Power 100%
- #21 RHR Pump OOS for last six hours (24 hour LCO action time)

While operating with the above conditions the control room receives a report that a lubricating oil leak on the oil cooler for #21 EDG has been discovered making #21 EDG INOPERABLE (7 day LCO action time). It is estimated that it will take 72 hours to repair the leak. Which of the following statements correctly describes the actions that must be taken to ensure compliance with technical specifications?

A. Complete repairs on both #21 EDG and #21 RHR within 18 hours or place the plant in Hot Shutdown

B. Complete repairs on #21 EDG within 18 hours or place the plant in Hot Shutdown

C. Verify operability of remaining safeguards equipment and continue operation observing Technical Specification limits and action times for equipment out of service.

D. Complete repairs on #21 RHR Pump or be in Hot Shutdown within seven hours.

REACTOR OPERATOR EXAMINATION

You have been directed to conduct a plant cooldown from Hot Shutdown (HSD) to 350°F at 50°F/hr using the condenser steam dumps. Approximately one hour after positioning the steam dump valves in the MANUAL mode on the Steam Pressure controller to establish the desired cooldown rate you notice that the cooldown rate has decreased from 50°F/hr to 20°F/hr. Steam dump valve position has not been changed.

Which of the following statements correctly describes the reason that the cooldown rate has decreased and the actions necessary to maintain a constant cooldown rate?

A. Steam flow has decreased due to reduced steam pressure. Valves must be gradually opened as the cooldown progresses.

B. Steam dump pressure controller will not allow Main Steam Pressure to decrease below the dial setting in MANUAL or AUTO. Setpoint must be gradually reduced as the cooldown progresses.

C. ΔT between the steam temperature and the condenser cooling water (circulating water) has decreased. Cooldown rate cannot be increased unless circulating water flow is increased.

D. ΔT between RCS temperature and feedwater temperature has decreased. Feed flow must be increased to increase heat removal.

REACTOR OPERATOR EXAMINATION

Consider the following initial conditions when selecting your answer:

- Reactor Power 10%
- RCS T_{ave} 548°F
- Control Rods Bank D 125 steps/MANUAL
- Main Turbine Startup - approaching synchronous speed
- HP Steam Dumps AUTOMATIC/Steam Pressure Mode

While performing a plant startup, the turbine trips due to an overspeed condition. Which of the following selections is correct regarding the effect that this malfunction will have on the following RCS parameters:

- RCS T_{ave}
- Average Loop ΔT
- Pressurizer Pressure
- Pressurizer Level

A. RCS T_{ave} INCREASE
Average Loop ΔT INCREASE
RCS Pressure INCREASE
Pressurizer Level INCREASE

B. RCS T_{ave} DECREASE
Average Loop ΔT INCREASE
RCS Pressure DECREASE
Pressurizer Level DECREASE

C. RCS T_{ave} INCREASE
Average Loop ΔT INCREASE
RCS Pressure DECREASE
Pressurizer Level DECREASE

D. RCS T_{ave} INCREASE
Average Loop ΔT DECREASE
RCS Pressure INCREASE
Pressurizer Level INCREASE

In Accordance with System Operating Procedures, WHEN RCS temperature is GREATER THAN or EQUAL TO 350°F, the CCW Pump Auto Start key switch must be in the NORMAL position, and WHEN RCS temperature is LESS THAN 350°F, the CCW Pump Auto Start key switch is placed in the BYPASS Position.

Which ONE of the following statements correctly describes the reason for placing the CCW Pump Auto Start key switch in the BYPASS position WHEN RCS temperature is LESS THAN 350°F?

A. Block the CCW standby pump auto start feature to prevent water hammer when the RHR system is in service.

B. Allow operation of the CCW system with three pumps running to meet heavy demand imposed by RHR heat load.

C. Technical Specifications allow defeating the Auto start feature below 350°F if three CCW pumps are OPERABLE.

D. Permit operation of the CCW system with less than three pumps running when CCW is flowing through RHR Heat Exchangers.

During power operation, the control room coordinates with the NPOs to shift the Essential Service Water Header from the 1-2-3 Header to the 4-5-6 Header. When the necessary valving is completed the SWP Mode Control Switch on the SBF-1 panel is inadvertently left in the 1-2-3 position. The service water system is operating in the Three Header Configuration.

Which ONE of the following components will be supplied with service water if a Safety Injection signal is initiated?

A. Instrument Air Compressor Heat Exchangers

B. Emergency Diesel Generators

C. CCW Heat Exchangers

D. Containment Fan Cooler Units

Which ONE of the following selections identifies ALL of the conditions which will initiate a Containment Ventilation Isolation signal?

A. High Radiation (containment, R-41, R-42)
Containment Isolation Phase B
Containment Phase A Isolation Signal
Containment Hi-Hi Pressure Signal
Manual Containment Spray Signal

B. High Radiation (containment, R-41, R-42)
High radiation (plant vent, R-44)
Containment Phase A Isolation Signal
Containment Hi-Hi Pressure Signal
Manual Containment Spray Signal

C. High Radiation (containment, R-41, R-42)
High radiation (plant vent, R-44)
Containment Phase A Isolation Signal
Containment Hi-Hi Pressure Signal
Station Blackout

D. High Radiation (containment, R-41, R-42)
High radiation (plant vent, R-44)
Containment Phase A Isolation Signal
High Radiation (containment R-2, R-7)
Manual Containment Spray Signal

Which ONE of the following conditions would be considered a loss of containment integrity during normal operation?

A. An automatic containment isolation valve is found to be inoperable in the CLOSED position.

B. Weld channel zone pressure indicates 50 psig.

C. Personnel hatch inner door indicates OPEN, outer door indicates CLOSED.

D. Weld channel seal to the equipment hatch is lost.

In accordance with procedure Residual Heat Removal (RHR) System flow rate should be AT LEAST 2500 GPM per RHR Pump/RHR Heat Exchanger.

