U. S. NUCLEAR REGULATORY COMMISSION
REGION I

DOCKET/REPORT NOS: 50-423/95-43 (OL)

LICENSEE: Northeast Nuclear Energy Company
Hartford, Connecticut

FACILITY: Millstone Nuclear Power Station, Unit 3

LOCATED AT: Waterford, Connecticut

EXAMINATION DATES: November 20-December 1, 1995

EXAMINER: _____ 1/23/96
Paul Bissett, Senior Operations Engineer Date
Operator Licensing and
Human Performance Branch
Division of Reactor Safety

APPROVED BY: _____ 1/22/96
Glenn W. Meyer, Chief Date
Operator Licensing and
Human Performance Branch
Division of Reactor Safety

EXECUTIVE SUMMARY

EXAMINATION REPORT NO. 50-423/95-43

Operations

Initial examinations were administered to four reactor operator (RO) and six senior reactor operator (SRO) candidates at Millstone Unit 3. Of the six SRO candidates, three were instant SROs and three were SRO upgrades. Nine of 10 candidates passed all portions of the written and operating examination and, thus, were issued licenses. One SRO upgrade candidate failed the written portion of the examination.

Strengths were noted by the examiners in the candidates' performance during the operational portion of the examination, especially in the area of communications and command and control. Based upon observation of the simulator examinations, it was noted that crew communications and teamwork, for the most part, were strong and effective. The candidates' knowledge of plant procedures and familiarity of plant equipment was evident as demonstrated during the job performance measure (JPM) portion of the examination. No safety-significant deficiencies were identified during this examination.

DETAILS

1.0 TYPE OF EXAMINATIONS: Initial

Examination Results:

	RO Pass/Fail	SRO Pass/Fail
Written	4/0	5/1
Simulator	4/0	6/0
Walk-through	4/0	6/0
Overall	4/0	5/1

2.0 PREEXAMINATION REVIEW

The written examinations, job performance measures (JPMs) and simulator scenarios were developed by Millstone Unit 3 (MP3) representatives in accordance with generic letter guidelines, GL 95-06, "Changes in the Operator Licensing Program." The exam development team was comprised of two MP3 and one Seabrook training representatives and one MP3 operations representative. All individuals signed onto a security agreement once the development of the examination commenced. The NRC subsequently reviewed the proposed examination and relayed their review comments to the MP3 facility representative, whereupon MP3 incorporated the NRC's comments and resubmitted a final version of the examination. This final submission was subsequently reviewed and approved by the NRC for eventual administering to the candidates. A desktop review of the simulator scenarios and JPMs was also performed by the NRC in the regional office, and the scenarios were subsequently run on the site-specific simulator, prior to their administration, to familiarize all examiners with all desired operator actions.

3.0 EXAMINATION OVERVIEW

3.1 Written Examination

The written examinations were administered by the facility at the Millstone Nuclear Training Center on November 20, 1995, in accordance with the guidelines provided with GL 95-06. The reactor operator and senior reactor operator examinations were developed, as previously mentioned, by MP3 representatives in accordance with the guidelines of 10 CFR 55.41, 55.43, and NUREG-1022, "Examiners Handbook for Developing Operator Licensing Written Examinations." Both written examinations consisted of 100 questions written in multiple choice format. The written examinations and answer keys are enclosed as Attachments 1 and 2 to this report.

Following the administration of the written examination, an additional review was performed by MP3 representatives to again validate any questions incorrectly answered by the candidates. From this review, it was determined that two questions were in error. One question had the incorrect answer listed in the answer key, and the other question was determined to have two correct answers. MP3 representatives provided to the NRC written comments,

along with recommended changes in regard to these two written examination questions. The NRC agreed with the facility's comments and recommended changes, which are provided in Attachment 3.

3.4 Operating Test

The operating tests were administered by the NRC from November 28 through December 1, 1995. The operating tests consisted of at least two dynamic simulator scenarios and 10 JPMs for each of the ROs and instant SROs, and five JPMs for each of the three SRO upgrade candidates. Two oral questions were asked at the completion of each JPM. All candidates were also examined concerning administrative requirements at Millstone 3 station, in addition to the scenarios and JPMs.

3.4.1 Dynamic Simulator Examination

The candidates were divided into crews for the dynamic simulator examination. Each crew was made up of two RO positions and one SRO position. All candidates participated in at least two scenarios, and the ROs rotated between the reactor operator and the balance of plant board positions. The instant SROs stood the position of the supervisory control operator (SCO) in one scenario and the RO position during the other scenario. All candidates were successful in completing the scenarios under which they were examined.

Communications were accomplished effectively, for the most part, throughout the conduct of the scenarios for all crews involved. Command and control was especially strong, together with questioning attitudes and suggestions by team members. Participation of all members as part of a crew was evident throughout all scenarios.

A generic weakness was identified during the loss of feedwater scenarios. Once feedwater was reestablished via a main feedwater pump, the crew appeared to be unaware of the significant flowrate that subsequently resulted. In two instances, a safety injection actuation occurred as a result of a rapid cooldown of the RCS that, in turn, caused a feedwater isolation signal. Although, in both instances, feedwater was eventually reestablished, an unnecessary delay had resulted. The examiners stated at the exit meeting that either the candidates did not have a good comprehension of the ensuing feedwater flowrates and subsequent necessary control actions or a simulator modeling problem existed. The MP3 representatives acknowledged the NRC's concern and indicated that they would further evaluate this concern.

3.4.2 Job Performance Measures (JPMs)

In the JPM area of examination, the candidates exhibited excellent performance on the seven JPMs administered either on the simulator or in the control room. The examiners noted also that the ROs exhibited plant familiarity while locating various valves and switches during the performance of the three in-plant JPMs. All candidates demonstrated a through working knowledge of personnel and radiological safety practices.

A generic weakness was identified during one JPM in which two candidates demonstrated unfamiliarity with valve control switches. In this instance, they were not aware of the fact that some valves require the switch to continually be held in the closed or open position during the full stroke of the valve; otherwise, the valve will not reposition.

4.C SUMMARY OF NRC COMMENTS MADE AT THE EXIT MEETING

The NRC expressed appreciation to the training and operations staves for providing the necessary assistance during the examination process. The NRC examiners also commented that the examination administered was a well-developed, performance-based, safety-oriented examination.

The NRC also stated that emphasis needed to be placed on the development of JPM follow-up and administrative questions. For these two areas, too many of the originally-submitted questions had been either direct look-up or simple memory or recall questions.

5.0 PERSONNEL CONTACTED

Northeast Nuclear Energy Company:

* J. Arsenault, MP3 LOIT Coordinator
* M. Brown, Director, Nuclear Training
* D. Lazarony, MP3 Supervisor, Operator Training
* R. Lueneburg, MP3 Senior Instructor
* B. Parrish, MP3 Assistant Supervisor, Operator Training
* B. Pinkowitz, MP3 Operations Manager
* J. Smith, Manager, Operator Training (MP3/CY)
* D. Tilton, MP3 Instructor

NRC Examiners:

S. Barr, Operations Engineer/Examiner
* P. Bissett, Senior Operations Engineer/Examiner
* J. D'Antonio, Operations Engineer/Examiner
* M. Modes, Acting Chief, Operator Licensing and Human Performance Branch
J. Prell, Senior Operations Engineer/Examiner

*Denotes those personnel present at the exit meeting on December 1, 1995.

Other plant, technical, training and management personnel were contacted during the course of the examination.

Attachments:
1. Written RO Examination and Answer Key
2. Written SRO Examination and Answer Key
3. MP3 Written Examination Comment Summary and NRC Resolution
4. Simulation Facility Report

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ATTACHMENT 1

WRITTEN RO EXAMINATION AND ANSWER KEY

Reactor Operator
Licensing
Examination

Question:

The following conditions are given:

- A shutdown bank "D" rod has dropped into the core
- A dropped rod recovery is in progress per AOP 3552
- Rod recovery has commenced

Which one of the following describes the status of the rod control Urgent Failure alarm under these circumstances?

A. Urgent Failure alarm is present since all the lift coils are disconnected for the group.

B. Urgent Failure alarm is present since the bank overlap unit is bypassed during rod recovery.

C. Urgent Failure alarm is not present because shutdown bank D contains only one group of rods.

D. Urgent Failure alarm is not present because shutdown banks C, D & E have no master cycler input.

Answer:

C. Urgent Failure alarm is not present because shutdown bank D contains only one group of rods.

Reference: AOP 3552 Basis Document page 16

Justification: Urgent failure occurs when trying to move a group of rods with all lift coil disconnects open. With all lift coils in the unaffected group disconnected, a regulation failure occurs. S/D banks C, D & E have only one group of rods per bank, so all lift coils will NOT be disconnected on dropped rod recovery.

K/A: 001050A2.01 3.7/3.9

Objective: ROD06C

Comments: NRC change- modify to remove negatives in stem and distractors

Question:

Which one of the following inputs determines the rod insertion limits as calculated by the Rod Insertion Monitor?

A. process computer calorimetric

B. auctioneered high power range NI power

C. turbine impulse pressure

D. auctioneered high loop ΔT

Answer:

D. auctioneered high loop ΔT

Reference: Lesson Plan RPI-01-C Page 12

Justification: $RIL = K_1 (Tavg - 557) + K_2 (\%\Delta T) + K_3$

K/A: 001000K5.04 4.3/4.7

Objective: RPI06C

Comments: NRC change- be consistant using turbine impulse pressure vs 1st stage pressure

Question:

Which one of the following is the reason that RCP #1 seal leakoff is isolated at RCS pressures below 125 psia?

A. Leakoff flow decreases at low pressures so #1 seal leakoff is isolated to force more flow through #2 seal.

B. Backflow from the VCT through the seal leakoff line could flush contaminants into the seals.

C. Controlled Leakage limits may be exceeded due to excessive seal injection flow at low pressures.

D. Leakoff flow instruments are not accurate at low pressures and excessive leakoff could go undetected.

Answer:

B. Backflow from the VCT through the seal leakoff line could flush contaminants into the seals.

Reference: RCP003C Reactor Coolant Pump

Justification: The VCT may be at higher pressure than the RCS and backflow through the seal leakoff line may result. This flow is not subject to filtration (no seal injection filters on this line) so VCT contaminants may be introduced into the seals and result in mechanical damage when the pump is restarted.

K/A: 003000K1.03 3.3/3.6

Objective: RCP06C

Comments : SRO #4

Question:

Which one of the following describes the status of RCP lower radial bearing cooling when seal injection flow is isolated?

A. Lower radial bearing cooling is lost

B. RPCCW cools the oil which cools the bearings

C. thermal barrier heat exchanger coils cool the bearings

D. #1 seal leakoff flow passes around the bearings

Answer:

C. thermal barrier heat exchanger coils cool the bearings

Reference: RCP003T Reactor Coolant Pumps Text Pg. 19-20

Justification: Bearings are water-cooled, normally from seal injection flow. If seal injection is lost, thermal barrier HX cools RCS flow which, in turn, cools the lower radial bearing.

K/A: 003000K4.04 2.8/3.1

Objective: RCP06C

Comments: NRC change- modify question from normal cooling supply to cooling supply when seal injection lost.

Question:

4160V Bus 34D is de-energized due to a bus differential. Which one of the following groups contains loads which have all been lost?

A. "B" AFW Pump, 480V Load Center 32Y, "B" RPCCW Pump

B. 480V Load Center T, "B" Safety Injection Pump, "B" RHR Pump

C. "C" Charging Pump, "B" Cont. Recirc. Pump, 480V Load Center 32R

D. 480 V Load Center 32X, "D" Service Water Pump, "B" Charging Pump

Answer:

D. 480 V Load Center 32X, "D" Service Water Pump, "B" Charging Pump

Reference: CVCS System Description
Print 12179-EE-1M

Justification: Bus 34D loads include most "B" and "D" loads, "C" charging pump and 480V load centers 32U, V, W, X

K/A: 004000K2.03 3.3/3.5

Objective: CHS03C

Comments: SRO #6
NRC change- change distractors away from "A" and "C" designated loads

Question:

Which one of the following situations requires a **normal** RCS boration?

A. inadequate shutdown margin exists as determined by OP-3209B, Shutdown Margin

B. pressurizer boron concentration is 50 ppm greater than RCS boron concentration

C. control bank D rods approach their Rod Insertion low limit during normal operation

D. 1/M plot predicts that criticality will be achieved outside the ECP band

Answer:

C. control bank D rods approach their rod insertion limit during normal operation

Reference: OP 3204 Rev. 11

Justification:
A. This requires immediate boration IAW AOP-3566
B. Requires forcing spray flow by turning on heaters
C. When control banks are inserted to insertion limits they can no longer be a source of negative reactivity addition for NORMAL operation. Borating the RCS will add negative reactivity and allow backing the control rods out away from the insertion limits.
D. Requires full insertion of control bank rods.

K/A: 004000A4.01 3.8/3.9

Objective: CHS06C

Comments: NRC change- replace distractor "A"

Question:

The following occurs on Unit-3:

- MSIVS shut resulting in an automatic reactor trip signal.
- Only one reactor trip breaker opens automatically or manually.
- Several MS safety valves fail open resulting in an automatic safety injection.

It is desired to reset SI and take manual control of ECCS equipment. Which one of the following describes the ability to reset SI?

A. SI cannot be reset until both reactor trip breakers are opened to actuate P-4 on both trains

B. SI can be reset as soon as the SI timing relays have timed out

C. SI cannot be reset because the actuation signal is still present

D. SI can be reset after timing relays time out and pressurizer pressure increases > P-11

Answer:

A. SI cannot be reset until both reactor trip breakers are opened to actuate P-4 on both trains

Reference: RPS012T pages 73-77

Justification: P-4 is an input to the SI reset circuitry. P-4 will not be satisfied if the trip breaker and bypass breaker are not open. In this example, P-4 is not satisfied as one trip breaker is still shut.

K/A: 013000A4.02 4.3/4.4

Objective: RPS04C

Comments: NRC change- alter question from having both RTBs fail to open to just one RTB fails to open.

Question:

Which one of the following automatically occurs when turbine impulse pressure increases above the P-13 setpoint?

A. main steam line isolation on negative pressure rate is unblocked

B. low pressurizer pressure reactor trip is unblocked

C. low steam line pressure SI is unblocked

D. steam dump system operation is unblocked

Answer:

B. low pressurizer pressure reactor trip is unblocked

Reference: RPS012T Pages 78-79

Justification: When turbine impulse pressure exceeds P-13, the trips automatically blocked by P-7 are automatically reinstated. P-11 blocks the low steamline pressure SI and the MSLI on negative rate. Steam dump system operation is blocked/unblocked via P-12.

K/A: 013000A3.01 3.7/3.9

Objective: RPS04C

Comments: NRC change- replace distractor "D", & make all distractors "unblock".

Question:

As reactor power increases above 10% during a plant startup, which one of the following occurs?

A. power range NI low trip setpoint is automatically blocked

B. automatic reactor trip on 1 of 4 RCS loops low flow reinstated

C. manual block of intermediate range (IR) rod stop is permitted

D. high pressurizer pressure reactor trip is automatically reinstated

Answer:

C. manual block of intermediate range (IR) rod stop is permitted

Reference: NIS015T

Justification: A. This is a manual action which is permitted
B. 2/4 low flow trip is reinstated by P-7
C. Correct
D. This trip function always activated

K/A: 015000K6.01 2.9/3.2

Objective: NIS02C

Comments: SRO #10

Question:

It has been decided to trip the "A" RCP due to increasing vibration levels. Plant power is just below the P-8 setpoint. Which one of the following actions should be taken to prevent a reactor trip when the RCP is tripped?