Which ONE of the following statements is correct regarding the reason for this limitation?

A. Minimize turbulence downstream of the associated RHR Heat Exchanger and HCV-638 and HCV-640.

B. Prevent the RHR pumps from reaching runout conditions.

C. Prevent vortexing at the RCS Hot Leg loop connection.

D. Ensure RHR flow is evenly distributed via the SI manifold to all RCS cold legs.

Which ONE of the following indications is used by the Emergency Operating Procedures to verify that Natural Circulation Cooling flow has been established?

A. SG Pressure - at saturation pressure for RCS Hot Leg temperature

B. RCS Cold Leg Temperatures - at saturation temperature for SG pressure

C. Core Exit Temperatures - at saturation temperature for current RCS pressure

D. RCS Pressure - stable or increasing

When using Emergency Operating Procedure(EOP) E-1, Loss of Reactor or Secondary Coolant, you are directed to operate Reactor Coolant Pumps (RCP) in accordance with EOP FR-C.2, Response to Degraded Core Cooling, rather than tripping the RCPs as directed by EOP E-1, if the RVLIS Dynamic Range indication indicates less than the value obtained from the following table:

44%	-	4 RCPs I/S
30%	-	3 RCPs I/S
20%	-	2 RCPs I/S
13%	-	1 RCP I/S

Which of the following statements is correct regarding the reason for the guidance described above?

A. RCPs operating under these conditions may seize when tripped preventing their restart in future recovery actions.

(B.) Tripping RCPs under these conditions could lead to core uncovering and an Inadequate Core Cooling Condition.

C. Tripping RCPs under these conditions would result in an increase of mass flow from the break due to phase separation of the fluid.

D. Tripping RCPs under these conditions will result in a loss of RCS pressure control due to the loss of Pressurizer spray capability.

REACTOR OPERATOR EXAMINATION

Which ONE of the following conditions is a condition which would required a control rod to be declared INOPERABLE?

A. Control Rod H-8 in Control Bank D is mis-aligned from the bank by -9 steps for 12 hours with the bank at 218 steps.

B. Control Rod H-14 in Shutdown Bank C is mechanically bound at 223 steps and will not trip.

C. Control Rod D-4 in Shutdown Bank A has a rod drop time of 1.6 seconds.

D. Control Rod L-3 in Control Bank A, Rod Bottom Bistable fails to initiate a turbine runback.

Consider the following indications when selecting your answer:

- Reactor Power 50%
- RCS T_{ave} 553°F
- #23 RCP #2 Seal Standpipe LOW Level Alarm actuated
- #23 RCP Seal injection flow 8.0 GPM
- #23 RCP #1 Seal Leakoff flow 3.0 GPM

While operating with the above conditions and indications the Reactor Operator reports that the RCS leakage calculation is normal. Which ONE of the following failures or malfunctions would support ALL of the above indications?

A. Failure of the #23 RCP #2 Seal

B. Failure of the #23 RCP #3 Seal

C. Failure of the #23 RCP #1 Seal

D. Failure of the #23 RCP Seal Package (#1,#2, and #3 Seal)

While the reactor is in the Refueling Condition the Reactor Operator identifies an unexplained increase in Source Range count rate and a steady positive 0.15 DPM Source Range Startup Rate indication. The Senior Reactor Operator directs the Reactor Operator to initiate boration of the RCS per A3.4, Uncontrolled Reactivity Addition.

Which one of the following boration flow paths and methods is the preferred method for completing this task in accordance with A3.4, Uncontrolled Reactivity Addition?

A. RWST via LCV 112B, Emergency RWST Makeup Stop

B. MOV-333, Emergency Boration Stop to charging pump suction

C. Normal boration flowpath at maximum rate to charging pump suction

D. Normal boration flowpath at maximum rate to Volume Control Tank

Which ONE of the following will NOT result in the automatic start of a Component Cooling Water Pump?

A. A loss of offsite electrical power is followed by Unit Trip AND Safety Injection.

B. CCW Header Pressure decreases to 60 psig with two CCW pumps running.

C. An inadvertent Safety Injection Signal is actuated due to an instrument failure. The 480 VAC busses are ALL energized from offsite electrical power.

D. A Station Blackout Signal is actuated due to a loss of offsite power following a Unit Trip. No Safety Injection signal is present.

Consider the following event when selecting your answer:

- The controlling Pressurizer pressure channel (CHANNEL I) has failed high with the unit at 100% power
- RCS pressure is stabilized at 2115 psig by manually closing the Pressurizer spray valves

After the plant is stabilized the Senior Reactor Operator reviews the following Technical Specification Limit:

Reactor Coolant System Pressure, Temperature and Flow Rate

The following DNB related parameters pertain to four loop steady state operation at power levels greater than 98% of full rated power:

- Reactor Coolant System T_{avg} ≤ 587.2°F
- Pressurizer Pressure ≥ 2190 psia
- Reactor Coolant System Total Flow Rate ≥ 331,840 GPM

Item (b), Pressurizer pressure, is not applicable during either a thermal power change in excess of 5% of rated thermal power per minute, or a thermal power step change of 10% of rated thermal power.

Under the applicable operating conditions, should reactor coolant T_{avg}, or Pressurizer pressure exceed the values given in items (a) and (b) the parameter shall be restored to its applicable range within 2 hours.

Which of the following statements correctly describes the appropriate response, if any, the Senior Reactor Operator should take to ensure compliance with this technical specification?

A. Specification for Pressurizer Pressure is not applicable because the transient was induced by an instrument failure.

B. Specification for Pressurizer Pressure is not applicable because there was no change in reactor power.

C. Specification is not applicable because it only applies when power is ≥ 98% of RATED FULL POWER which is equivalent to 98% of the maximum attainable power level of 108% (High Flux Trip Setpoint)

D. Restore RCS pressure to GREATER THAN 2190 psia within 2 hours or reduce power to ≤ 98%.

A main steam line rupture has occurred with the plant at hot shutdown resulting in a Main Steam Line ΔP Safety Injection Signal. After analyzing the following indications determine which ONE of the following selections correctly identifies the portion of the Main Steam System that has ruptured?