A. shut the pressurizer spray valve associated with the loop the RCP is on

B. lower power below 30% since the temperature increase following the pump trip will cause the NIs to read high

C. insert rods in manual to reduce temperature in anticipation of the post-trip RCS heatup

D. manually raise affected loop steam generator level to 65% in anticipation of shrink

Answer:

D. manually raise affected loop steam generator level to 65% in anticipation of shrink

Reference: AOP 3554 Rev. 5

Justification: When the affected loop RCP is taken out of service, heat transfer in the affected S/G will decrease dramatically. This decrease in heat transfer has the same effect as a rapid power decrease on the affected loop and affected S/G level will shrink dramatically.

K/A: 015000G007 3.3/3.4

Objective: RCS07C

Comments: NRC change- adjust wording of distractor "B"

Question:

A small break LOCA and concurrent loss of all AC power has occurred on Unit-3. ECA-0.0 has been implemented. The plant process computer failed after 30 minutes and an I & C technician is stationed at the ICC cabinets to monitor core exit thermocouple temperature. He reports that the highest CET temperature is 727°F and all CETs indicate > 720°F. Which one of the following is the correct response to this situation?

A. open one PORV to reduce RCS pressure to inject accumulators

B. continue to attempt to restore power while monitoring CET temperature

C. transition to FR-S.1 if startup rate is positive during S/G depressurization

D. concurrently implement FR-C.2, Response to Degraded Core Cooling

Answer:

B. continue to attempt to restore power while monitoring CET temperature

Reference: ECA-0.0 Rev. 10 Note prior to Step 1

Justification: This CET temperature would normally indicate an "orange path" transition to FR-C.2. However, because no AC electrical buses are available, mitigation strategies of FR-C.2 will not be effective. Concurrent implementation of FRPs and ECA-0.0 are not allowed. One AC emergency bus must be energized and a transition made to ECA-0.1 or ECA-0.2 before FRP implementation is allowed. ECA-0.0 step 3 ensures PORVs are shut, step 17 has operators depressurize S/Gs to minimize break flow.

K/A: 017000G015 3.8/4.0

Objective: AOO04C

Comments: NRC change- distractors "C" and "D" replacement

Question:

Unit-3 tripped on low-low S/G level due to a loss of all feedwater. The operating crew has just transitioned to FR-H.1 to restore S/G level and feed flow. While implementing FR-H.1 the STA notes that core exit thermocouple temperatures rise steadily into superheat conditions. Which one of the following is occurring?

A. S/G level is dropping below the top of the U-tubes

B. phase separation of the reactor coolant

C. voiding in the reactor vessel head region

D. core uncovery

Answer:

D. core uncovery

Reference: MCD Lesson Plan pages 10-11

Justification: Level dropping below the top of the S/G U-tubes and voiding in the vessel head will not result in appreciable CET temperature increases. Reflux cooling is an effective core cooling mechanism under certain specific circumstances and will not result in CET increases. Fuel uncovery will result in rapidly increasing fuel temperatures and superheat conditions as steam flow past the uncovered fuel becomes the core heat sink..

K/A: 017020A2.02 3.6/4.1

Objective: ACA07C

Comments: SRO #11
NRC change- replace distractor "B"

Question:

Which one of the following automatically occurs as a result of a safety injection signal?

A. "C" CAR cooling supply shifts from chilled water to RPCCW

B. "A" CAR fan receives a "stop" signal

C. "B" CAR fan receives a "start" signal

D. "A" CAR cooling supply shifts from RPCCW to chilled water

Answer:

C. "B" CAR fan receives a "start" signal

Reference: Lesson Plan CVS-01-C
Print EM-122B

Justification: "C" CAR cooling does not have an alternate supply.
"A" CAR cooling shifts to RPCCW on a containment phase A isolation.
"A" CAR fans trip on CDA

K/A: 022000A3.01 4.1/4.3

Objective: RPS05C

Comments: SRO #12

Question:

During adverse containment conditions, pressurizer level indication will read differently from actual pressurizer level. Which one of the following correctly describes this discrepency?

A. Indicated pressurizer level reads higher than actual level due to the increased density of the reference leg.

B. Indicated pressurizer level reads higher than actual level due to the decreased density of the reference leg.

C. Indicated pressurizer level reads lower than actual level due to the increased density of the reference leg.

D. Indicated pressurizer level reads lower than actual level due to the decreased density of the reference leg.

Answer:

B. Indicated pressurizer level reads higher than actual level due to the decreased density of the reference leg.

Reference: Mitigating Core Damage Section 7 "Instrument Qualification"

Justification: As containment temperature increases, so does the temperature of the fluid in the reference leg of the Pzr level transmitter. As this temperature increases, the density decreases. Since this transmitter is calibrated for the density of a 100°F reference leg, the D/P between the reference and variable legs decreases. This lower D/P is interpreted by the instrument to be a higher level (less level discrepency between variable/reference legs).

K/A: 022000K3.02 3.0/3.3

Objective: PPL06C

Comments:

Question:

The plant is operating at full power with two condensate pumps in operation. Which one of the following will cause the standby condensate pump to start?

A. low condensate header discharge pressure

B. low feed pump suction pressure

C. CONDENSATE PUMP FLOW LOW annunciator actuates

D. either running condensate pump trips

Answer:

D. either running condensate pump trips

Reference: CNM056T pages 7-8

Justification: This is the way the control circuit is designed.

K/A: 056000K1.03 2.6/2.6

Objective: CNM04C

Comments: NRC change- replace distractor "C"

Question:

Extraction steam non-return valves automatically shut on a turbine trip. Which one of the following is the reason for this automatic action?

A. reduces the temperature transients within the feedwater heaters

B. protects the feedwater heater shell side from overpressurization

C. reduces turbine vibration due to unbalanced loading of turbine blades during coastdown

D. prevents turbine overspeed due to steam backflow through the extraction lines

Answer:

D. prevents turbine overspeed due to steam backflow through the extraction lines

Reference: LP ESS-01-C

Justification: Steam can be trapped at significantly high pressures and volumes in the feedwater heater shells and piping after turbine stop and control valves shut. Similar to the function of the turbine intercept/intermediate stop valves, the extraction steam non-return valves shut to prevent the backflow of this volume of steam from re-entering the turbine and causing an overspeed condition.

K/A: 056000G001 2.6/2.8

Objective: ESS03C

Comments: SRO #15

Question:

Unit-3 is operating at 100% power when a feedline rupture inside containment results in low-low S/G levels and automatic safety injection actuation on high containment pressure. Which one of the following accurately describes the features of AFW pump control?

A. If "A" motor-driven AFW pump is in LOCAL, it will automatically start on the safety injection signal.

B. "A" motor-driven AFW pump can be stopped by depressing the "Aux FW Isol Reset" pushbutton on MB 5.

C. If "A" motor-driven AFW pump is running in LOCAL, all automatic trips are bypassed.

D. "A" motor-driven AFW pump will trip if a low lubricating oil pressure occurs.

Answer:

D. "A" motor-driven AFW pump will trip if a low lubricating oil pressure occurs.

Reference: Lesson Plan FWA-01-C
Text FWA061T
Print 25212-39001 SH4016

Justification:
A. When "A" AFW pump is in LOCAL, it will not auto-start.
B. This will only work if it is running as a result of low-low S/G level only, not SI actuation.
C. Automatic trips are not bypassed in LOCAL

K/A: 059000K3.02 3.6/3.7

Objective: FWA04C

Comments: NRC change- limit distractors to "A" AFW pump

Question:

Which one of the following alone will result in a signal which will shut the feedwater isolation valves?

A. P-4

B. Low Tavg

C. High Steam Pressure Negative Rate Main Steam Line Isolation

D. P-14

Answer:

D. P-14

Reference: Print 25212-39001 SH4014

Justification: Plant Design

K/A: 059000K4.19 3.2/3.4

Objective: RPS04C

Comments:

Question:

OP 3322, "Auxiliary Feedwater System", places limits on motor-driven AFW pump amperage. Which one of the following is this limit designed to prevent?

A. overloading the 4160 volt bus supplying the pump

B. overloading the EDG during pump operation on LOP

C. motor-driven AFW pump runout

D. overheating AFW pump motor

Answer:

C. motor-driven AFW pump runout

Reference: OP 3322 Rev. 16

Justification: Procedural requirement

K/A: 061000K2.02 3.7/3.7

Objective: FWA05C

Comments: NRC change- add title to procedure number in stem.

Question:

Which one of the following represents the suction source for AFW in order of preference during an accident situation?

A. Demineralized Water Storage Tank, Service Water, Condensate Storage Tank

B. Condensate Storage Tank, Demineralized Water Storage Tank, Service Water

C. Demineralized Water Storage Tank, Condensate Storage Tank, Service Water

D. Condensate Storage Tank, Demineralized Water Storage Tank, Condensate Surge Tank

Answer:

C. Demineralized Water Storage Tank, Condensate Storage Tank, Service Water

Reference: OP 3322 Rev. 15 Section 7.6

Justification: DWST and CST are the cleanest sources of water which can be readily aligned. Service water is an impure source of water which must be aligned via spool pieces.

K/A: 061000A1.04 3.9/3.9

Objective: FWA03C

Comments: NRC change- rearrange the order of distractor "D"

Question:

Which one of the following groups contains conditions which will all cause 3LWS-HV77, Waste to Discharge Tunnel Stop Valve to automatically shut?

A. high liquid waste discharge flowrate, loss of air, high liquid waste discharge radiation level

B. failure of liquid waste radiation monitor LWS-RE70, loss of air, high liquid waste discharge radiation level

C. loss of air, failure of liquid waste radiation monitor LWS-RE70, high liquid waste discharge flowrate

D. high liquid waste discharge radiation level, high liquid waste discharge flowrate, failure of liquid waste radiation monitor LWS-RE70

Answer:

B. failure of liquid waste radiation monitor LWS-RE70, loss of air, high liquid waste discharge radiation level

Reference: Lesson Plan LWS-01-C
Print 12179-EM-106A

Justification: Plant Design

K/A: 068000A3.02 3.6/3.6

Objective: LWS06C

Comments:

Question:

Containment radiation monitors RMS41-1 & RMS42-1 automatically isolate containment purge and exhaust on high radiation levels. Which one of the following events is this designed to protect against?

A. large break LOCA

B. loss of shutdown cooling/RCS inventory

C. letdown line leak

D. fuel handling accident

Answer:

D. fuel handling accident

Reference: RMS073T Pages 6-7

Justification: Containment purge and exhaust system is only in service during modes 5-6 to provide ventilation during refueling outages and are specifically designed to isolate in the event of a dropped or damaged fuel assembly.

K/A: 072000K4.01 3.3/3.6

Objective: RMS04C

Comments: SRO #19

Question:

Which one of the following groups contains conditions which would all require evacuation of personnel from the containment?

A. loss of shutdown cooling, fuel handling accident, refueling cavity seal failure

B. fuel handling accident, loss of shutdown cooling, high RCS activity

C. refueling cavity seal failure, high RCS activity, loss of shutdown cooling

D. high RCS activity, refueling cavity seal failure, fuel handling accident

Answer:

A. loss of shutdown cooling, fuel handling accident, refueling cavity seal failure

Reference: AOPs 3553, 3572, 3505, 3502

Justification: All the listed events involve an imminent danger to people inside containment except for high RCS activity.

K/A: 072000G002 3.1/3.6

Objective: CVS05C

Comments: NRC comment- remove non-essential from stem.

Question:

The following plant conditions exist:

- mode 5 following normal plant cooldown
- no RCPs running

Under which of the following conditions is it permissible to start a reactor coolant pump?

A. RCS temperature is 174°F with RHR cooling in service and the reactor coolant system is solid.

B. RCS temperature is 158°F and cold overpressure protection is established using both RHR suction relief valves.

C. RCS temperature is 155°F and RCP seal injection flows are 6 gpm/pump and leakoff flows are all 0.1 gpm

D. The only operable reactor coolant pump tripped 20 minutes ago and the SCO requests you to restart it immediately.

Answer:

B. RCS temperature is 158°F and cold overpressure protection is established using both RHR suction relief valves.

Reference: OP 3201 Rev. 10 General Precautions
OP 3301D Rev. 10

Justification: Procedures do not allow starting an RCP without a bubble in the pressurizer. RCP starting duty requirements call for 30 minutes cooling time before restarting a tripped pump. Higher seal injection and leakoff flows are required.

K/A: 002000A1.05 3.4/3.7

Objective: RCS05C

Comments: SRO #21

Question:

Unit 3 is in mode 2 with power at 3% and slowly increasing. Tavg is 543°F. Which one of the following actions is required?

A. Restore Tavg to ≥ 551°F within 15 minutes

B. Restore Tavg to ≥ 551°F within 30 minutes

C. log Tavg once every 30 minutes until Tavg ≥ 561°F

D. log Tavg once an hour until Tavg ≥ 561°F

Answer:

A. Restore Tavg to ≥ 551°F within 15 minutes

Reference: L.C.O. 3.1.1.4

Justification: Technical Specification

K/A: 002000G005 3.6/4.1

Objective: RCS08C

Comments: SRO #22

Question:

Which one of the following describes the basis for the 2200°F limit on maximum peak cladding temperature?

A. maintains clad temperature 500°F below the cladding melting point

B. zirc-water reaction becomes accelerated at temperatures > 2200°F

C. zirconium undergoes a metallic phase change at temperatures > 2200°F

D. clad temperatures higher than this correlate to fuel melt temperatures at the fuel centerline

Answer:

B. zirc-water reaction becomes accelerated at temperatures > 2200°F

Reference: 10CFR50.46
Westinghouse TAA Text

Justification: Zircalloy cladding melt temperature is approximately 3300°F. Zirconium does not undergo a metallic phase change at 2200°F. Fuel melt temperature is > 5000°F.

K/A: 006000G010 3.4/3.7

Objective: TAA05C

Comments:

Question:

Safety Injection accumulator #2 outlet check valve has developed a leak such that accumulator level needs to be lowered once per day to stay within Technical Specifications limits. Which one of the following will also occur as a result of this problem?

A. excessive thermal stresses on the accumulator outlet line

B. accumulator boron concentration decreases

C. entry into Technical Specifications Action Statement for controlled leakage

D. accumulator boron concentration increases

Answer:

B. accumulator boron concentration decreases

Reference: Technical Specification LCO 3.5.1 Surveillance Requirements

Justification: The RCS has a lower boron concentration than the 2600 - 2900 ppm conentration required by the technical specifications for the accumulators. As the RCS leaks into the accumulator, the accumulator boron concentration becomes diluted. Unless the accumulator is drained and refilled from a source of water with a higher boron concentration, it will slowly dilute down below technical specifications limits.

K/A: 006000A1.02 3.0/3.6

Objective: RCS08C

Comments: NRC change- replace distractor "C"
SRO #23

Question:

Which one of the following conditions would be caused by the master pressure controller output failing to 0%?

A. pressurizer spray valves open and control heaters go to zero output

B. pressurizer PORV PCV 456 opens and pressurizer spray valves open

C. backup heaters energize and pressurizer spray valves shut

D. control bank heaters fully energize and pressurizer spray valves shut

Answer:

A. pressurizer spray valves open and control heaters go to zero output

Reference: A71771T Section 3.2

Justification: Master pressure controller output failing to zero is the same as the input signal being greater than the setpoint signal. The master pressure controller sees a high pressurizer pressure and responds to try and lower pressure by opening spray valves and de-energizing heaters. PCV 456 will not open as it is controlled by the secondary control channel.

K/A: 010000K3.01 3.8/3.9

Objective: PPL04C

Comments: NRC change- include both spray valves in distractors
SRO #24

Question:

Which one of the following describes the operation of the pressurizer heater **control group**?

A. When the control switch is in AUTO after OFF, the heaters will energize in response to master pressure controller output.

B. When the control switch is in AUTO after ON, the heaters will de-energize in response to a low pressurizer level.

C. When the control switch is in AUTO after OFF, the heaters will energize in response to a pressurizer insurge.

D. When the control switch is in AUTO after ON, the heaters will energize in response to backup pressure control channel output.

Answer:

B. When the control switch is in AUTO after ON, the heaters will de-energize in response to a low pressurizer level.

Reference: PPL010T

Justification: Pzr heater control switch is ON/OFF spring return to AUTO.

A. AUTO after OFF opens the heater breaker and will not shut it until ON is selected.
B. Correct
C. AUTO after OFF opens the heater breaker and will not shut it until ON is selected.
D. Energizes in response to master pressure controller output.

K/A: 011000K6.03 2.9/3.3

Objective: PPL04C

Comments: SRO #25

Question:

Which one of the following statements accurately describes how the overpower ΔT (OPΔT) trip setpoint is developed?

A. Auctioneered high NI power and loop Tavg determine the overall OPΔT setpoint

B. Auctioneered high NI power and loop ΔT determine the OPΔT setpoint for that loop

C. Loop Tavg and rated thermal power ΔT determine the OPΔT setpoint for that loop

D. Tavg and auctioneered low pressurizer pressure determine the overall OPΔT setpoint

Answer:

C. Loop Tavg and rated thermal power ΔT determine the OPΔT setpoint for that loop

Reference: RPS012T pages 60-61
TS Table 2.2-1

Justification: Plant circuitry design

K/A: 012000K6.03 3.1/3.5

Objective: RCS04C

Comments: NRC change- modify distractors A, B & D

Question:

Which one of the following groups contains reactor trips which all provide core protection from departure from nucleate boiling?

A. low shaft speed - RCP, S/G low low level, OPΔT

B. reactor coolant flow low, OTΔT, pressurizer pressure low

C. OPΔT, OTΔT, power range neutron flux high rate

D. turbine trip, safety injection actuation, reactor coolant flow low

Answer:

B. Reactor Coolant flow low, OTΔT, pressurizer pressure low

Reference: Technical Specification LSSS Bases

Justification: OPΔT trip protects fuel integrity. Turbine trip and safety injection are redundant trips. S/G level trips protect against a loss of heat sink. Power range rate trips protect against rod ejection/rod drop accidents.

K/A: 012000K5.01 3.3/2.8

Objective: RCS02C

Comments: NRC comment- be consistent with distractor nomenclature.
SRO #27

Question:

Which one of the following is an automatic closure signal for LCV-459 & 460, the letdown containment isolation valves?

A. all letdown orifice isolation valves shut

B. pressurizer level decreases below 22%

C. containment isolation phase A (CIA)

D. letdown heat exchanger outlet temperature exceeds 150°F

Answer:

B. pressurizer level decreases below 22%

Reference: Lesson Plan CHS-01-C

Justification: Plant design. All orifice isolation valves shut allows manual closure of these letdown isolation valves. Letdown HX high temperatures provide alarms, not isolation. CIA does not affect these valves.

K/A: 016000K1.01 3.4/3.4

Objective: CHS04C

Comments: NRC change- replace distractor "C"
SRO #29

Question:

The following conditions exist on Unit-3:

- reactor power is 15%
- steam dump mode select switch is in the STEAM PRESSURE position
- main turbine-generator is synchronized to the grid
- "A" turbine-driven main feed pump is in AUTO
- operators are shifting from the FRV bypass valves to the FRVs

Main steam header pressure transmitter PT-507 fails high. Which one of the following describes the plant response with no operator action?

A. steam dump valves modulate open then rapidly shut

B. "A" turbine-driven main feed pump speed decreases

C. EHC system shifts to throttle pressure limiting

D. main feed regulating bypass valves throttle open

Answer:

A. steam dump valves modulate open then rapidly shut

Reference: A71771T Pages 81-82

Justification: Steam Dumps will modulate open due to the high steam pressure signal. This results in a plant cooldown to lo-lo Tavg which shuts the steam dump valves. Automatic feed pump speed control will try to increase feed pump speed as it sees a small or no D/P between main steam header and feed header pressures. As feed pump speed increases, S/G level will increase and feed regulating valves will modulate closed to control D/P and level. PT 507 has no input to turbine EHC throttle pressure limiter circuitry.

K/A: 016000A2.01 3.0/3.1

Objective: SDS04C

Comments:

Question:

Which one of the following is the reason that trisodium phosphate (TSP) is installed in baskets on the containment floor?

A. it facilitates converting soluble iodine into insoluble iodine

B. controls sump pH to limit corrosion inside containment

C. chemically recombines hydrogen generated as a result of a LOCA

D. raises recirculation spray conductivity to entrain radioactive gases

Answer:

B. controls sump pH to limit corrosion inside containment

Reference: RF05 PDCR Manual

Justification: TSP is added to keep sump water pH basic. Keeping the water basic limits corrosion and helps put and keep soluble iodine in solution so it does not escape as a gas in the event the containment barrier is breached. The long term goal of controlling corrosion is to limit hydrogen generation. TSP has no properties which recombine hydrogen or raise conductivity.

K/A: 026000G007 3.5/3.7

Objective: CDA02C

Comments: NRC change- will allow use of the question this time because it is a plant-specific priority, but review comments were not favorable.
SRO #14

Question:

The following plant conditions exist:

- RCS Tavg 400°F
- RCS pressure 610 psia
- 2 RCPs in operation
- 20°F/hr plant heatup in progress

Which one of the following is the required status of the containment purge system at this time?

A. one containment purge isolation valve is open, all purge dampers are shut

B. all containment purge isolation valves are shut with air to their actuators isolated

C. all containment purge isolation valves are open with their keyswitches in AUTO

D. one containment purge isolation valve is shut and the air handling unit inlet dampers are shut

Answer:

B. all containment purge isolation valves are shut with air to their actuators isolated

Reference: OP 3313F

Justification: Containment purge is required to be isolated in modes 1-4 by Technical Specifications. The plant conditions listed are mode 3.

K/A: 029000A3.01 3.8/4.0

Objective: CVS08C

Comments: NRC change- modify the stem to list plant conditions

Question:

Spent fuel pool level is decreasing and requires immediate makeup. Neither the RWST nor primary grade water systems are available as a source of makeup water. Which one of the following is the next preferred source of makeup water?

A. service water

B. blended makeup from chemical and volume control system

C. fire protection water

D. condensate storage tank

Answer:

C. fire protection water

Reference: EOP 3505A Attachment A

Justification: Plant/procedure design

K/A: 033000A2.03 3.1/3.5

Objective: SFC06C

Comments: NRC change- modify distractor "B" to make more plausible.
SRO #31

Question:

A plant startup is in progress. The main turbine-generator has been synchronized to the grid. Reactor power is 15% and operators are shifting load from the steam dumps to the main turbine when the main turbine trips. Following this transient, reactor power will be _____ reactor power prior to the turbine trip and Tavg will be _____ Tavg prior to the turbine trip. (assume no operator actions)

A. less than, less than

B. equal to, equal to

C. less than, equal to

D. equal to, less than

Answer:

B. equal to, equal to

Reference: Steam Dumps Lesson Plan

Justification: Steam load lost when the turbine tripped was transferred to the steam dumps. Without any other complicating factors, reactor power and Tavg will be the same.

K/A: 039000A2.05 3.3/3.6

Objective: SDS03C

Comments:

Question:

A plant startup is in progress and the turbine is being synchronized to the grid. The synchroscope is rotating slowly in the slow direction. Which one of the following is required to be performed to address this condition prior to paralleling the main turbine?

A. generator terminal voltage lowered

B. generator terminal voltage raised

C. main turbine speed increased

D. main turbine speed decreased

Answer:

D. main turbine speed decreased

Reference: OP 3203 Rev. 11

Justification: Synchroscope rotating slow in the slow direction indicates that the incoming (turbine-generator) speed (frequency) is slightly higher than the running (grid) frequency. Turbine speed must be decreased to slightly below grid frequency (slow in the fast direction) so that the grid will "pick up" the turbine speed when the generator output breaker is shut. Terminal voltage is manually adjusted to be slightly higher than grid voltage and has no effect on the synchroscope.

K/A: 062000A4.03 2.8/2.9

Objective: GEN04C

Comments: SRO #34

Question:

Which one of the following describes the effect a loss of DC power has on an emergency diesel generator?

A. air start solenoid fails open

B. standby jacket coolant pump and heater deenergize

C. fuel rack moves to the "no fuel" position

D. the auxiliary fuel oil pump is unavailable

Answer:

D. the auxiliary fuel oil pump is unavailable

Reference: MP3 BOP Diesel Generator & Support Systems
LP EGS-01-C
LP A63763C

Justification: Plant Design

K/A: 063000K3.02 2.9/3.1

Objective: EGS07C

Comments:

Question:

Which one of the following buses, if deenergized, would result in a total loss of all Unit-3 main control board annunciators?

A. VIAC-1

B. VIAC-4

C. Battery Bus 1 (DC bus 301A-1)

D. Battery Bus 5 (DC bus 301C-1)

Answer:

D. Battery Bus 5 (DC bus 301C-1)

Reference: AOP-3563 Rev. 4

Justification: Plant design

K/A: 063000K2.01 2.9/3.1

Objective: 12507C

Comments:

Question:

Which one of the following conditions will cause a diesel generator to trip following an emergency start?

A. reverse power

B. jacket water high temperature

C. low lubricating oil pressure

D. overcurrent

Answer:

C. low lubricating oil pressure

Reference: EDG064T

Justification: Plant Design

K/A: 064000K4.02 3.9/4.2

Objective: EDG06C

Comments: SRO #35

Question:

Which one of the following describes the operation of an emergency diesel generator if the UNIT/PARALLEL switch is in the UNIT position?

A. The diesel generator will start on a loss of power but the output breaker will not automatically shut.

B. The diesel generator is aligned for isochronous operation at a constant frequency.

C. Speed and voltage droop features are inserted into the engine governor system.

D. The engine can only be started and loaded locally.

Answer:

B. The diesel generator is aligned for isochronous operation at a constant frequency.

Reference: EDG064T

Justification: Unit/Parallel switch determines the mode of EDG operation- sole source of power to the bus (UNIT) or paralleled with other sources (PARALLEL). In the UNIT position, the governor is aligned for constant speed and frequency operation to maintain ESF and ECCS equipment operating at its design levels following an emergency start. Speed and voltage droop are characteristics of parallel operation. Any emergency start signal automatically places the EDG in the UNIT mode.

K/A: 064000A4.01 4.0/4.3

Objective: EDG04C

Comments: SRO #36

Question:

Which one of the following process radiation monitors has a control function associated with its HIGH alarm?

A. RPCCW 3CCP-RE31

B. Condenser Air Ejector Discharge 3ARC-RE21

C. Auxiliary Condensate 3CNA-RE47

D. Failed Fuel Monitor 3CHS-RE69

Answer:

C. Auxiliary Condensate 3CNA-RE47

Reference: RMS073T

Justification: Plant Design

K/A: 073000K1.01 3.6/3.9

Objective: RMS04C

Comments: SRO #37

Question:

A normal plant cooldown is in progress. RHR has just been placed in service at the maximum allowable RCS temperature. Which one of the following describes the parameter(s) varied to control the RCS cooldown rate?

A. both RPCCW flowrate and RHR flowrate are manually varied through the RHR heat exchanger to obtain desired cooldown rate

B. RPCCW flowrate is varied up to 7000 gpm through the heat exchanger while RHR flowrate through the heat exchanger is kept constant

C. RHR flowrate through the heat exchanger is varied while allowing the heat exchanger bypass valve to adjust to maintain total RHR system flowrate constant

D. total RHR system flowrate is directed through the heat exchanger and is adjusted between 3000 gpm and 5000 gpm to obtained desired cooldown rate

Answer:

C. RHR flowrate through the heat exchanger is varied while allowing the heat exchanger bypass valve to adjust to maintain total RHR system flowrate constant

Reference: OP 3208 Rev. 13

Justification: System Design

K/A: 005000A1.01 3.5/3.6

Objective: RHS03C

Comments: SRO #38

Question:

Which one of the following describes the method used to adjust the RHR boron concentration prior to placing RHR in service?

A. RHR is sampled and boron is added through the sampling connection located on the RHR pump recirculation line.

B. RHR is sampled via the system drain lines and boron is added through the system vent lines.

C. RHR is recirculated through the CVCS letdown purification subsystem. Boron is added via the VCT.

D. RHR is recirculated through the RWST and RWST boron is adjusted as necessary via the normal CVCS connections.

Answer:

D. RHR is recirculated through the RWST and RWST boron is adjusted as necessary via the normal CVCS connections

Reference: OP 3310A Rev. 12

Justification: System Design

K/A: 005000K5.09 3.2/3.4

Objective: RHS03C

Comments:

Question:

Which one of the following groups contains actuation signals which **all** will result in RPCCW pumps automatically being started by the emergency generator load sequencer?

A. Safety Injection alone
Loss of Power alone
Containment Depressurization Actuation alone

B. Containment Depressurization Actuation alone
Loss of Power alone
Safety Injection coincident with Loss of Power

C. Containment Depressurization Actuation alone
Containment Depressurization Actuation coincident with Loss of Power
Safety Injection alone

D. Safety Injection alone
Safety Injection coincident with Loss of Power
Loss of Power alone

Answer:

D. Safety Injection alone
Safety Injection coincident with Loss of Power
Loss of Power alone

Reference: Text SEQ-01-T (Figure EDS 1)

Justification: System Design

K/A: 008000K4.01 3.1/3.3

Objective: EDS04C

Comment: NRC change- change distractors C & D to include multiple signals. (To do this, the question stem had to be changed from a "not" stem to a positive stem and all choices had to be re-written.)

Question:

Which one of the following is the greatest long-term (> 5 days after accident) contributor to hydrogen in containment during the entire postulated sequence of events following a design basis LOCA?

A. radiolysis in the core and containment sump

B. hydrogen formerly dissolved in the RCS coolant coming out of solution

C. corrosion of certain metals inside containment

D. zircalloy-water reaction in the core

Answer:

A. radiolysis in the core and containment sump

Reference: FSAR Figure 6.2-48

Justification: FSAR Analyses indicates this source is the greatest long-term contributor.

K/A: 028000K5.03 2.9/3.6

Objective: ACA02C

Comments: SRO #30

Question:

Which one of the following is a requirement which must be met prior to performing core alterations or moving irradiated fuel in the reactor vessel?

A. Direct communications must be maintained between the refueling station and the spent fuel pool.

B. Qualified STA present in the control room.

C. The reactor must have been subcritical for at least 100 hours.

D. Two RHR loops must be operable.

Answer:

C. The reactor must have been subcritical for at least 100 hours

Reference: Technical Specifications LCO section 3/4-9

Justification: Technical Specification requirements

K/A: 034000G010 2.7/2.9

Objective: FHS08C

Comments: NRC Change- replace original question with new question
SRO #32

Question:

Which of the following is the most severe load rejection the plant can withstand (by design) without either a reactor trip or atmospheric steam dump lifting?

A. 10% step load rejection

B. 10% per minute ramp to P-9

C. 10% per minute ramp to 50%

D. 50% step load rejection

Answer:

D. 50% step load rejection

Reference: ROD001T

Justification: Condenser steam dumps will accomodate 40% step load rejection. Automatic rod control can accomodate 10% step load rejection. Total of 50% step load rejection.

K/A: 041020K6.03 2.7/2.9

Objective: ROD01C

Comments:

Question:

Which one of the following EHC circuits acts to limit the turbine-generator speed increase during a large loss-of-load transient?

A. Speed Control Unit

B. Speed Matching Circuit

C. Throttle Pressure Limiter

D. Power/Load Unbalance

Answer:

D. Power/Load Unbalance

Reference: Lesson Plan EHC048C
Text EHC048T

Justification: Power/Load Unbalance circuit compares reheat pressure and generator current and quickly closes the control & intercept valves if there is a > 40% difference. Throttle pressure limiter circuit protects the turbine from abnormally low steam pressure. Speed Control Unit compares actual turbine speed to desired speed and produces a speed error signal to send to the low value gate. Speed Matching circuit aids in synchronizing the turbine by matching its speed with grid frequency.

K/A: 045000A3.05 2.6/2.9

Objective: EHC04C

Comments: SRO #41

Question:

Which one of the following service water valves automatically opens upon actuation of a Safety Injection signal?