- 21 SG Pressure 780 psig (decreasing slowly)
- 22 SG Pressure 782 psig (decreasing slowly)
- 23 SG Pressure 340 psig (decreasing rapidly)
- 24 SG pressure 775 psig (decreasing slowly)
- ALL Main Steam Line Flow Indications 0 lbm/hr
- ALL MSIVs indicate OPEN

A. 23 SG Main Steam Line Upstream of MSIVs outside containment

B. 23 SG Main Steam Line between SG and Flow Element

C. 23 SG Main Steam Line Downstream of MSIVs

D. 23 SG Main Steam Line between Flow Element and Containment Penetration

REACTOR OPERATOR EXAMINATION

Consider the following event when selecting your answer:

- Reactor Power 15%
- T_{ave} 547°F
- Turbine Startup in progress Turbine at synchronous speed
- HP Steam Dumps Pressure Mode - AUTOMATIC

During plant startup with the above conditions, a failure of the exhaust boot (expansion joint) between #21 Low Pressure Turbine and the condenser results in a rapid loss of condenser vacuum to atmospheric pressure over a period of 5 minutes.

Assuming that all plant protection AND control systems function as designed, which ONE of the following statements correctly describes the response (INCREASE/DECREASE/NO CHANGE) of the following parameters as a result of this event.

- Reactor Power
- Turbine RPM
- Steam Flow
- T_{ave}

A. Reactor Power DECREASE
Turbine RPM DECREASE
Steam Flow NO CHANGE
T_{ave} NO CHANGE

B. Reactor Power DECREASE
Turbine RPM DECREASE
Steam Flow INCREASE
T_{ave} INCREASE

C. Reactor Power DECREASE
Turbine RPM DECREASE
Steam Flow DECREASE
T_{ave} INCREASE

D. Reactor Power NO CHANGE
Turbine RPM DECREASE
Steam Flow NO CHANGE
T_{ave} NO CHANGE

During power operation at 100% reactor power a loss of electrical power to #21 Instrument Bus occurs causing the loss of PT-412A. Which ONE of the following statements describes the response of the High Pressure Steam Dump System?

Assume that no equipment or instrumentation was out of service before the failure.

A. Loss of load interlock will trip arming the steam dumps.

B. T_{ref} will fail to 547°F causing the High Pressure Steam Dumps to OPEN.

C. Steam dump actuation will be inhibited due to loss of power to loss of load interlock.

D. T_{ref} will fail to 547°F. High Pressure Steam Dumps will OPEN IF loss of load interlock trips.

The Conventional NPO has reported a small electrical fire on the 5' elevation of the Turbine Building. The following fire fighting equipment is available to the Fire Brigade. Select the equipment which is most suitable for extinguishing a fire of this type?

A. Portable CO_2 fire extinguisher

B. High pressure water hose

C. Water stream portable fire extinguisher

D. Dry chemical fire extinguisher

REACTOR OPERATOR EXAMINATION

A fire in the control building requires that the Central Control Room be evacuated. You have been designated as the First RO, and the SRO has directed you to trip the Reactor locally. Which of the following selections list in ORDER the locations and equipment from which you would accomplish this task, assuming that you are unsuccessful after each attempt.

A. Cable Spreading Room - Reactor Trip Breakers
Cable Spreading Room - Rod Drive MG Set Breakers
480V Switchgear Room - Rod Drive MG Set Breakers
480V Switchgear Room - Bus 3A and 5A Supply Breakers
6.9 KV Switchgear - Station Service Transformer 3 and 5 Supply Breakers

B. 480V Switchgear Room - Rod Drive MG Set Breakers
480V Switchgear Room - Bus 2A and 6A Supply Breakers
Cable Spreading Room - Reactor Trip Breakers
Cable Spreading Room - Rod Drive MG Set Breakers
6.9 KV Switchgear - Station Service Transformer 2 and 6 Supply Breakers

C. Cable Spreading Room - Reactor Trip Breakers
Cable Spreading Room - Rod Drive MG Set Breakers
480V Switchgear Room - Rod Drive MG Set Breakers
480V Switchgear Room - Bus 2A and 6A Supply Breakers
6.9 KV Switchgear - Station Service Transformer 2 and 6 Supply Breakers

D. Cable Spreading Room - Reactor Trip Breakers
Cable Spreading Room - Rod Control System Power Cabinets
480V Switchgear Room - Rod Drive MG Set Breakers
480V Switchgear Room - Bus 2A and 6A Supply Breakers
6.9 KV Switchgear - Station Service Transformer 2 and 6 Supply Breakers

While verifying Containment Isolation valves are in the correct position following a Safety Injection actuation due to Hi-Hi Containment Pressure, you notice the following "Two is True" indication for MOV-222, RCP Seal Leakoff Containment Isolation Valve:

- Left side of light — Illuminated - AMBER light
- Right side of light — Extinguished

Which ONE of the following selections is correct regarding the expected position of MOV-222, AND the indicated position of MOV-222 with respect to the "Two is True" indicating lights?

A. Expected Position — OPEN
Indicated Position — OPEN

B. Expected Position — CLOSED
Indicated Position — CLOSED

C. Expected Position — OPEN
Indicated Position — CLOSED

D. Expected Position — CLOSED
Indicated Position — OPEN

REACTOR OPERATOR EXAMINATION

Following a reactor trip you are directed by ES-0.1, Reactor Trip Response, to verify that all control rods are fully inserted. Which ONE of the following rod position indications would meet the criteria for a control rod **NOT** being fully inserted?

A. Individual IRPI reading 5.5 inches

B. Group Step Counters not indicating zero (0)

C. Proteus Computer rod position 13 steps

D. 0.05 Volts on Digital Volt Meter (DVM)

While conducting a natural circulation cooldown using ES-0.2, Natural Circulation Cooldown, you are directed to control SG levels. Which ONE of the following actions if performed could temporarily impede or reduce natural circulation flow?

A. Steam generator #21 level is allowed to slowly increase to 55% as seen on the narrow range level indicator.

B. Auxiliary feed flow to #21 SG is rapidly increased from 50 GPM to 200 GPM

C. Steam generator #21 level is allowed to slowly decrease to 35% as seen on the narrow range level indicator

D. Auxiliary feed flow to #21 SG is rapidly decreased from 200 GPM to 50 GPM

While performing the immediate actions for a Reactor Trip using EOP E-0, Reactor Trip or Safety Injection, you are directed to de-energize 480 VAC busses 2A and 6A to trip the reactor since a reactor trip cannot be verified by available indication. After 480 VAC bus 2A and 6A are re-energized you are directed to depress the Blackout Relay Reset 480V push button on the SC panel IF the Main Generator Output Breakers are CLOSED.

Which ONE of the following statements is correct regarding the reason that the Blackout Relay must be reset at this time?

A. Resetting the Blackout Relay at this time removes the 480 VAC Bus 2A Undervoltage Signal from the blackout logic and IF the Main Generator Output Breakers are CLOSED (86P and 86BU relays reset) a station blackout signal will be avoided.

B. Resetting the Blackout Relay at this time removes the 480 VAC Bus 2A AND 480 VAC Bus 6A Undervoltage Signals from the blackout logic and IF the Main Generator Output Breakers are CLOSED (86P and 86BU relays reset) a station blackout signal will be avoided.

C. Resetting the Blackout Relay at this time removes the 480 VAC Bus 6A Undervoltage Signal from the blackout logic and IF the Main Generator Output Breakers are CLOSED (86P and 86BU relays reset) a station blackout signal will be avoided.

D. Main Generator Output Breakers should already be OPEN. Depressing the Blackout Relay Reset 480V push button will trip the 86P and 86BU relays causing the Main Generator Output Breakers to OPEN.

Consider the following event when selecting your answer:

- A small break LOCA has occurred and a Safety Injection (SI) Signal has been actuated.
- All SI equipment has operated as designed.
- EOP ES-1.2, Post LOCA Cooldown and Depressurization has been implemented.

While performing the actions required by EOP ES-1.2, you are directed to establish maximum charging flow to the RCS. Which ONE of the following statements is correct regarding the reason for this action when performing a Post LOCA Cooldown and Depressurization?

A. Maximum charging flow is established in order to provide maximum auxiliary spray flow capability in the event that Reactor Coolant Pumps are not running and normal spray is unavailable.

B. Maximum charging flow is established to ensure that maximum boration capability exists.

C. Maximum charging flow is established in an attempt to achieve SI Termination Criteria, thus avoiding the tedious task of SI Reduction.

D. Maximum charging flow is established in order to provide sufficient makeup so that SI pumps can be more readily reduced during the SI Reduction sequence.

Consider the following conditions when selecting your answer:

- Large Break LOCA has occurred
- NIS Power Range indication 0%
- Intermediate Range Startup Rate -1/3 DPM
- Core Exit Thermocouple Temperatures 1210°F
- Reactor Coolant Pumps TRIPPED
- RVLIS Natural Circulation Range 30%
- RCS Subcooling < 0°F
- Steam Generator Narrow Range Levels All < 10%
- Total Auxiliary feedwater flow 740 gpm
- Pressurizer Level 0%

The Watch Engineer reports that the above conditions require entry into the Functional Restoration Procedures.

Which ONE of the following Functional Restoration Procedures must be implemented?

A. FR-C.1, Response to Inadequate Core Cooling

B. FR-S.1, Response to Nuclear Power Generation / ATWS

C. FR-H.1, Response to Loss of Secondary Heat Sink

D. FR-I.3, Response to Voids in the Reactor Vessel

Following a Manual Reactor Trip you are unable to verify that the Reactor is tripped in accordance with E-0, Reactor Trip or Safety Injection, and FR-S.1, Response to Nuclear Power Generation/ATWS, is implemented.

Which ONE of the following describes the sequence of actions directed to be used in FR-S.1, Response to Nuclear Power Generation/ATWS, for tripping the Main Turbine if the turbine trip cannot be verified? (Assume each action is unsuccessful in tripping the turbine)

A. Manually trip the turbine from the CCR, CLOSE the MSIVs from the CCR and trip the turbine locally at the "Front Standard."

B. Manually trip the turbine from the CCR, manually runback the turbine in the CCR and trip the turbine locally at the "Front Standard."

C. Manually trip the turbine from the CCR, CLOSE the MSIVs from the CCR, manually runback the turbine in the CCR and locally CLOSE the MSIVs

D. Manually trip the turbine from the CCR, trip the turbine locally from the "Front Standard," manually runback the turbine in the CCR and locally CLOSE the MSIVs

Consider the following events when selecting your answer:

- A tube rupture has occurred in #22 Steam Generator (SG)
- A Safety Injection (SI) signal has been actuated
- EOP E-3, Steam Generator Tube Rupture has been implemented
- #22 SG has been isolated
- The SI signal has been reset

The SRO has directed you to dump steam from the intact SGs at the maximum rate to establish a Core Exit Temperature of 488°F AND then stop the cooldown.

Which ONE of the following statements correctly describes the reason for reducing RCS temperature to this value?

A. Reduce RCS pressure by causing an outsurge from the Pressurizer to minimize leakage into the #22 SG.

B. Establish sufficient subcooling in the RCS so that the RCS will remain subcooled after pressure is decreased to #22 SG pressure.

C. Establish sufficient subcooling in the RCS so that the Reactor Coolant Pumps will not have to be tripped when the RCS pressure is decreased to #22 SG pressure.

D. Reduce temperature of RCS fluid leaking into #22 SG to reduce #22 SG pressure to minimize potential of radioactive release through the atmospheric steam dump valve.