A. Containment Recirculation Cooler Supply Valves (3SWP*MOV54 A-D)

B. RPCCW Heat Exchanger Supply Valves (3SWP*MOV50 A-B)

C. Diesel Generator Cooling Water Heat Exchangers Supply Isolation valves (3SWP*AOV 39 A-B)

D. TPCCW Heat Exchanger Supply Valves (3SWP*MOV71 A-B)

Answer:

C. Diesel Generator Cooling Water Heat Exchangers Supply Isolation valves

Reference: Text SWP076T

Justification: Plant/System Design

K/A: 076000K1.05 3.8/4.0

Objective: SWP03C

Comments:

Question:

Which one of the following describes the actions required by technical specifications if the Reactor Coolant System pressure exceeds 2750 psia while the unit is in Mode 3?

A. Reduce Reactor Coolant System pressure to less than 2750 psia within 15 minutes. Notify the NRC Operations Center within 15 minutes.

B. Reduce Reactor Coolant System pressure to less than 2750 psia within 5 minutes. Notify the NRC Operations Center within 1 hour.

C. Reduce Reactor Coolant System pressure to less than 2750 psia within 1 hour. Notify the NRC Operation Center within 1 hour.

D. Reduce Reactor Coolant System pressure to less than 2750 psia within 1 hour. Notify the NRC Operations Center within 15 minutes.

Answer:

B. Reduce Reactor Coolant System pressure to less than 2750 psia within 5 minutes. Notify the NRC Operations Center within 1 hour.

Reference: Technical Specification 2.1.2 and 6.7

Justification: Technical Specification 2.1.2 specifically directs RCS pressure to be reduced to less than 2750 psia within 5 minutes. Thus making 'B' the only correct choice. In addition, neither specification 2.1.2 or section 6.7 of Technical Specifications require a mode change to occur after a violation of a safety limit; thus further disqualifying choices 'C' and 'D'.

K/A: 000027K3.03 3.7/4.1

Objective: RCS08C

Comments: SRO #73

Question:

During implementation of FR-C.1 "Response to Inadequate Core Cooling", steam generator depressurization is initially stopped when S/G pressure is < 140 psig. Which one of the following is the basis for stopping the cooldown here?

A. evaluate reactor vessel pressurized thermal shock considerations

B. shut accumulator isolation valves to prevent nitrogen injection

C. evalute SI and RHR injection flow prior to injecting accumulators

D. prevent exceeding RCP operational limits

Answer:

B. shut accumulator isolation valves to prevent nitrogen injection

Reference: Westinghouse Owners Group Emergency Response Guidelines - FR-C.1 Response to Inadequate Core Cooling
Westinghouse Mitigating Core Damage - Core Cooling

Justification:
A. Not a priority during an ICC situation
B. Correct
C. Accumulators have already injected
D. Specifically not a concern in ICC

K/A: 000074K3.11 4.0/4.4

Objective: FC103C

Comments: SRO #62

Question:

The following conditions exist:

- An Anticipated Transient Without Scram (ATWS) event occurs on Unit-3 from 100% power.
- The crew is presently reducing RCS pressure via manual pressurizer PORV operation per FR-S.1 "Response to Anticipated Transient Without Trip" step 4.

Which one of the following describes why RCS pressure is reduced in this condition?

A. prevents challenging pressurizer code safety valves

B. increases charging flow to the RCS

C. decreases RCS pressure to the safety injection automatic actuation setpoint

D. increases negative reactivity added via the moderator temperature coefficient

Answer:

B. increases charging flow to increase boration flow to the RCS

Reference: FR-S.1 "Response to Anticipated Transient Without Trip"

Justification: WOG ERG basis specifically states this.

K/A: 000024K3.02 4.2/4.4

Objective: FS103C

Comments: NRC change- replace distractor "C"
SRO #47

Question:

Which one of the following correctly describes why AOP 3559 "Loss of Condenser Vacuum" places limits on how LOW turbine power can be reduced (360 Mwe) with a degraded vacuum (5"Hg)?

A. prevents damage to the low pressure turbine blading

B. ensures the steam dumps are still operable to handle the decay heat following the reactor and turbine trip

C. prevents development of large rotor bows due to excessive rubbing

D. prevents damage to the condensate pumps from cavitation due to decreased condensate subcooling

Answer:

A. prevents damage to the low pressure turbine blading.

Reference: General Electric Turbine Generator Tech Manual

Justification: Tech manual states that the turbine will be damaged from buffeting due to degraded vacuum conditions, making 'A' the correct choice. 'B' is a credible distractor but is not the stated reason for the limit. Rotor bows are a function of improper rotor rotation during shutdown conditions. The reduction of condenser vacuum (and resultant increase in condenser saturation temperature) will decrease slightly the amount of condensate depression available, but not to the point of causing condensate pump caviatation.

K/A: 000051A2.02 3.9/4.1

Objective: A5903C

Comments: SRO #55

Question:

Which one of the following correctly describes when the Reactor Coolant Pump #1 seal leakoff valve must be closed following a failure of the #1 seal?

A. when total seal leakoff flow from the affected RCP exceeds 8 gpm

B. 2 minutes after exceeding 8 gpm and just prior to tripping the affected reactor coolant pump

C. 2 minutes after the affected reactor coolant pump has been tripped

D. when #1 seal leakoff flow drops below 0.8 gpm from the affected RCP

Answer:

C. 2 minutes after the affected reactor coolant pump has been tripped.

Reference: AOP 3554 "RCP Trip or Stopping a RCP at Power"

Justification: AOP 3554 "RCP Trip or Stopping a RCP at Power" states that the seal leakoff valve should not be closed until the pump has stopped rotating.

K/A: 000015A1.07 3.5/3.4

Objective: RCS03C

Comments: NRC change- check numbers used in distractors
SRO #46

Question:

The following conditions exist:

- The unit has tripped from 100% power.
- A LOCA has occurred.
- RCS pressure has decreased to 400 psia.
- Containment temperature is 230°F.
- RCS Subcooling is 10°F.

Which one of the following is an indication of degraded core cooling while on natural circulation?

A. Core exit thermocouples indicate 750°F.

B. Reactor coolant system wide range cold leg temperatures indicate 500°F.

C. RVLMS plenum indicates 82%.

D. Reactor coolant system wide range hot leg temperatures indicate 500°F.

Answer:

A. Core exit thermocouples indicate 750°F.

Reference: Critical safety function status tree F-0-2

Justification: Procedural requirement

K/A: 000074GO12 4.3/4.4

Objective: EOU10C

Comments: SRO #63

Question:

Which one of the following correctly states the **worst** case long-term effect of a sustained loss of all A.C. power?

A. depletion of the station batteries in approximately 2 hours, resulting in a loss of accident instrumentation

B. loss of RCS inventory to containment and core uncovery due to Reactor Coolant Pump seal failure

C. depletion of the DWST resulting in loss of heat sink and resultant core cooling challenges

D. loss of natural circulation cooling due to steam bubble in the reactor vessel head

Answer:

B. loss of RCS inventory to containment and core uncovery due to Reactor Coolant Pump seal failure

Reference: WOG ERG ECA-0.0

Justification: WOG ERG for ECA-0.0 page 3 specifically states the greatest long-term threat to plant safety from a loss of all A.C.power as being the RCS inventory loss from the failed RCP seals.

K/A: 000055GO07 3.6/3.7

Objective: A0003C

Comments: NRC change- time in distractor "A" to 2 hours
SRO #56

Question:

The following conditions exist:

- FR-H.1 implemented
- Bleed & Feed core cooling established
- feed flow capability has been restored

Which one of the following describes feed flow limitations under these plant conditions?

A. If core exit thermocouple temperatures are increasing and intact S/G wide range level is > 6%, limit total feed flow to 100 gpm to one S/G.

B. If core exit thermocouple temperatures are increasing and intact S/G wide range level is < 6%, establish maximum feed flow to all intact S/G.

C. If core exit thermocouple temperatures are decreasing and intact S/G wide range level is > 6%, limit total feed flow to 100 gpm per S/G.

D. If core exit thermocouple temperatures are decreasing and intact S/G wide range level is < 6%, establish 100 gpm feed flow to one S/G.

Answer:

D. If core exit thermocouple temperatures are decreasing and intact S/G wide range level is < 6%, establish 100 gpm feed flow to one S/G.

Reference: FR-H.1 Rev. 8 Step 18

Justification: Procedural requirement

K/A: 000040K1.07 3.4/4.2

Objective: FH103C

Comments:

Question:

Which one of the following is **NOT** an entry condition for AOP 3552 "Malfunction of the Rod Drive System"?

A. ROD POSITION DEVIATION (MB4C 6-9) MB annunciator.

B. Unexplained or uncontrolled rod movement.

C. Rod M12 (control bank 'D') DRPI position indicates 168 steps, Control Bank 'D' group demand position indicates 158 steps.

D. ONE ROD BOTTOM (MB4C 5 - 10) MB annunciator.

Answer:

C. Rod M12 (control bank 'D') DRPI position indicates 168 steps, Control Bank 'D' group demand position indicates 158 steps.

Reference: AOP 3552 "Malfunction of the Rod Drive System"

Justification: Selections 'A', 'B', and 'D' are all listed as entry conditions for AOP 3552 "Malfunction of the Rod Drive System". Selection 'C' is the correct answer as the rods in control bank D are within +/-12 steps, thus entry into the AOP is not required.

K/A: 000005GO11 3.5/3.6

Objective: ROD03C

Comments:

Question:

Given the following conditions:

- The unit is in mode 1 and a turbine runback has occurred.
- Control bank 'D' rods are below the low-low rod insertion limit.
- Emergency borate valve 3CHS*MV8104 failed to open.

Which one of the following actions should be taken to initiate boration in accordance with AOP 3566 "Immediate Boration"?

A. Open one charging pump suction valve from the RWST, open at least one gravity feed boration valve, and completely open charging flow control valve.

B. Open at least one gravity feed boration valve, close at least one VCT outlet valve and limit charging flow to less than 100 gpm.

C. Open one charging pump suction valve from the RWST, close at least one VCToutlet isolation valve, and limit charging flow to 100 gpm.

D. Open at least one gravity feed boration valve and completely open the charging flow control valve.

Answer:

B. Open at least one gravity feed boration valve, close at least one VCT outlet valve and limit charging flow to less than 100 gpm.

Reference: AOP 3566 "Immediate Boration

Justification: Step 1 of AOP 3566 "Immediate Boration directs the operators to open at least one gravity feed boration valve and limit charging flow to less than 100 gpm. This makes selection 'B' the correct answer.

K/A: 000024A1.20 3.2/3.3

Objective: SRO #48

Question:

Which one of the following correctly describes the affect a Safety Injection Signal has on the RPCCW system?

A. Starts the standby RPCCW pumps.

B. Isolates RPCCW Ctmt supply and return header isolation valves 3CCP*MOV49A/B, 48A/B, 45A/B.

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

D. Isolates containment header cross-connect valves 3CCP*AOV179A&B and 3CCP*AOV180A&B.

Answer:

D. Isolates containment header cross-connect valves 3CCP*AOV179A&B and 3CCP*AOV180A&B.

Reference: RPCCW Lesson Plan, P&ID 121

Justification: The only actuation on the RPCCW system that occurs during an SIS/CIA actuation is isolation of the containment header cross-connect valves 3CCP*AOV179A&B and 3CCP*AOV180A&B.

K/A: 000026K3.02 3.6/3.9

Objective: SRO #49

Question:

Which one of the following statements is correct regarding the effects from a loss of 120 VAC Vital Instrument Bus VIAC-1? (Assume no operator action)

A. Main turbine overspeed trip protection is lost.

B. The 'A' emergency diesel generator will automatically start.

C. The 'A' train loads will not start on an ESF actuation.

D. Condenser steam dump valves fail closed.

Answer:

C. The 'A' train loads will not start on an ESF actuation.

Reference: AOP 3564 "Loss of One Protective System Channel".

Justification: A caution statement prior to step 6 of AOP 3564 "Loss of One Protective System Channel" states that the 'A' train loads will not start on an ESF actuation. This makes selection 'C' the correct answer. Distractors A & D occur on a loss of battery bus 5.

K/A: 000057A2.19 4.0/4.3

Objective: 12007C

Comments: NRC change- replace distractors A & D to avoid compromise with question #83. SRO #57

Question:

Which one of the following groups of indications are provided on the Auxiliary Shutdown Panel?

A. steam generator pressure, pressurizer level, power range nuclear instrumentation

B. pressurizer level, power range nuclear instrumentation, RCS loop hot leg temperature

C. reactor trip breaker position, RCS loop hot leg temperature, steam generator pressure

D. RCS loop hot leg temperature, steam generator pressure, pressurizer level

Answer:

D. RCS loop hot leg temperature, steam generator pressure, pressurizer level

Reference: Technical Specification 3.3.3.5

Justification: Technical Specification 3.3.3.5 Table 3.3-9 lists the instrumentation provided at the Auxiliary Shutdown Panel.

K/A: 000068K2.01 3.9/4.0

Objective: RCS06C

Comments: NRC change- modify distractor C.
SRO #60

Question:

The following conditions exist:

- A reactor trip occurred from 100% power.
- Control rods A7, D4, and D12 failed to insert upon the trip.
- The crew is performing ES-0.1 "Reactor Trip Response" step 3 "Verify all Control Rods - FULLY INSERTED".

Which one of the following describes when the immediate boration may be terminated?

A. When 200 ppm of boration has been completed.

B. When ES-0.1 has been completed.

C. Once Shutdown Margin has been verified to meet technical specification requirements.

D. When 600 ppm of boration has been completed.

Answer:

D. When 600 ppm of boration has been completed.

Reference: ES-0.1 "Reactor Trip Response"

Justification: ES-0.1 "Reactor Trip Response" step 3 directs immediate boratation 200 ppm for each stuck rod. Since three rods are not fully inserted, a total boration of 600 ppm must be completed prior to securing the immediate boration. Selection 'D' is the correct answer.

K/A: 000024K3.01 4.1/4.4

Objective: E0003C

Comments:

Question:

A safety injection has occurred on low pressurizer pressure. A review of MB1/2 ESF status reflects the following conditions:

HVQ*FN5A	Aux FW Pp Area Fan	Running
HVQ*FN5B	Aux FW Pp Area Fan	Off
HVR*FN6A	Aux Bldg FLTR Exh Fan	Running
HVR*FN6B	Aux Bldg FLTR Exh Fan	Off
CHS*MV8100	RCP Seal Return Isol	Closed
CHS*MV8511A	Charging Pp Miniflow Isol	Closed
VRS*CTV21	Gas Vent Isol	Open

Which one of the following actions would restore the proper design configuration for this plant condition?

A. start the non-running fans and close VRS*CTV21

B. start the non-running fans and open CHS*MV8100

C. close VRS*CTV21 and open CHS*MV8511A

D. open CHS*MV8100 and open CHS*MV8511A

Answer:

C. close VRS*CTV21 and open CHS*MV8511A

Reference: EOP 35 E-0 Attachment A

Justification: Procedural Requirement

K/A: 000069A2.02 3.9/4.4

Objective: E0003C

Comments:

Question:

The following conditions exist:

- The unit is operating at 100% power, steady state, all rods out.
- All control systems are in automatic.
- Rod bottom lights for rods D4 and M12 energize along with receiving the "Rod at Bottom" and "Two Rods at Bottom" alarm.
- RCS temperature, pressure, and Pressurizer level are all decreasing.

Which one of the following correctly states the required operator response in accordance with AOP 3552 "Malfunction of the Rod Drive System".

A. Notify reactor engineering and insure remaining control rods are above the rod insertion limit.

B. Verify control systems are operating to restore pressurizer level, pressure, and RCS temperature; place rod control system in manual.