Consider the following event when selecting your answe

- Reactor Power 80%
- RCS T_{ave} 557°F
- Control Rods Control Bank D - 195 steps/Manual
- RCS Boron Concentration 980 ppm Beginning Of Life (BOL)

After withdrawing control rods to adjust T_{ave} you note that when you release the In-Hold-Out switch Control Bank D rods continue to withdraw for an additional 20 steps. As a result T_{ave} increases to 5°F above program.

Which ONE of the following statements is TRUE regarding this event?

A. IF the same event occurred at EOL the INCREASE in RCS T_{ave} would have been GREATER.

B. IF the same event occurred at EOL the INCREASE in RCS T_{ave} would have been the SAME.

C. IF the same event occurred at EOL there would have been NO INCREASE in RCS T_{ave}.

D. IF the same event occurred at EOL the INCREASE in RCS T_{ave} would have been LOWER.

REACTOR OPERATOR EXAMINATION

Which ONE of the following AUTOMATIC actions will NOT occur if a single control rod drops from the fully withdrawn position to the fully inserted position while the reactor is at 100% power?

A. NIS Dropped Rod Stop

B. APPROACHING ROD INSERTION LIMIT 12.5" and the ROD INSERTION LIMIT 0" alarms actuate on Panel SA.

C. Turbine Runback

D. Rod Bottom Rod Stop

Consider the following event when selecting your answer:

- A Pressurizer Power Operated Relief Valve (PORV) has OPENED and failed to close.
- A Safety Injection Signal has been actuated.
- RCS Pressure stable 1480 psig 1495
- Pressurizer Relief Tank pressure 30 psig 45
- Pressurizer relief tank temperature 140°F
- Pressurizer Level 90%
- Pressurizer Relief Tank Level 82%

Using the indications and conditions provided determine which ONE of the following temperatures would be indicated on the PORV Downstream Temperature Indicator?

A. 593°F

(B) 275°F

C. 250°F

D. 140°F

Abnormal Operating Instruction (AOI) 3.1, Chemical Volume Control System (CVCS) Malfunctions, states that when preparing to start a Charging Pump, the Charging Pump controller must be placed in manual AND set for approximately 20% before the pump is started?

Which ONE of statements is correct regarding the reason for setting the controller to 20% before starting the Charging Pump?

A. Ensure that a low bearing oil pressure trip does not occur during the charging pump start.

B. Minimize starting current on the charging pump motor.

C. Balance MANUAL signal with AUTO signal before the pump is started.

D. Provided a minimum of 8 GPM seal injection flow to each RCP as soon as the pump is started.

When operating the Residual Heat Removal System (RHR) with the Reactor Coolant System (RCS) at reduced inventory care must be taken to control RCS water level such that the RHR pumps do not cavitate or become airbound. SOP 1.2, Draining Reactor Coolant System, and AOI 4.2.1, Loss of Residual Heat Removal System, impose restrictions on minimum RCS water level based on certain conditions/parameters.

Which ONE of the following parameters/conditions is a factor in determining the minimum allowable RCS water level?

A. Reactor Coolant System Temperature

B. RHR System Flow Rate

C. Which RHR Heat Exchanger is in service

D. Which RHR Pump is Running

Consider the following indications when selecting your answer:

- Reactor Startup in progress
- Source Range N31 Count Rate $2X10^3$ cps
- Source Range N32 Count Rate $4X10^5$ cps
- Intermediate Range N35 Current $<1X10^{-11}$ amps
- Intermediate Range N36 Current $<1X10^{-11}$ amps
- Source Range N31 Startup Rate 0.5 dpm
- Source Range N32 Startup Rate 0.1 dpm
- Control Rods Control Bank D - 100 steps/MANUAL

Using ONLY the information provided determine which ONE of the following actions and associated reason is appropriate regarding the continuation of the reactor startup?

A. Source Range N32 is reading high due to a failure in the Pulse Height Discrimination circuitry and should be considered inoperable. The startup may continue without further action.

B. Source Range N31 is reading low due to a failure in the Pulse Height Discrimination circuitry and should be considered inoperable. The startup may continue without further action.

C. Intermediate Range N35 AND N36 are not responding. Startup may continue as long as neutron flux remains in the source range.

D. Nuclear instrumentation is NOT indicating as anticipated. The approach to criticality SHALL be stopped AND no actions SHALL be taken which could add positive reactivity until the discrepancy is resolved.

Technical Specifications requires that RCS activity (for nuclides other than tritium with half lives of more than 30 minutes) be LESS THAN 60/E-bar µCi/cc when RCS temperature is GREATER THAN 500°F. Which of the following correctly describes the reason for the 500°F limit?

A. The probability for a Large Break LOCA has increased

(B) The saturation pressure for 500°F is less than the S/G Safety valves' lift setpoint for Steam Generator Tube Rupture release concerns.

C. The volatility of the radioactivity increases above 500°F.

D. Nuclear Instrumentation may be indicating incorrectly due to the increased radiation levels resulting from the high activity.

REACTOR OPERATOR EXAMINATION

A reactor trip has occurred. Approximately 30 minutes after the reactor has tripped the Reactor Operator is performing actions directed by ES-0.1, Reactor Trip Response, when he notes the following indications:

- Intermedjate Range N35 $1X10^{-8}$ amps (stable)
- Intermediate Range N36 $1X10^{-11}$ amps (stable)
- Source Range N31 0 cps (stable)
- Source Range N32 0 cps (stable)
- SOURCE RANGE LOSS OF DETECTOR VOLTAGE annunciator illuminated

Which ONE of the following statements correctly describes the response, if any, that the Reactor Operator should take regarding these indications?

A. Manually re-energize the Source Range NIS by depressing the Train A and Train B Intermediate Range Permissive Override push buttons.

B. Manually re-energize the Source Range NIS by depressing the Train A and Train B Power Range Permissive Override push buttons.

C. Initiate rapid boric acid injection in accordance with, A 3.4, Uncontrolled Reactivity Addition.

D. Monitor Intermediate Range N35, and verify reinstatement of the Source Range NIS when both Intermediate Range instruments are less than $1X10^{-10}$ amps.

Consider the following indications when selecting your answer:

- Reactor Power 100%
- T_{ave} 559°F
- Total RCS Leakage 0.8 gpm (includes SG tube leakage)
- SG Tube Leakage 0.25 gpm

The Senior Watch Supervisor has directed the control room operators to perform a reactor shutdown due to increasing secondary side activity caused by SG tube leakage. Which ONE of the following statements is correct regarding the anticipated change, if any, in total RCS leakage as a result of the plant shutdown?

A. Total RCS leakage will increase, due to the increase in SG tube leakage.

B. Total RCS leakage will remain the same, increase in SG leakage will be offset by decrease in other RCS leakage.

C. Total RCS leakage will decrease, due to the decrease in SG tube leakage.

D. Total RCS leakage will remain the same, decrease in SG leakage will be offset by increase in other RCS leakage.

REACTOR OPERATOR EXAMINATION

While recovering from a Reactor Trip due to a loss of both main feedwater pumps, EOP FR-H.5, Response to Steam Generator Low Level is entered due to indication that water level in #23 SG has decreased to 0% WIDE RANGE level. In accordance with FR-H.5, the SRO directs you to feed #23 SG at LESS THAN 100 GPM UNTIL water level is GREATER THAN 10% as indicated on #23 SG WIDE RANGE level indication.

Which ONE of the following statements is TRUE regarding the reason for limiting feed water flow to #23 SG to LESS THAN 100 GPM until level is GREATER THAN 10%?

A. Feed flow is limited to prevent a rapid RCS cooldown which could result in challenge to the INTEGRITY critical safety function.

B. Feed flow is limited to prevent unnecessary thermal shock to a "Hot Dry SG" which could result in SG tube failure.

C. Feed flow is limited to prevent flashing in the SG which could result in lifting the SG Safety Valves.

D. Feed flow is limited to prevent runout conditions on the #23 Auxiliary Feed Water Pump.

Which ONE of the following indications is used by the Emergency Operating Procedure E-3, Steam Generator Tube Rupture, to IDENTIFY the RUPTURED Steam Generator?

A. Safety Injection flow deviation for the affected loop.

B. Affected Steam Generator's pressure increase.

C. High Radiation from the affected SG blowdown line.

D. High Radiation from the Steam Jet Air Ejector Vent Radiation Monitor R45.

Following a manual Reactor Trip from 100% power you are performing Step 3 of E-0, Reactor Trip and Safety Injection, "Check If SI Actuated", and note the following indications:

• SI Annunciator	Not Lit
• SI Pumps	None Running
• Pressurizer Pressure	2090 psig (stable)
• Steamline ΔP	All < 50 psid (stable)
• Steam Line flow	All < 100,000 lbm/hr (stable)
• Containment Pressure	0.75 psig (stable)
• Pressurizer Level	5% (stable)
• RCS Subcooling Margin	87°F (stable)

Which ONE of the following actions should you take in response to these indications, and WHY is the action required?

A. Manually initiate Safety Injection due to Pressurizer Low Level.

B. Manually Initiate Safety Injection due to failure of the High Steam Flow SI to actuate.

C. Check RCS subcooling table. If LESS THAN required, manually initiate Safety Injection due to Low Subcooling.

D. Transition to ES-0.1, Reactor Trip Response, Safety Injection is NOT required.

REACTOR OPERATOR EXAMINATION

A Large Break Loss of Coolant Accident has occurred. Approximately 24 hours after aligning for Cold Leg Recirculation, the Senior Reactor Operator implements EOP ES-1.4, Transfer to Hot Leg Recirculation. Which ONE of the following statements is correct regarding the reason for placing Hot Leg Recirculation in service at this time?

A. Hot leg recirculation is implemented to sweep non-condensable gasses from the reactor head region.

B. Hot leg recirculation is implemented to cool the reactor head to enable RCS depressurization without additional void formation.

C. Hot leg recirculation is implemented to refill the reactor vessel and preclude fuel rod damage at the top of the core.

D. Hot leg recirculation is implemented to prevent boron precipitation in the core.

REACTOR OPERATOR EXAMINATION

Following a Steam Generator Tube Rupture, EOP E-3, Steam Generator Tube Rupture, has been implemented. The Reactor Operator is controlling RCS temperature using the Condenser Steam Dumps. The Senior Reactor Operator directs the Reactor Operator to depressurize the RCS to LESS THAN RUPTURED SG pressure using the Pressurizer PORVs since the RCPs have been tripped.

The Reactor Operator CLOSES the steam dump valves and prepares for depressurization of the RCS. Before commencing depressurization the Reactor Operator notes the following indications:

- RCS Wide Range Cold Leg Temperatures — 480°F (increasing slowly)
- Core Exit Temperature — 488°F (increasing slowly)
- Pressurizer Level (hot calibrated) — 0%
- RCS Pressure — 1370 psig (increasing)
- Ruptured SG Level — 88% (Narrow Range - increasing)
- Ruptured SG Pressure — 1015 psig (increasing slowly)

Just prior to closing the Pressurizer PORVs the Reactor Operator notes the following indications:

- RCS Wide Range Cold Leg Temperatures — 506°F (increasing slowly)
- Core Exit Temperature — 510°F (stable)
- Pressurizer Level (hot calibrated) — 80% (increasing rapidly)
- RCS Pressure — 760 psig (increasing)
- Ruptured SG Level — 85% (Narrow Range - decreasing slowly)
- Ruptured SG Pressure — 1005 psig (decreasing slowly)

Which ONE of the following statements could explain ALL of the changes in the above indications that have occurred since the Reactor Operator commenced depressurization?