C. Notify reactor engineering and place control rods in manual.

D. Trip the reactor and go to E-0 "Reactor Trip or Safety Injection".

Answer:

D. Trip the reactor and go to E-0 "Reactor Trip or Safety Injection".

Reference: AOP 3552 "Malfunction of the Rod Drive System".

Justification: The abnormal procedure for a dropped control rod is very specific in directing a reactor trip for multiple (>1) dropped rods. This invalidates all other choices, leaving 'D' as the correct answer.

K/A: 000003G007 3.4/3.6

Objective: ROD07C

Comments: SRO #42

Question:

The following situation exists:

- A reactor trip and safety injection have occurred from 100% power.
- The operators have just completed step 4 of E-0 "Reactor Trip or Safety Injection".
- The crew notes that total AFW flow cannot be increased above 300 gpm.

Which one of the following describes when the crew should transition from E-0 to FR-H.1 "Response to Loss of Secondary Heat Sink" (Assume SG levels are below minimum required).

A. Once the immediate actions of E-0 are completed.

B. When directed to transition to FR-H.1 "Response to Loss of Secondary Heat Sink" from E-0 based on SG levels/AFW flow.

C. After performing E-0 step 29 "Initiate Monitoring of Critical Safety Functions"

D. Immediately.

Answer:

B. When directed to transition to FR-H.1 "Response to Loss of Secondary Heat Sink" from E-0 based on SG levels/AFW flow.

Reference: E-0 "Reactor Trip or Safety Injection"

Justification: Referring to the rules of usage of EOPs, a transition to a Functional Restoration procedure via status trees is not to be performed until a transition out of E-0 is directed, thus making selections 'A' and 'D' invalid. Prior to reaching the diagnostic steps of E-0 (selection 'C') step 17 RNO directs a transition to FR-H.1, making selection 'B' the correct answer.

K/A: 000007G012 3.8/3.9

Objective: E0003C

Comments: SRO #65

Question:

Which one of the following correctly describes the Technical Specification leakage type and operational requirements if a Pressurizer PORV is leaking at a rate of 1 gpm to the PRT while at 100% power?

A. IDENTIFIED leakage that requires a shutdown.

B. UNIDENTIFIED leakage that requires a shutdown.

C. IDENTIFIED leakage that does not require a shutdown.

D. UNIDENTIFIED leakage that does not require a shutdown.

Answer:

C. IDENTIFIED leakage that does not require a shutdown.

Reference: Technical Specification 3.4.6.2 Operational Leakage

Justification: Technical Specifications definitions states that IDENTIFIED leakage is leakage "into closed systems, such as pump seal or valve packing leaks that are captured and conducted to a sump or collecting tank", thus making both selections 'A' and 'C' as possible correct answers. The LCO for operational leakage places a limit of 10 gpm on IDENTIFIED leakage, thus the unit can continue operating, making selection 'C' the correct answer.

K/A: 000008GO03 3.1/3.5

Objective: RCS08C

Comments: SRO #66

Question:

An inadvertant SI has occurred and E-0, Reactor Trip or Safety Injection, has been implemented. The following conditions exist:

- Pressurizer pressure 2300 psia and increasing
- One charging pump has been manually stopped
- Both SI pumps are operating

Subsequently, one PORV opens and fails to reclose. Attempts to close the PORV block valve fail. For this event, which one of the following conditions will result in the operators manually restarting the non-operating charging pump?

A. Pressurizer level decreases < 15%

B. Subcooling decreases < 32°F

C. RCS pressure decreases < 1650 psia

D. any Steam Generator narrow range level decreases < 6%

Answer:

B. Subcooling decreases < 32°F

Reference: E-0 Step 16 Caution

Justification: Distractor A is incorrect as pressurizer level will be increasing with ECCS in operation and a pressurizer steam space break. Distractor C is the RCS pressure above which charging pumps can be stopped in E-0. Distractor D is a point at which ECCS cannot be stopped, but feed flow must be maximized and monitored.

K/A: 000008K1.01 3.2/3.7

Objective: E0003C

Comments: NRC change- replace original question with another.
SRO #67

Question:

Which one of the following correctly states the basis for the RCP trip criteria?

A. Prevents RCP pump damage from overspeeding in the event of a large break LOCA.

B. Retains the RCPs operable in the event of a challenge to the core cooling status tree.

C. Limits the amount of heat added to the RCS during accident conditions.

D. Limits RCS inventory depletion and possible core uncovery during a small break LOCA.

Answer:

D. Limits RCS inventory depletion and possible core uncovery during a small break LOCA.

Reference: WOG ERG Generic Issue "RCP Trip/Restart".

Justification: WOG ERG Generic Issue "RCP Trip/Restart" specifically states the basis for the RCP trip criteria to limit RCS inventory depletion and possible core uncovery that would result from an RCP trip during a small break LOCA. Thus selection 'D' is the correct answer.

K/A: 000009K3.23 4.2/4.3

Objective: RCS05C

Comments: SRO #69

Question:

Which of the following correctly describes why the ECCS system is aligned for Hot Leg Recirculation following a LOCA?

A. Prevents corrosion in the upper head internals.

B. Minimizes damage to upper internals and upper core regions.

C. Prevents blocking of flow channels by precipitated boron on core internals.

D. Ensures the 'hard' gas bubble in the upper head is removed.

Answer:

C. Prevents blocking of flow channels by precipitated boron on core internals.

Reference: WOG ERG ES-1.4 "Transfer to Hot Leg Recirculation"

Justification: The listed reference states the hot leg recirculation is performed to flush precipitated boron from the core internals to prevent flow channel blockage. Selection 'C' is the correct answer.

K/A: 000011K3.13 3.8/4.2

Objective: ECC03C

Comments: NRC change- shorten distractors

Question:

The following situation exists:

- A large break LOCA has occurred.
- All ECCS systems are operating properly.
- The crew is performing E-1."Loss of Reactor or Secondary Coolant" step 12 to restore control power to RHR to Cold Leg Injection Valves 3SIL*MV8809A&B.

Which one of the following correctly describes why control power is re-established to RHR to Cold Leg Injection Valves 3SIL*MV8809A&B.

A. It allows the operators to open the valves for increased cold leg injection.

B. It allows the valves to automatically close to prevent pump runout.

C. It allows the operators to close the valves for hot leg recirculation.

D. It allows the operators to close the valves for cold leg recirculation.

Answer:

D. It allows the operators to close the valves for cold leg recirculation.

Reference: E-1."Loss of Reactor or Secondary Coolant"

Justification: E-1 directs the operators to re-establish control power to the RHR to Cold Leg Injection Valves 3SIL*MV8809A&B. This aligns the valves for subsequent closing by the operators while aligning the RHR system for cold leg recirculation. Selection 'D' is the correct answer.

K/A: 000011G007 3.7/3.9

Objective: ECC06C

Comments:

Question:

The following conditions exist:

- VCT is at 40% level with an automatic makeup in progress.
- VCT level transmitter LT-112 fails LOW.

Which one of the following correctly describes the immediate system response?

A. 3CHS-LCV112D & E (RWST suction valves) will open

B. 3CHS*AOV71 (Letdown divert to degasifiers) diverts flow to the VCT

C. 3RCS-LCV459 & 460 (Letdown Isolation valves) will shut

D. automatic VCT makeup will stop

Answer:

B. 3CHS*AOV71 (Letdown divert to degasifiers) fails to divert flow to the VCT

Reference: CHS004T

Justification: Level transmitter LT-112 controls automatic makeup to the VCT and AOV 71 position. A failure low would result in a continuous demand for automatic VCT makeup, thus making selection D incorrect. It takes both VCT level channels failing low to cause a suction swap over to the RWST. LT-112 has no input to the letdown isolation valves.

K/A: 000022A1.08 3.4/3.3

Objective: CHS07C

Comments: NRC change- change question from 112 failing high to 112 failing low
SRO #71

Question:

Which of the following correctly describes the reason for maintaining a hot leg vent path while operating in reduced inventory conditions in accordance with OP 3270A "Reduced Inventory Operation Mode 5"?

A. Prevents RCS pressurization from heatup which could limit RCS makeup flowrate.

B. Prevents forming a "hard" gas bubble in the upper reactor vessel head that could inhibit natural circulation cooling.

C. Prevents the formation of a "loop seal" that can cause an erroneously low RVLMS level indication.

D. Allows for feed and bleed cooling if the RHR train is lost.

Answer:

A. Prevents RCS pressurization from heatup which could limit RCS makeup flowrate.

Reference: OP 3270A "Reduced Inventory Operation Mode 5", WOG ARG on Loss of Shutdown Cooling in Reduced inventory Conditions.

Justification: WOG ARG on Loss of Shutdown Cooling in Reduced inventory Conditions states that the hot leg vent path is maintained to prevent RCS pressurization which would limit RCS makeup flow. Selection 'A' is the correct answer.

K/A: 000025K1.01 3.9/4.3

Objective: RCS03C

Comments: NRC change- reword distractor C
SRO #72

Question:

Which one of the following correctly describes the setpoints, coincidences and time delays required to actuate the AMSAC system?

A. 2/2 Turbine impulse pressures > 40% with 3/4 S/G levels < 5% for >25 seconds

B. 1/2 Turbine impulse pressures > 40% with 3/4 S/G levels < 5% for >25 seconds

C. 2/2 Turbine impulse pressures > 40% with 2/4 S/G levels < 5% for >25 seconds

D. 1/2 Turbine impulse pressures > 40% with 2/4 S/G levels < 5% for >25 seconds

Answer:

A. 2/2 Turbine impulse pressures > 40% with 3/4 S/G levels < 5% for >25 seconds

Reference: AMS-01-C Lesson plan.

Justification: System design.

K/A: 000029A1.15 4.1/3.9

Objective: AMS04C

Comments: SRO #50

Question:

Which one of the following is the basis for tripping the main turbine in the immediate action steps of FR-S.1 "Response to Nuclear Power Generation/ATWS"?

A. results in an RCS heatup which adds negative reactivity to the core

B. provides another reactor trip signal to open the reactor trip breakers

C. prevents an uncontrolled cooldown of the RCS

D. causes turbine impulse pressure to decrease to the AMSAC actuation setpoint

Answer:

C. prevents an uncontrolled cooldown of the RCS

Reference: FR-S.1 "Response to Nuclear Power Generation/ATWS" Basis Document

Justification: FR-S.1 "Response to Nuclear Power Generation/ATWS" background document specifically states the overriding concern is uncontrolled cooldown.

K/A: 000029K2.06 2.9/3.1

Objective: FS103C

Comments: SRO #52

Question:

Given the following conditions:

- Unit 3 is in Mode 6 with core alterations in progress.
- Power Range NI N-44 is out of service for calibration.
- Both Source Range channels indicate 4 cps and steady.

Which one of the following correctly describes the actions to be taken in the event that Power Range NI channel N-41 were to fail high?

A. Verify stable Source Range counts and continue with core alterations.

B. Trip affected Power Range NI channel bistables within 6 hours, continue with core alterations.

C. Continue with core alterations, repair either N-41 or N-44 prior to mode change to Mode 5.

D. Immediately suspend all operations involving core alterations or positive reactivity changes.

Answer:

D. Immediately suspend all operations involving core alterations or positive reactivity changes.

Reference: Technical Specification 3.9.2

Justification: The failing of two power range channels will enable the P-10 permissive, de-energizing high volts to the source range channels resulting in a loss of both N31 and N32 source range channels. Technical Specification 3.9.2 states that with less than two source range channels operable, you must immediately suspend all operations involving core alterations or positive reactivity changes. Thus, selection 'd' is the correct answer.

K/A: 000032GO11 3.1/3.4

Objective: NIS08C

Comments: NRC change- modify choice D
SRO #75

Question:

The following conditions exist:

- A steam generator tube rupture has occurrred.
- The operators are presently at step 4 of E-3 "Steam Generator Tube Rupture" verifying ruptured SG level greater than 6% (42% adverse) level.

Which one of the following correctly describes the basis for maintaining the ruptured SG level greater than 6% (42% adverse) level?

A. Provides filtering of the elemental iodine present in the ruptured steam generator.

B. Maintains a thermal stratification layer over the ruptured SG U-tubes, allowing for depressurization of the RCS to the ruptured SG pressure.

C. Maintains an adequate heat sink available.

D. Limits the leak rate of reactor coolant into the affected steam generator.

Answer:

B. Maintains a thermal stratification layer over the ruptured SG U-tubes, allowing for depressurization of the RCS to the ruptured SG pressure.

Reference: 000037K3.07 4.2/4.4

Justification: WOG ERG background document for E-3 "Steam Generator Tube Rupture" states the reason for a minimum level in the SG is to maintain a thermal stratification layer over the top of the SG U-tubes. This will insulate the affected SG steam bubble and prevent the affected SG from depressurizing during the RCS cooldown. This makes selection 'B' correct.

K/A: 000037K3.07 4.2/4.2

Objective: E0303C

Comments: SRO #77

Question:

Given the following:

- A steam generator tube rupture has occurred.
- The crew is performing E-3 "Steam Generator Tube Rupture" procedure.
- The crew has just completed the rapid cooldown of the RCS per step 14 of E-3.
- All ECCS equipment is operating normally.
- The operators note that RCS pressure is 1400 psia and decreasing slowly.

Which one of the following describes the actions required regarding RCP trip criteria?

A. The RCPs should be tripped immediately.

B. The RCPs should not be tripped as the RCP trip criteria apply to a small break LOCA, not a steam generator tube rupture.

C. The RCPs should not be tripped as RCP trip criteria no longer apply after step 14 for rapid cooldown of the RCS has been initiated.

D. The RCPs should be tripped when directed by step 14 of E-3 for the RCS depressurization to minimize leakage.

Answer:

C. The RCPs should not be tripped as RCP trip criteria no longer apply after step 14 for rapid cooldown of the RCS has been initiated.

Reference: E-3 "Steam Generator Tube Rupture", WOG ERG Executive Volume Generic Issue "RCP Trip/Restart"

Justification: E-3 "Steam Generator Tube Rupture" fold out page criteria specifically states that RCP trip criteria no longer apply after the RCS cooldown has commenced. This is backed up by the Generic Issue "RCP Trip/Restart" that states that the RCPs do not need to be tripped once a forced cooldown of the RCS has commenced.

K/A: 000038K3.08 4.1/4.2

Objective: E0303C

Comments:

Question:

Which one of the following correctly describes the basis for checking ruptured SG pressure greater than 420 psig in procedure E-3 "Steam Generator Tube Rupture" step 13 just prior to commencing RCS cooldown?

A. Ensures the affected steam generator U-tubes do not become uncovered in the subsequent cooldown.

B. Ensures an automatic main steamline isolation will not occur during the cooldown, thus avoiding a release due to atmospheric dump valve operations.

C. Ensures that the subsequent RCS cooldown does not result in any Pressurized Thermal Shock concerns.

D. Minimizes the RCS-to-ruptured SG pressure difference, thus minimizing the primary to secondary leak rate during the cooldown.

Answer:

C. Ensures that the subsequent RCS cooldown does not result in any Pressurized Thermal Shock concerns.

Reference: WOG ERG E-3 "Steam Generator Tube Rupture"

Justification: WOG ERG specifically states the basis for verifying ruptured SG pressure greater than 420 psig is to prevent any PTS concerns during the subsequent RCS cooldown. Thus, selection 'C' is the correct answer.

K/A: 000038A2.15 4.2/4.4

Objective: E0303C

Comments: NRC change- check setpoints
SRO #78

Question:

The following conditions exist:

- All S/G levels are decreasing.
- All Loop ΔTs are constant.
- Containment temperature is increasing.
- Steam Generator pressures initially increases slightly, then decreases.

Which one of the following correctly identifies the transient taking place?

A. Steam generator tube rupture.

B. Steam line break inside containment.

C. LOCA inside containment.

D. Feed line break inside containment.

Answer:

D. Feed line break inside containment.

Reference: UFSAR Chapter 15 "Accident Analysis" Section 15.2.8 "Feedwater System Pipe Break.