A. RCS cold leg temperature has INCREASED due to flashing of RCS fluid in the hot leg.
PZR level has INCREASED due to increased makeup flow AND voiding in the Reactor Head.
Ruptured SG level is DECREASING due to SG fluid backfilling the RCS.

B. RCS cold leg temperature has INCREASED due to REDUCED natural circulation flow rate.
Pressurizer level has INCREASED due to increased makeup flow AND voiding in the Reactor Head.
Ruptured SG level is DECREASING due to SG fluid backfilling the RCS.

C. RCS cold leg temperature has INCREASED due to REDUCED natural circulation flow rate.
Indicated Pressurizer level has INCREASED to due flashing of the fluid in the level instrument reference leg.
Ruptured SG level is DECREASING due to SG fluid backfilling the RCS.

D. RCS cold leg temperature has INCREASED due to STOPPAGE of natural circulation flow.
Pressurizer level has INCREASED due to increased makeup flow AND voiding in the Reactor Head.
Ruptured SG level is DECREASING due to steaming through the Atmospheric Steam Dump valve.

A refueling operator has just placed an irradiated fuel assembly in the containment side upender when the Refueling SRO notes that refueling cavity water level is decreasing rapidly.

Which ONE of the following actions should be directed by the Refueling SRO?

A. Disengage the manipulator from the fuel assembly and lower the upender to the fully lowered position, THEN evacuate containment.

B. Withdraw the fuel assembly from the upender and move it to the reactor, THEN evacuate containment.

C. Withdraw the fuel assembly from the upender and store it in the manipulator mast, THEN evacuate containment.

D. Disengage the manipulator from the fuel assembly and leave upright, THEN evacuate containment.

Following a Reactor Trip and Safety Injection you are verifying 480 VAC Busses energized by offsite power using EOP E-0, Reactor Trip or Safety Injection. You note the following conditions associated with the 480 VAC Busses:

- All 480 VAC Bus Normal Feeder Breakers — OPEN
- All Emergency Diesel Generators — In Service
- All 480 VAC Bus Emergency Feed Breakers — CLOSED

Which ONE of the following actions should NOT be performed when completing this IMMEDIATE ACTION step?

A. Start ONE charging pump in MANUAL at maximum speed.

B. Ensure the following MCCs - ENERGIZED
- MCC 26A
- MCC 26B
- MCC 26C

C. Reset Lighting

D. Ensure the following MCCs - ENERGIZED
- MCC 24A
- MCC 27A
- MCC 29A
- MCC 211

REACTOR OPERATOR EXAMINATION

You have been directed to coordinate the completion of a System Check Off List (COL) for the Safety Injection System. You note that the COL requires Independent Verification.

Which ONE of the following statements is TRUE regarding acceptable practice when conducting INDEPENDENT VERIFICATION of items contained in a COL that requires independent verification?

A. Operator Performing the INDEPENDENT VERIFICATION (Second Checker) may perform the verification at the same time as the First Checker and should record the AS FOUND position of each component.

B. Operator Performing the INDEPENDENT VERIFICATION (Second Checker) must perform the verification independently of the First Checker, and should reposition the component IF it is not in the required position.

C. Operator Performing the INDEPENDENT VERIFICATION (Second Checker) may perform the verification at the same time as the First Checker and should record the AS LEFT position of each component.

D. Operator Performing the INDEPENDENT VERIFICATION (Second Checker) must perform the verification independently of the First Checker, and should record the AS FOUND position of each component.

A maintenance mechanic calls the control room and reports that a valve he has just removed from the service water system has a black and white STOP TAG attached to the handwheel that requires the valve to be in the OPEN position.

Which ONE of the following actions should you take regarding this report?

A. No action is necessary since the valve was tagged OPEN. Work protected by this tagout is not affected.

B. Obtain the tagout number and report the finding to the work control center. Work protected by this tagout should cease until the discrepancy is corrected.

C. Direct the maintenance mechanic to place the tag on an adjacent valve which is in the OPEN position. Work protected by this tagout is not affected.

D. Obtain the tagout number and record the finding in the SRO log. Work protected by this is not affected since the valve was tagged in the OPEN position.

While investigating an increase in RCS leakage, preparations are made to make a containment entry using SAO 219, Containment Entry and Egress. Which of the following actions is required if entry is to be made within the crane wall on the 46' elevation?

A. Reactor Shutdown is required for all entries in the vicinity of RCS loop piping.

B. Health Physics and Operations will determine the need to reduce radiation levels by reducing power.

C. If entry is planned for less than 10 minutes within crane wall, no power reduction is required.

D. If total job exposure will exceed 5 person-rem, a power reduction to 75% is required.

During power operation the DIESEL BLDG. FIRE PROT OPERATION annunciator is activated. Which ONE of the following statements is correct regarding the response of the Emergency Diesel Generator (EDG) Building Fire Protection System?

A. EDG Building Fire Protection System is a dry line system and water spray is activated to the entire building by a deluge valve. The above alarm could mean that all of the spray nozzles have actuated.

B. EDG Building Fire Protection System is a concentrated foam system and foam is activated locally by thermostats. The above alarm could mean that the foam system has actuated.

C. EDG Building Fire Protection System is a HALON system and HALON is activated locally by thermostats. The above alarm could mean that HALON system has actuated.

D. EDG Building Fire Protection System is a wet line system and water spray is activated locally at each spray nozzle. The above alarm could mean that one of the spray nozzles has actuated.

Station policy requires that each individual follow practices that ensure that their personal radiation exposure is kept As Low As Reasonably Achievable (ALARA). Consider the following situation:

- Maintenance must be performed in an area where the general area radiation levels are 5 mr/hr γ and 100 mr/hr β
- The maintenance activity is estimated to take 1 person 1 hour
- Most of the radiation is due to contamination of the floor in the area.

Select the ALARA practice that would result in the lowest achievable radiation exposure (total Person-Rem) for this job?