Justification: Steam Generator levels are decreasing which would rule out selection 'A' as a SGTR which would have SG levels increasing as a symptom. No mention of RCS pressure, PZR level, or containment radiation is made, nor would a LOCA cause SG pressure to initially increase, so selection 'C' can be ruled out. Since loop ΔTs are not changing then thermal power is not changing, which allows us to rule out selection 'B'. A feedline break inside containment would result in increased containment temperature, an initial increase in SG pressure due to the loss of feedwater followed by a pressure decrease due to the loss of mass from the SG. In addition, the SG mass loss would not cause a measurable increase in heat removal from the SG, resulting in loop ΔTs remaining constant.

K/A: 000054K1.01 4.1/4.3

Objective: E0303C
Comments:

Question:

The unit is at 100% power. An electrical failure has caused the following:

- Feedwater Regulating and Regulating Bypass valves shut.
- MSIV's shut.
- Thermal barrier cooling to the A and D RCPs is lost.

A loss of which of the following buses would cause these indications?

A. Loss of DC Bus 301B-1 (Battery Bus 2).

B. Loss of 120 VAC bus VIAC-1.

C. Loss of 120 VAC bus VIAC-2.

D. Loss of DC Bus 301A-1 (Battery Bus 1).

Answer:

D. Loss of DC Bus 301A-1 (Battery Bus 1).

Reference: AOP 3563 "Loss of DC Bus Power"

Justification: These indications provided are from Attachment A of the listed reference, making selection 'D' the correct answer.

K/A: 000058A2.01 3.7/4.1

Objective: 12507C

Comments: SRO #80

Question:

Given the following conditions:

- The unit is at 100% power.
- Pzr level instrument LT-459 has failed LOW.
- All actions of AOP 3571 "Instrument Failure" Attachment C are complete.

Which one of the following describes the immediate plant response to a subsequent failure of Pzr level instrument LT-460 HIGH? (Assume no operator actions are taken)

A. Pressurizer level alarm actuates.

B. Reactor Trip.

C. Charging flow decreases to minimum flow.

D. Charging flow increases to maximum flow.

Answer:

B. Reactor Trip.

Reference: AOP 3571 "Instrument Failure" Attachment C
Pzr Level and Pressure Control Lesson Plan.

Justification: With all actions of the AOP complete, the bistable associated with the high Pzr level Rx. trip has been placed in a tripped condition. With LT-460 failing high, once the channel is greater than 92% level, it's associated bistable will trip, meeting the required coincidence of 2/3 channels. Selection 'B' is the correct answer.

K/A: 000028A1.01 3.8/3.9

Objective: PPL07C

Comments: NRC change- replace distractor A

Question:

The following situation exists:

- Refueling operations are in progress in accordance with OP 3210B "Refueling Operations".
- A spent fuel assembly is in the fuel transfer car within the fuel transfer tube.
- The transfer car drive assembly has failed such that an emergency retrieval using the emergency pull out cable is required.

Which one of the following describes the retrieval location of the transfer car and what crane is utilized for this retrieval?

A. Retrieved into the containment using the Sigma refueling machine auxiliary hoist.

B. Retrieved into the containment using the polar crane.

C. Retrieved into the fuel building using the new fuel handling crane.

D. Retrieved into the fuel building using a portable hoist.

Answer:

C. Retrieved into the fuel building using the new fuel handling crane.

Reference: OP3303C "Fuel Transfer System"

Justification: Procedural requirement

K/A: 000036K2.01 2.9/3.5

Objective: FHS06C

Comments: SRO #82

Question:

Which one of the following valves would fail open on a loss of instrument air?

A. 3RCS*PCV455B Loop 1 spray valve.

B. 3CHS*FCV121 Charging Line Flow Control valve.

C. 3RHS*FCV618 'A' RHR HX Total Flow control valve.

D. 3CCP*AOV10B RPCCW Non-Safety Header Supply valve.

Answer:

B. 3CHS*FCV121 Charging Line Flow Control valve.

Reference: CVCS Lesson Plan

Justification: 3CHS*FCV121 Charging Line Flow Control valve is the only valve listed that fails to the open position upon a loss of air, making selection 'B' correct.

K/A: 000065A2.08 2.9/3.3

Objective: CHS07C

Comments:

Question:

Which one of the following symptoms would be indicative of a Reactor Coolant Pump thermal barrier leak?

A. auto closure of the RPCCW return line isolation valve for the affected RCP

B. high #1 seal leakoff flow on the affected RCP

C. low seal injection flow to the affected RCP

D. RPCCW surge tank level decreasing slowly

Answer:

A. auto closure of the RPCCW return line isolation valve for the affected RCP

Reference: P&ID 121

Justification: A leak in the RCP thermal barrier would result in high pressure reactor coolant leaking into the RPCCW system. This would be shown as increased RPCCW flow from the affected RCP as well as increased radiation in the RPCCW system. Selections 'B' and 'C' would be indicative of a seal package problem on the RCP. Selection 'D' would be indicative of a leak out of RPCCW vice an inleakage problem that would exist with a thermal barrier leak.

K/A: 000026A1.07 2.9/3.0

Objective: CCP07C

Comments:

Question:

A PEO is directed to perform an independent verification of a system valve lineup. During the verification, the PEO finds a valve closed which is required to be open. Which one of the following actions is required?

A. Document that the valve was in the wrong position on the verification sheet and continue with the verification.

B. Open the valve and notify the Shift Supervisor when the independent verification is complete.

C. Stop the independent verification and notify the Shift Supervisor.

D. Open the valve and continue the independent verification process.

Answer:

C. Stop the independent verification and notify the Shift Supervisor.

Reference: WC-6 Rev. 0 Step 1.3.8

Justification: Procedural requirement

K/A: 194001K1.01 3.6

Objective:

Comments: SRO #84

Question:

An AWO involves a clearance which requires a pneumatically-operated valve to be used as an isolation component. The valve fails open on a loss of air or power. Assuming the valve is shut, which one of the following must be done to use the valve as an isolation component?

A. nothing- pneumatic valves cannot be used as isolation components

B. tag shut the air supply to the valve operator

C. tag the power supply to the valve control solenoid

D. block the valve in position

Answer:

D. block the valve in position

Reference: WC-2 Rev.2 Attachment 2

Justification: Procedure requirements

K/A: 194001K1.02 3.7

Objective:

Comments: SRO #85

Question:

The breaker for the "B" service water pump is racked down and danger tagged. Maintenance has decided to remove the breaker assembly from the cubicle for repair. Which one of the following describes how will the danger tag on the breaker be handled?

A. After the breaker is removed from the cubicle, a danger tag will be attached to the cubicle.

B. Prior to removing the breaker, the danger tag will be moved from the breaker to the cubicle.

C. Prior to removing the breaker, a danger tag will be attached to the control room switch.

D. After removing the breaker with the danger tag still attached, an additional danger tag will be hung on the breaker cubicle.

Answer:

B. Prior to removing the breaker, the danger tag will be moved from the breaker to the cubicle.

Reference: WC-2

Justification: Procedural requirement

K/A: 194001A1.12 3.1/4.1

Objective:

Comments: NRC Change- replace distractor D
SRO #86

Question:

The HR-3 key, assigned for controlling access to the Unit 3 MIDs area, can only be given out with the permission from which of the following?

A. Shift Supervisor or Duty Officer

B. Duty Officer or Unit Director

C. HP Manager

D. RPS Supervisor

Answer:

B. Duty Officer or Unit Director

Reference: RPM 5.1.3

Justification: Procedural requirement

K/A: 194001K1.04 3.5

Objective:

Comments: SRO #87

Question:

Which one of the following people must authorize divers to enter the intake structure bay to perform maintenance?

A. Security Shift Supervisor

B. Shift Supervisor

C. Duty Officer

D. Operations Manager

Answer:

B. Shift Supervisor

Reference: ODI 3-OPS-1.05

Justification: ODI requirement

K/A: 194001K1.05 3.1/3.4

Objective: 60003C

Comments: SRO #88

Question:

Which one of the following groups contains cautions which **all** apply to local breaker operations?

A. Do not attempt to rack a breaker with the breaker closed.
Stand aside when operating breakers manually.
Control board switches must be tagged prior to local breaker operations.

B. Flash gear must be worn when manually racking down a 6900V breaker.
Breakers have no protective trips when control power is removed.
Do not leave a breaker in the TEST position if testing is not being done.

C. Control power must be available during manual breaker operation.
Manual 4160V breaker racking is used as a last resort.
Stand aside when operating breakers manually.

D. Breakers have no protective trips when control power is removed.
Control board switches must be tagged prior to local breaker operations.
Flash gear must be worn when manually racking down a 6900V breaker.

Answer:

B. Flash gear must be worn when manually racking down a 6900V breaker.
Breakers have no protective trips when control power is removed.
Do not leave a breaker in the TEST position if testing is not being done.

Reference: OP 3370A Rev. 10

Justification: Procedural requirements

K/A: 194001K1.07 3.6/3.7

Objective:

Comments: SRO #89

Question:

An emergency entry is required into a heat stress environment. Which one of the following people must approve this task?

A. Shift Supervisor

B. Duty Officer

C. Unit Director

D. Operations Manager

Answer:

A. Shift Supervisor

Reference: SH 4

Justification: Procedural requirement

K/A: 194001K1.08 3.4

Objective:

Comments: NRC change- replace distractor C
SRO #90

Question:

Which one of the following actions is required if main generator hydrogen purity cannot be maintained above 90% while the generator is in operation at 100% load?

A. Reduce generator load 20 Mwe for each 1% purity is below 90%.

B. Reduce generator load to less than 50% until purity can be maintained > 90%

C. Shutdown the main generator and purge the hydrogen with carbon dioxide.

D. Continuously monitor purity and shutdown the generator if purity drops below 50%.

Answer:

C. Shutdown the main generator and purge the hydrogen with carbon dioxide.

Reference: OP 3324C Rev. 8 Precaution 6.4

Justification: Procedural requirement

K/A: 194001K1.15 3.8

Objective:

Comments: SRO #91

Question:

Which one of the following represents the minimum Unit 3 Fire Brigade composition as required by WC-7, Fire Protection Program?

A. 1 Fire Brigade Leader from an unaffected unit and 4 Fire Brigade members from the unaffected units

B. 1 Fire Brigade Leader from an unaffected unit and 3 Fire Brigade members from the unaffected and/or affected units

C. 1 Fire Brigade Leader from the affected unit and 4 Fire Brigade members from the unaffected and/or affected units

D. 1 Fire Brigade Leader from the affected unit and 3 Fire Brigade members from the affected unit

Answer:

C. 1 Fire Brigade Leader from the affected unit and 4 Fire Brigade members from the unaffected and/or affected units

Reference: WC-7

Justification: Procedural requirement

K/A: 194001K1.16 3.5

Objective:

Comments: SRO #92

Question:

"Emergency Boration", "Refueling Operations", and "Emergency Diesel Generator A Operability Test" are all procedures which are characterized as which one of the following?

A. General Level of Use

B. Continuous Level of Use

C. Information Level of Use

D. Urgent Level of Use

Answer:

B. Continuous Level of Use procedure

Reference: DC 4

Justification: Procedural requirement

K/A: 194001A1.02 3.9

Objective:

Comments: NRC change- substitute description of procedures with plant activities in stem SRO #93

Question:

In accordance with EOP 3505 "Loss of Shutdown Cooling", which of the following actions are designed to protect personnel working inside the containment?

A. Fully open containment purge supply and exhaust dampers.

B. Perform a plant page announcement to evacuate personnel in containment.

C. Sound containment evacuation alarm and establish containment closure.

D. Have security evacuate non-essential personnel from containment.

Answer:

B. Perform a plant page announcement to evacuate personnel in containment.

Reference: EOP 3505 "Loss of Shutdown Cooling"

Justification: EOP 3505 "Loss of Shutdown Cooling" directs the operators to make a plant page announcement to evacuate personnel in containment. Selection 'B' is the correct answer.

K/A: 194001A1.04 3.0

Objective:

Comments: NRC change- replace distractor A
SRO #94

Question:

Which one of the following describes when a non-intent procedure change must be reviewed by PORC?

A. prior to implementation

B. within 7 days of implementation

C. within 14 days of implementation

D. within 30 days of implementation

Answer:

C. within 14 days of implementation

Reference: DC-1

Justification: Procedural requirement

K/A: 194001A1.13 4.3/4.1

Objective:

Comments: SRO #95

Question:

Unit-3 is operating at full power with condensate conductivity and hotwell sodium levels trending up over the last several days. Chemistry reports that the latest samples indicate condensate cation conductivity has increased to 2 micromhos above steady state levels. Which one of the following actions is required?

A. Reduce power to < 50% within 12 hours.

B. Reduce power to < 30% within 8 hours.

C. Shutdown within 12 hours.

D. Trip the reactor and close the feedwater isolation trip valves.

Answer:

D. Trip the reactor and close the feedwater isolation trip valves.

Reference: AOP 3558 Rev. 3

Justification: Procedural Requirement

K/A: 194001A1.14 2.5

Objective: 60004C

Comments: NRC change- replace original question

REACTOR OPERATOR

Candidate's Name: KEY

ANSWER SHEET

Multiple Choice: Circle or **X** your choice. If you change your answer, write your selection in the blank.

001	A	B	(C)	D	____	021	A	(B)	C	D	____
002	A	B	C	(D)	____	022	A	B	C	(D)	____
003	A	(B)	C	D	____	023	(A)	B	C	D	____
004	A	B	(C)	D	____	024	A	(B)	C	D	____
005	A	B	C	(D)	____	025	(A)	B	C	D	____
006	A	B	(C)	D	____	026	A	(B)	C	D	____
007	(A)	B	C	D	____	027	A	(B)	C	D	____
008	A	(B)	C	D	____	028	(A)	B	C	D	____
009	A	B	(C)	D	____	029	A	(B)	C	D	____
010	A	B	C	(D)	____	030	A	B	(C)	D	____
011	A	(B)	C	D	____	031	A	(B)	C	D	____
012	A	B	C	(D)	____	032	A	(B)	C	D	____
013	A	B	(C)	D	____	033	(A)	B	C	D	____
014	A	(B)	C	D	____	034	A	(B)	C	D	____
015	A	B	C	(D)	____	035	A	(B)	C	D	____
016	A	B	C	(D)	____	036	A	B	(C)	D	____
017	A	B	C	(D)	____	037	A	(B)	C	D	____
018	A	B	C	(D)	____	038	A	B	(C)	(X)	Post exam change 11/29/95
019	A	B	(C)	D	____	039	A	B	C	(D)	____
020	A	B	(C)	D	____	040	A	B	C	(D)	____

Candidate's Name: _____

ANSWER SHEET

Multiple Choice: Circle or X your choice. If you change your answer, write your selection in the blank.

041	A	B	(C)	D	____	061	A	(B)	C	D	____
042	A	(B)	C	D	____	062	A	B	C	(D)	____
043	A	(B) or	(C)	D	post exam comment 11/29/95	063	A	B	(C)	D	____
044	A	B	(C)	D	____	064	A	B	C	(D)	____
045	A	B	C	(D)	____	065	A	B	C	(D)	____
046	A	B	C	(D)	____	066	A	B	(C)	D	____
047	(A)	B	C	D	____	067	A	B	C	(D)	____
048	A	B	(C)	D	____	068	A	(B)	C	D	____
049	A	B	C	(D)	____	069	A	B	(C)	D	____
050	A	B	C	(D)	____	070	A	(B)	C	D	____
051	A	B	(C)	D	____	071	A	B	C	(D)	____
052	A	(B)	C	D	____	072	A	B	(C)	D	____
053	A	(B)	C	D	____	073	A	B	C	(D)	____
054	A	(B)	C	D	____	074	A	(B)	C	D	____
055	(A)	B	C	D	____	075	(A)	B	C	D	____
056	A	B	(C)	D	____	076	(A)	B	C	D	____
057	(A)	B	C	D	____	077	A	B	(C)	D	____
058	A	(B)	C	D	____	078	A	B	C	(D)	____
059	A	B	C	(D)	____	079	A	(B)	C	D	____
060	A	B	(C)	D	____	080	A	B	(C)	D	____

Candidate's Name: _____

ANSWER SHEET

Multiple Choice: Circle or **X** your choice. If you change your answer, write your selection in the blank.