A. Equip worker with a plastic face shield and protective clothing against contamination.

B. Decontaminate the area (requires 2 people and 45 minutes) before commencing work.

C. Cover the floor in plastic sheeting (requires 2 people 30 minutes) before commencing work.

D. Cover floor with lead blankets (requires 3 people 30 minutes) before commencing work.

REACTOR OPERATOR EXAMINATION

You are performing a Reactor Startup. Your watch relief from the on-coming shift has arrived just as you are ready to withdraw the control banks to approach criticality.

Which ONE of the following statements describes an acceptable practice when conducting shift turnover during a Reactor Startup?

A. Conduct shift turnover at the flight panel as you continue the reactor startup ensuring that you are not distracted from monitoring neutron flux.

B. Do not begin control bank withdrawal. Conduct shift turnover after neutron flux has stabilized.

C. Allow your relief to continue the reactor startup as you relay pertinent watch turnover information to him.

D. Continue the startup while the rest of the crew conducts watch turnover.

While performing a plant cooldown to 350°F using POP 3.3, Plant Cooldown, you notice a handwritten notation in the right margin of the page that contains the following information TPC 96-153.

Which ONE of the following statements is correct regarding the significance of this notation?

A. The notation refers to a Temporary Procedure Change. I must refer to the Temporary Procedure Change Log Book located in the control room to perform the associated step.

B. The notation refers to a Temporary Procedure Change. The step that is affected will be lined out and the correct information entered in the body of the procedure.

C. Handwritten notations are not permitted in plant procedures. I should notify the SRO and obtain a clean copy of the procedure.

D. The notation refers to a Temporary Procedure Change. I must refer to the Temporary Procedure Change Request Form located at the beginning of the procedure to perform the associated step.

REACTOR OPERATOR EXAMINATION

An event has occurred requiring notification of the NRC within 1 hour. Which ONE of the following communications systems should be used to perform this notification?

A. Emergency Notification System (ENS) Phone

B. Radiological Emergency Communication System (RECS)

C. Microwave Phone Line

D. State Emergency Management Office (SEMO) Radio

Which ONE of the following is the minimum Emergency Plan Classification which would require "Level 2" staffing?

A. Site Area Emergency

B. General Emergency

C. Alert

D. Notification of Unusual Event

Which ONE of the following statements correctly identifies the tagout protection that must be provided if a worker is going to work in a tank that is connected to any system?

A. Each source of energy must be isolated by TWO CLOSED, and TAGGED isolation valves or by ONE CLOSED, LOCKED and TAGGED isolation valve.

B. Each source of energy must be isolated (if available) by TWO CLOSED, LOCKED and TAGGED isolation valves.

C. Each source of energy must be isolated (if available) by TWO CLOSED, LOCKED and TAGGED isolation valves or a safety person must be stationed at the tank opening to assist in emergency egress.

D. Each source of energy must be isolated by at least ONE CLOSED, LOCKED and TAGGED isolation valves, or a safety person must be stationed at the tank opening to assist in emergency egress.

Which ONE of the following gas samples would indicate that the gas space in the associated equipment contains a potentially flammable mixture?

A. Volume Control Tank	97% Hydrogen, 1% Oxygen, 2% Nitrogen
B. 23 CVCS Holdup Tank	26% Hydrogen, 0% Oxygen, 74% Nitrogen
C. 22 Large Gas Decay Tank	7% Hydrogen, 17% Oxygen, 76% Nitrogen
D. Main Generator	94% Hydrogen, 1% Oxygen, 5% Nitrogen

REACTOR OPERATOR EXAMINATION

Choose the selection that correctly completes the following statement:

The Condensate Storage Tank area is a(n) _____, the Simulator Building is in the _____ and the Main Turbine is in the _____.

A. Isolation Zone, Owner Controlled Area, Protected Area

B. Vital Area, Isolation Zone, Protected Area

C. Exclusion Zone, Owner Controlled Area, Protected Area

D. Vital Area, Owner Controlled Area, Protected Area

REACTOR OPERATOR EXAMINATION

Which ONE of the following Central Control Room log sheet entry examples would require that the entry be RED circled and explained in the remarks section?

A. A NON-Technical Specification reading exceeds NORMAL limits specified on log sheet.

B. An error is made entering a reading and a correction to the entry is required.

C. Reading exceeds MIN/MAX limit specified on log sheet.

D. Reading is taken one hour late due to startup activities.

Reactor Operator Exam Answer Key - 3/8/97

1.	C	26.	D	51.	A	76.	A
2.	B	27.	B	52.	B	77.	B
3.	C	28.	C	53.	B	78.	D
4.	C	29.	A	54.	B	79.	A
5.	C	30.	C	55.	B	80.	C
6.	A	31.	B	56.	C	81.	B
7.	D	32.	D	57.	A	82.	C
8.	C	33.	B	58.	D	83.	A
9.	B	34.	C	59.	B	84.	D
10.	D	35.	A	60.	C	85.	B
11.	A	36.	D	61.	D	86.	A
12.	D	37.	B	62.	A	87.	C
13.	C	38.	B	63.	C	88.	D
14.	B	39.	D	64.	D	89.	B
15.	B	40.	C	65.	B	90.	B
16.	C	41.	A	66.	C	91.	D
17.	D	42.	B	67.	B	92.	A
18.	B	43.	D	68.	C	93.	B
19.	A	44.	D	69.	D	94.	D
20.	C	45.	A	70.	A	95.	A
21.	D	46.	D	71.	C	96.	C
22.	B	47.	D	72.	B	97.	B
23.	D	48.	C	73.	D	98.	C
24.	B	49.	B	74.	B	99.	D
25.	A	50.	D	75.	B	100.	C

ATTACHMENT 2

SIMULATION FACILITY REPORT

FACILITY LICENSEE: Consolidated Edison

FACILITY DOCKET NO: 50-247

Operating Tests administered: 3/18 1997

This form is to be used only to report 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, which may be used in future evaluations. No licensee action is required in response to these observations.

No discrepancies in simulator performance were noted.