081	A	B	(C)	D	____
082	A	B	C	(D)	____
083	A	B	C	(D)	____
084	A	(B)	C	D	____
085	A	B	(C)	D	____
086	A	(B)	C	D	____
087	(A)	B	C	D	____
088	A	B	(C)	D	____
089	A	B	C	(D)	____
090	A	(B)	C	D	____
091	A	(B)	C	D	____
092	A	(B)	C	D	____
093	A	(B)	C	D	____
094	(A)	B	C	D	____
095	A	B	(C)	D	____
096	A	B	(C)	D	____
097	A	(B)	C	D	____
098	A	(B)	C	D	____
099	A	B	(C)	D	____
100	A	B	C	(D)	____

ATTACHMENT 2

WRITTEN SRO EXAMINATION AND ANSWER KEY

Senior Reactor Operator
Licensing
Examination

Question:

A load increase is in progress in accordance with OP 3204, "At Power Operation", with the plant currently at 90% power and bank D control rods at 190 steps. Indicated axial flux difference (AFD) is approaching the positive edge of the target band. Which one of the following describes actions which should be taken to restore AFD to within its limits?

A. dilute and increase reactor power while maintaining programmed Tavg and constant rod position

B. borate and withdraw control rods while maintaining constant reactor power and Tavg

C. decrease reactor power and insert control rods while maintaining programmed Tavg and constant boron concentration

D. decrease reactor power and allow Tavg to increase above Tref while maintaining constant boron concentration and rod position

Answer:

A. dilute and increase reactor power while maintaining programmed Tavg and constant rod position

Reference: OP 3204 Rev. 11

Justification: Increasing Tavg with increasing load will cause AFD to shift negative. Rods should not be inserted unless it is absolutely necessary to drive AFD down.

K/A: 001000K5.10 3.9/4.1

Objective: NIS04C

Comments: NRC change- reword distractor D

Question:

Which one of the following is used to generate the Tref signal?

A. auctioneered high Tavg

B. auctioneered high power range NI signals

C. auctioneered high ΔT

D. turbine impulse pressure

Answer:

D. turbine impulse pressure

Reference: L.P. ROD-01-C

Justification: Plant Design

K/A: 001000K6.02 2.8/3.3

Objective: ROD02C

Comments: NRC comment- replace distractors B, C

Question:

The following situation exists:

- The unit is at 50% power with a power increase in progress.
- Rod control system is in automatic.
- Rods start to move rapidly at 72 steps per minute in the inward direction.
- Rod control is placed in manual and the control rods continue to move.

Which one of the following correctly describes the a... is required to be taken in accordance with AOP 3552 "Malfunction of the Rod Drive System"?

A. Place the control rod bank selector switch in the control bank 'D' position, reduce turbine load to minimize the Tavg-Tref deviation.

B. Trip the reactor and go to E-0 "Reactor Trip or Safety Injection".

C. Return control rod bank selector switch to automatic, reduce turbine load to minimize the Tavg-Tref deviation.

D. Place the control rod bank selector switch in the control bank 'D' position, borate as necessary to minimize the Tavg-Tref deviation.

Answer:

B. Trip the reactor and go to E-0 "Reactor Trip or Safety Injection".

Reference: AOP 3552 "Malfunction of the Rod Drive System"

Justification: Step 1 of AOP 3552 "Malfunction of the Rod Drive System" directs the operators to manually trip the reactor if rods continue to move after the rod bank selector switch is taken to MANUAL. Thus, selection 'B' is the correct answer.

K/A: 000001GO12 3.7/3.9

Objective: ROD07C

Comments: NRC change- the stem originially specified outward rod motion which made "adjusting turbine load" distractors somewhat incredible. Stem changed to inward rod motion.

Question:

Which one of the following is the reason that RCP #1 seal leakoff is isolated at RCS pressures below 125 psia?

A. Leakoff flow decreases at low pressures so #1 seal leakoff is isolated to force more flow through #2 seal.

B. Backflow from the VCT through the seal leakoff line could flush contaminants into the seals.

C. Controlled Leakage limits may be exceeded due to excessive seal injection flow at low pressures.

D. Leakoff flow instruments are not accurate at low pressures and excessive leakoff could go undetected.

Answer:

B. Backflow from the VCT through the seal leakoff line could flush contaminants into the seals.

Reference: RCP003C Reactor Coolant Pump

Justification: The VCT may be at higher pressure than the RCS and backflow through the seal leakoff line may result. This flow is not subject to filtration (no seal injection filters on this line) so VCT contaminants may be introduced into the seals and result in mechanical damage when the pump is restarted.

K/A: 003000K1.03 3.3/3.6

Objective: RCP06C

Comments : RO #3

Question:

Unit 3 is in mode 6 with loop 1 full. Maintenance work is in progress and "A" RCP is uncoupled. Which one of the following prevents leakage of reactor coolant up the RCP shaft?

A. Pump shaft lower radial bearing housing mates with the top of the thermal barrier assembly to form a low pressure coolant boundary.

B. Seal leakoff connections collect any RCS leakage and direct it back to the VCT.

C. Static head in the #3 seal standpipe is sufficient to prevent RCS leakage past the seals in this condition.

D. Nozzle dam installation prevents RCS water from entering the pump shaft area.

Answer:

A. Pump shaft lower radial bearing housing mates with the top of the thermal barrier assembly to form a low pressure coolant boundary.

Reference: RCP003T Section 2.1.3

Justification: Pump design

K/A: 003000K4.07 3.2/3.4

Objective: RCS02C

Comments:

Question:

4160V Bus 34D is de-energized due to a bus differential. Which one of the following groups contains loads which have all been lost?

A. "B" AFW Pump, 480V Load Center 32Y, "B" RPCCW Pump

B. 480V Load Center T, "B" Safety Injection Pump, "B" RHR Pump

C. "C" Charging Pump, "B" Cont. Recirc. Pump, 480V Load Center 32R

D. 480 V Load Center 32X, "D" Service Water Pump, "B" Charging Pump

Answer:

D. 480 V Load Center 32X, "D" Service Water Pump, "B" Charging Pump

Reference: CVCS System Description
Print 12179-EE-1M

Justification: Bus 34D loads include most "B" and "D" loads, "C" charging pump and 480V load centers 32U, V, W, X

K/A: 004000K2.03 3.3/3.5

Objective: CHS03C

Comments: RO #5
NRC change- change distractors away from "A" and "C" designated loads

Question:

Unit-3 is in mode 4. While conducting a routine surveillance, the CO inadvertantly resets train A low steamline pressure SI instead of train A feedwater isolation. Which one of the following describes the sequence of actions required to reset the Engineered Safety Features Actuation System/Solid State Protection System?

A. reset SI

B. shut reactor trip breakers, reset SI

C. shut reactor trip breakers, block low steam line pressure MSI, reset SI

D. block low steam line pressure MSI, shut reactor trip breakers, reset SI

Answer:

A. reset SI

Reference: Westinghouse Functional Diagrams Sheets 7 & 8

Justification: Plant design

K/A: 013000A2.06 3.7/4.0

Objective: RPS04C

Comments: NRC change- modify distractor B

Question:

A safety injection and concurrent loss of offsite power has occurred. Which one of the following groups of loads are restarted on the vital bus at EDG breaker closure?

A. safety injection pumps, auxiliary building exhaust fans

B. service water pumps, safety injection pumps

C. charging pumps, auxiliary building exhaust fans

D. charging pumps, charging & RPCCW exhaust fans

Answer:

D. charging pumps, charging & RPCCW exhaust fans

Reference: Drawing 252 Sheet 1

Justification: Plant Design

K/A: 013000A3.02 4.1/4.2

Objective: EDS04C

Comments:

Question:

Which one of the following describes the response of an intermediate range nuclear instrumentation channel following a reactor trip if it is "undercompensated" (compensating voltage set too low)?

A. indicates high - P-6 is prevented from automatically energizing the source range

B. indicates low - source range instruments are energized prematurely

C. indicates high - source range will be energized by P-6 from the other IR channel

D. indicates low - source range will not be energized until P-6 is supplied when both channels indicate low

Answer:

A. indicates high - P-6 is prevented from automatically energizing the source range

Reference: NIS015T

Justification: Plant design

K/A: 015000K4.07 3.7/3.8

Objective: NIS07C

Comments: NRC change- modify distractor D

Question:

As reactor power increases above 10% during a plant startup, which one of the following occurs?

A. power range NI low trip setpoint is automatically blocked

B. automatic reactor trip on 1 of 4 RCS loops low flow reinstated

C. manual block of intermediate range (IR) rod stop is permitted

D. high pressurizer pressure reactor trip is automatically reinstated

Answer:

C. manual block of intermediate range (IR) rod stop is permitted

Reference: NIS015T

Justification: A. This is a manual action which is permitted
B. 2/4 low flow trip is reinstated by P-7
C. Correct
D. This trip function always activated

K/A: 015000K6.01 2.9/3.2

Objective: NIS02C

Comments: RO #9

Question:

Unit-3 tripped on low-low S/G level due to a loss of all feedwater. The operating crew has just transitioned to FR-H.1 to restore S/G level and feed flow. While implementing FR-H.1 the STA notes that core exit thermocouple temperatures rise steadily into superheat conditions. Which one of the following is occurring?

A. S/G level is dropping below the top of the U-tubes

B. phase separation of the reactor coolant

C. voiding in the reactor vessel head region

D. core uncovery

Answer:

D. core uncovery

Reference: MCD Lesson Plan pages 10-11

Justification: Level dropping below the top of the S/G U-tubes and voiding in the vessel head will not result in appreciable CET temperature increases. Reflux cooling is an effective core cooling mechanism under certain specific circumstances and will not result in CET increases. Fuel uncovery will result in rapidly increasing fuel temperatures and superheat conditions as steam flow past the uncovered fuel becomes the core heat sink..

K/A: 017020A2.02 3.6/4.1

Objective: ACA07C

Comments: RO #12
NRC change- replace distractor "B"

Question:

Which one of the following automatically occurs as a result of a safety injection signal?

A. "C" CAR cooling supply shifts from chilled water to RPCCW

B. "A" CAR fan receives a "stop" signal

C. "B" CAR fan receives a "start" signal

D. "A" CAR cooling supply shifts from RPCCW to chilled water

Answer:

C. "B" CAR fan receives a "start" signal

Reference: Lesson Plan CVS-01-C
Print EM-122B

Justification: "C" CAR cooling does not have an alternate supply.
"A" CAR cooling shifts to RPCCW on a containment phase A isolation.
"A" CAR fans trip on CDA

K/A: 022000A3.01 4.1/4.3

Objective: RPS05C

Comments: RO #13

Question:

Following a large-break LOCA, RWST level decreases to 520,000 gallons. Which one of the following describes a reason for shifting to cold leg recirculation at this point?

A. there is insufficient level in the RWST for RHR pump NPSH

B. adequate level exists in the containment sump for recirculation spray pump operation

C. remaining RWST level is reserved in the event the containment sump strainers clog

D. injecting more RWST water will dilute the effectiveness of the trisodium phosphate in the containment

Answer:

B. adequate level exists in the containment sump for recirculation spray pump operation

Reference: L.P. CDA-01-C
ECC-01-C

Justification: Plant Design

K/A: 026000K4.01 4.2/4.3

Objective: ECC03C

Comments:

Question:

Which one of the following is the reason that trisodium phosphate (TSP) is installed in baskets on the containment floor?

A. it facilitates converting soluble iodine into insoluble iodine

B. controls sump pH to limit corrosion inside containment

C. chemically recombines hydrogen generated as a result of a LOCA

D. raises recirculation spray conductivity to entrain radioactive gases

Answer:

B. controls sump pH to limit corrosion inside containment

Reference: RF05 PDCR Manual

Justification: TSP is added to keep sump water pH basic. Keeping the water basic limits corrosion and helps put and keep soluble iodine in solution so it does not escape as a gas in the event the containment barrier is breached. The long term goal of controlling corrosion is to limit hydrogen generation. TSP has no properties which recombine hydrogen or raise conductivity.

K/A: 026000G007 3.5/3.7

Objective: CDA02C

Comments: NRC change- will allow use of the question this time because it is a plant-specific priority, but review comments were not favorable.
RO #34

Question:

Extraction steam non-return valves automatically shut on a turbine trip. Which one of the following is the reason for this automatic action?

A. reduces the temperature transients within the feedwater heaters

B. protects the feedwater heater shell side from overpressurization

C. reduces turbine vibration due to unbalanced loading of turbine blades during coastdown

D. prevents turbine overspeed due to steam backflow through the extraction lines

Answer:

D. prevents turbine overspeed due to steam backflow through the extraction lines

Reference: LP ESS-01-C

Justification: Steam can be trapped at significantly high pressures and volumes in the feedwater heater shells and piping after turbine stop and control valves shut. Similar to the function of the turbine intercept/intermediate stop valves, the extraction steam non-return valves shut to prevent the backflow of this volume of steam from re-entering the turbine and causing an overspeed condition.

K/A: 056000G001 2.6/2.8

Objective: ESS03C

Comments: RO #16

Question:

Which one of the following groups contains conditions which will **all** directly cause a turbine-driven main feed pump trip?

A. low suction pressure, high vibration, high feed discharge pressure

B. S/G level Hi-Hi, low steam supply pressure, high bearing oil temperature

C. Overspeed, high main feed flow, low turbine control oil pressure

D. low condenser vacuum, low pump bearing oil pressure, high thrust bearing wear

Answer:

D. low condenser vacuum, low pump bearing oil pressure, high thrust bearing wear

Reference: FWS059T

Justification: System Design- no trip signals on high feed pressure, low steam pressure, high oil temperature or high feed flow.

K/A: 059000G007 3.1/3.2

Objective: FWS04C

Comments: NRC change- original question too basic- replaced

Question:

The plant is running at 100% power. Which one of the following actions is required if ail auxiliary feed pumps are rendered inoperable?

A. restore 2 AFW pumps to operable status within 1 hour or be in hot standby within the next 6 hours

B. restore 1 AFW pump to operable status within 1 hour or be in hot standby within the following 6 hours

C. restore 1 AFW pump to operable status as soon as possible, then be in hot standby within the following 6 hours

D. restore 2 AFW pumps to operable status as soon as possible, then be in hot standby within the next 6 hours

Answer:

C. restore 1 AFW pump to operable status as soon as possible. then be in hot standby within the following 6 hours

Reference: L.C.O. 3.7.1.2 Actions c. & b.

Justification: Technical Specifications

K/A: 061000G005 3.3/4.0

Objective: FWA08C

Comments: NRC change- change distractors A & D to 6 hours (3.0.3)

Question:

Which one of the following supplies power to the vital instrument bus 1 (VIAC-1) rectifier?

A. Battery Charger 301A-1

B. Battery 301A-1

C. Bypass line 480V/120V transformer

D. 480V Bus 32-2T

Answer:

D. 480V Bus 32-2T

Reference: 125063T
120062T

Justification: Plant Design

K/A: 063000K2.01 2.9/3.1

Objective: 12003C

Comments:

Question:

Containment radiation monitors RMS41-1 & RMS42-1 automatically isolate containment purge and exhaust on high radiation levels. Which one of the following events is this designed to protect against?

A. large break LOCA

B. loss of shutdown cooling/RCS inventory

C. letdown line leak

D. fuel handling accident

Answer:

D. fuel handling accident

Reference: RMS073T Pages 6-7

Justification: Containment purge and exhaust system is only in service during modes 5-6 to provide ventilation during refueling outages and are specifically designed to isolate in the event of a dropped or damaged fuel assembly.

K/A: 072000K4.01 3.3/3.6

Objective: RMS04C

Comments: RO #22

Question:

Which one of the following groups contain radiation monitors, all of whose operability is prescribed by Technical Specifications?

A. RMS36-1 Spent Fuel Pool Area, CND07-1 Condensate Demin. Waste Neutr. Sump Effluent, SSR08-1 S/G Blowdown

B. RMS22-1 Control Room, DAS50-1 Turbine Building Floor Drains Sump Pump Discharge, CMS22-2 Containment Structure

C. RMS03-1 Incore Instrument Transfer Area, RMS36-1 Spent Fuel Pool Area, SSR08-1 S/G Blowdown

D. RMS24-1 Liquid Radwaste Area, CMS22-2 Containment Structure, DAS50-1 Turbine Building Floor Drains Sump Pump Discharge

Answer:

A. RMS36-1 Spent Fuel Pool Area, CND07-1 Condensate Demin. Waste Neutr. Sump Effluent, SSR08-1 S/G Blowdown

Reference: TS LCO 3.3.3.1 Table 3.3-6
TS LCO 3.3.3.9 Table 3.3-12

Justification: Technical Specifications

K/A: 072000A4.01 3.0/3.3

Objective: RMS08C

Comments: NRC change- requested to make this a comprehension level question, if possible. Added more distractors.

Question:

The following plant conditions exist:

- mode 5 following normal plant cooldown
- no RCPs running

Under which of the following conditions is it permissible to start a reactor coolant pump?

A. RCS temperature is 174°F with RHR cooling in service and the reactor coolant system is solid.

B. RCS temperature is 158°F and cold overpressure protection is established using both RHR suction relief valves.

C. RCS temperature is 155°F and RCP seal injection flows are 6 gpm/pump and leakoff flows are all 0.1 gpm.

D. The only operable reactor coolant pump tripped 20 minutes ago and the SCO requests you to restart it immediately.

Answer:

B. RCS temperature is 158°F and cold overpressure protection is established using both RHR suction relief valves.

Reference: OP 3201 Rev. 10 General Precautions
OP 3301D Rev. 10

Justification: Procedures do not allow starting an RCP without a bubble in the pressurizer. RCP starting duty requirements call for 30 minutes cooling time before restarting a tripped pump. Higher seal injection and leakoff flows are required.

K/A: 002000A1.05 3.4/3.7

Objective: RCS05C

Comments: RO #24

Question:

Unit 3 is in mode 2 with power at 3% and slowly increasing. Tavg is 543°F. Which one of the following actions is required?

A. Restore Tavg to $\geq$ 551°F within 15 minutes

B. Restore Tavg to $\geq$ 551°F within 30 minutes

C. log Tavg once every 30 minutes until Tavg $\geq$ 561°F

D. log Tavg once an hour until Tavg $\geq$ 561°F

Answer:

A. Restore Tavg to $\geq$ 551°F within 15 minutes

Reference: L.C.O. 3.1.1.4

Justification: Technical Specification

K/A: 002000G005 3.6/4.1

Objective: RCS08C

Comments: RO #25

Question:

Safety Injection accumulator #2 outlet check valve has developed a leak such that accumulator level needs to be lowered once per day to stay within Technical Specifications limits. Which one of the following will also occur as a result of this problem?

A. excessive thermal stresses on the accumulator outlet line

B. accumulator boron concentration decreases

C. entry into Technical Specifications Action Statement for controlled leakage

D. accumulator boron concentration increases

Answer:

B. accumulator boron concentration decreases

Reference: Technical Specification LCO 3.5.1 Surveillance Requirements

Justification: The RCS has a lower boron concentration than the 2600 - 2900 ppm conentration required by the technical specifications for the accumulators. As the RCS leaks into the accumulator, the accumulator boron concentration becomes diluted. Unless the accumulator is drained and refilled from a source of water with a higher boron concentration, it will slowly dilute down below technical specifications limits.

K/A: 006000A1.02 3.0/3.6

Objective: RCS08C

Comments: NRC change- replace distractor "C"
RO #27

Question:

Which one of the following conditions would be caused by the master pressure controller output failing to 0%?

A. pressurizer spray valves open and control heaters go to zero output

B. pressurizer PORV PCV 456 opens and pressurizer spray valves open

C. backup heaters energize and pressurizer spray valves shut

D. control bank heaters fully energize and pressurizer spray valves shut

Answer:

A. pressurizer spray valves open and control heaters go to zero output

Reference: A71771T Section 3.2

Justification: Master pressure controller output failing to zero is the same as the input signal being greater than the setpoint signal. The master pressure controller sees a high pressurizer pressure and responds to try and lower pressure by opening spray valves and de-energizing heaters. PCV 456 will not open as it is controlled by the secondary control channel.

K/A: 010000K3.01 3.8/3.9

Objective: PPL04C

Comments: NRC change- include both spray valves in distractors
RO #28

Question:

Which one of the following describes the operation of the pressurizer heater **control group**?

A. When the control switch is in AUTO after OFF, the heaters will energize in response to master pressure controller output.

B. When the control switch is in AUTO after ON, the heaters will de-energize in response to a low pressurizer level.

C. When the control switch is in AUTO after OFF, the heaters will energize in response to a pressurizer insurge.

D. When the control switch is in AUTO after ON, the heaters will energize in response to backup pressure control channel output.

Answer:

B. When the control switch is in AUTO after ON, the heaters will de-energize in response to a low pressurizer level.

Reference: PPL010T

Justification: Pzr heater control switch is ON/OFF spring return to AUTO.

A. AUTO after OFF opens the heater breaker and will not shut it until ON is selected.
B. Correct
C. AUTO after OFF opens the heater breaker and will not shut it until ON is selected.
D. Energizes in response to master pressure controller output.

K/A: 011000K6.03 2.9/3.3

Objective: PPL04C

Comments: RO #29

Question:

Which one of the following will result in a reactor trip?

A. The N-1 switches for loop 2 train A and loop 3 train B in ISOLATE position

B. Both reactor trip breakers and one reactor trip bypass breaker racked in and closed

C. Multiplexer Test switch for train A is placed in the INHIBIT position

D. SSPS mode selector switch in TEST for train B

Answer:

A. The N-1 switches for loop 2 train A and loop 3 train B in ISOLATE position

Reference: RPS012T
RPS012C

Justification: The other choices will all give SSPS General Warning alarms on one train only.

K/A: 012000A4.04 3.3/3.3

Objective: RPS04C

Comments: NRC change- requested changing distractor C to both trains, however, this will result in a trip on 2/2 general warnings.

Question:

Which one of the following groups contains reactor trips which all provide core protection from departure from nucleate boiling?

A. low shaft speed - RCP, S/G low low level, OPΔT

B. reactor coolant flow low, OTΔT, pressurizer pressure low

C. OPΔT, OTΔT, power range neutron flux high rate

D. turbine trip, safety injection actuation, reactor coolant flow low

Answer:

B. Reactor Coolant flow low, OTΔT, pressurizer pressure low

Reference: Technical Specification LSSS Bases

Justification: OPΔT trip protects fuel integrity. Turbine trip and safety injection are redundant trips. S/G level trips protect against a loss of heat sink. Power range rate trips protect against rod ejection/rod drop accidents.

K/A: 012000K5.01 3.3/3.8

Objective: RCS02C

Comments: NRC comment- be consistent with distractor nomenclature.
RO #31

Question:

Reactor power is 90% with steam generator water level control in auto. A leak develops on the variable (S/G side of level transmitter) leg of the controlling steam generator level instrument. Which one of the following will occur?

A. feed regulating valve will open

B. actual S/G level will stabilize at some level lower than the original level

C. indicated S/G level will increase on the affected channel

D. S/G level will initially increase, then return to program level

Answer:

A. feed regulating valve will open

Reference: FWS059T Sections 2.2.2.1/2.2.2.2

Justification: A leak on the variable leg will cause variable leg pressure to decrease. This will result in an increase in D/P across the transmitter. An increase in D/P is indicative of variable leg level (S/G level) decreasing and indicated level will decrease even though actual level is constant. This creates a level error signal. This level error is an input to toal error signal. Assuming flow error is zero, level error will result in an open signal sent to the feed regulating valve. As long as the leak persists, no other signal will act to mitigate this level increase until a trip on high level occurs.

K/A: 016000K3.12 3.4/3.6

Objective: FWS07C

Comments:

Question:

Which one of the following is an automatic closure signal for LCV-459 & 460, the letdown containment isolation valves?

A. all letdown orifice isolation valves shut

B. pressurizer level decreases below 22%

C. containment isolation phase A (CIA)

D. letdown heat exchanger outlet temperature exceeds 150°F

Answer:

B. pressurizer level decreases below 22%

Reference: Lesson Plan CHS-01-C

Justification: Plant design. All orifice isolation valves shut allows manual closure of these letdown isolation valves. Letdown HX high temperatures provide alarms, not isolation. CIA does not affect these valves.

K/A: 016000K1.01 3.4/3.4

Objective: CHS04C

Comments: NRC change- replace distractor "C"
RO #32

Question:

Which one of the following is the greatest long-term (> 5 days after accident) contributor to hydrogen in containment during the entire postulated sequence of events following a design basis LOCA?

A. radiolysis in the core and containment sump

B. hydrogen formerly dissolved in the RCS coolant coming out of solution

C. corrosion of certain metals inside containment

D. zircalloy-water reaction in the core

Answer:

A. radiolysis in the core and containment sump

Reference: FSAR Figure 6.2-48

Justification: FSAR Analyses indicates this source is the greatest long-term contributor.

K/A: 028000K5.03 2.9/3.6

Objective: ACA02C

Comments: RO #47

Question:

Spent fuel pool level is decreasing and requires immediate makeup. Neither the RWST nor primary grade water systems are available as a source of makeup water. Which one of the following is the next preferred source of makeup water?

A. service water

B. blended makeup from chemical and volume control system

C. fire protection water

D. condensate storage tank

Answer:

C. fire protection water

Reference: EOP 3505A Attachment A

Justification: Plant/procedure design

K/A: 033000A2.03 3.1/3.5

Objective: SFC06C

Comments: NRC change- modify distractor "B" to make more plausible.
RO #36

Question:

Which one of the following is a requirement which must be met prior to performing core alterations or moving irradiated fuel in the reactor vessel?

A. Direct communications must be maintained between the refueling station and the spent fuel pool.

B. Qualified STA present in the control room.

C. The reactor must have been subcritical for at least 100 hours.

D. Two RHR loops must be operable.

Answer:

C. The reactor must have been subcritical for at least 100 hours

Reference: Technical Specifications LCO section 3/4-9

Justification: Technical Specification requirements

K/A: 034000G010 2.7/2.9

Objective: FHS08C

Comments: NRC Change- replace original question with new question RO #48

Question:

Unit 3 is operating at 100% power when a steam break occurs in the "A" S/G steam header inside containment. Steam pressure in the "A" S/G rapidly drops to 300 psia and containment pressure rises to 18 psia. Which one of the following components, if in service at the time of the steam break, will continue to be in service?

A. letdown heat exchanger

B. "A" turbine-driven main feed pump

C. turbine-driven AFW pump

D. "A" reactor plant chiller unit

Answer:

C. turbine-driven AFW pump

Reference: RPS012T
CDS-01-C

Justification: Letdown isolates on an SI signal. Main steam is isolated on a steam break. AFW steam supply is upstream of MSIVs. Reactor plant chillers trip on low RPCCW flow after non-vital RPCCW is isolated.

K/A: 039000K3.03 3.2/3.5

Objective: RPS04C

Comments: NRC change- replace distractor A

Question:

A plant startup is in progress and the turbine is being synchronized to the grid. The synchroscope is rotating slowly in the slow direction. Which one of the following is required to be performed to address this condition prior to paralleling the main turbine?

A. generator terminal voltage lowered

B. generator terminal voltage raised

C. main turbine speed increased

D. main turbine speed decreased

Answer:

~~D. main turbine speed decreased~~ B

C. main turbine speed increased

Reference: OP 3203 Rev. 11

Justification: Synchroscope rotating slow in the slow direction indicates that the incoming (turbine-generator) speed (frequency) is slightly higher than the running (grid) frequency. Turbine speed must be decreased to slightly below grid frequency (slow in the fast direction) so that the grid will "pick up" the turbine speed when the generator output breaker is shut. Terminal voltage is manually adjusted to be slightly higher than grid voltage and has no effect on the synchroscope.

K/A: 062000A4.03 2.8/2.9

Objective: GEN04C

Comments: RO #38

Question:

Which one of the following conditions will cause a diesel generator to trip following an emergency start?

A. reverse power

B. jacket water high temperature

C. low lubricating oil pressure

D. overcurrent

Answer:

C. low lubricating oil pressure

Reference: EDG064T

Justification: Plant Design

K/A: 064000K4.02 3.9/4.2

Objective: EDG06C

Comments: RO #41

Question:

Which one of the following describes the operation of an emergency diesel generator if the UNIT/PARALLEL switch is in the UNIT position?

A. The diesel generator will start on a loss of power but the output breaker will not automatically shut.

B. The diesel generator is aligned for isochronous operation at a constant frequency.

C. Speed and voltage droop features are inserted into the engine governor system.

D. The engine can only be started and loaded locally.

Answer:

B. The diesel generator is aligned for isochronous operation at a constant frequency.

Reference: EDG064T

Justification: Unit/Parallel switch determines the mode of EDG operation- sole source of power to the bus (UNIT) or paralleled with other sources (PARALLEL). In the UNIT position, the governor is aligned for constant speed and frequency operation to maintain ESF and ECCS equipment operating at its design levels following an emergency start. Speed and voltage droop are characteristics of parallel operation. Any emergency start signal automatically places the EDG in the UNIT mode.

K/A: 064000A4.01 4.0/4.3

Objective: EDG04C

Comments: RO #42

Question:

Which one of the following process radiation monitors has a control function associated with its HIGH alarm?

A. RPCCW 3CCP-RE31

B. Condenser Air Ejector Discharge 3ARC-RE21

C. Auxiliary Condensate 3CNA-RE47

D. Failed Fuel Monitor 3CHS-RE69

Answer:

C. Auxiliary Condensate 3CNA-RE47
and
B Condenser A/E Discharge 3ARC-RE21 B

Reference: RMS073T

Justification: Plant Design

K/A: 073000K1.01 3.6/3.9

Objective: RMS04C

Comments: RO #43

Question:

A normal plant cooldown is in progress. RHR has just been placed in service at the maximum allowable RCS temperature. Which one of the following describes the parameter(s) varied to control the RCS cooldown rate?

A. both RPCCW flowrate and RHR flowrate are manually varied through the RHR heat exchanger to obtain desired cooldown rate

B. RPCCW flowrate is varied up to 7000 gpm through the heat exchanger while RHR flowrate through the heat exchanger is kept constant

C. RHR flowrate through the heat exchanger is varied while allowing the heat exchanger bypass valve to adjust to maintain total RHR system flowrate constant

D. total RHR system flowrate is directed through the heat exchanger and is adjusted between 3000 gpm and 5000 gpm to obtained desired cooldown rate

Answer:

C. RHR flowrate through the heat exchanger is varied while allowing the heat exchanger bypass valve to adjust to maintain total RHR system flowrate constant

Reference: OP 3208 Rev. 13

Justification: System Design

K/A: 005000A1.01 3.5/3.6

Objective: RHS03C

Comments: RO #44

Question:

A reactor coolant pump thermal barrier heat exchanger leak of 75 gpm develops while the plant is operating at 100% power. Which of the following is the method by which overpressurization of the RPCCW piping is prevented?

A. all thermal barrier heat exchangers automatically isolate

B. the affected thermal barrier heat exchanger automatically isolates

C. all thermal barrier heat exchangers are manually isolated

D. the affected thermal barrier heat exchanger is manually isolated

Answer:

B. the affected thermal barrier heat exchanger automatically isolates

Reference: CCP008T

Justification: Plant Design

K/A: 008000K1.04 3.3/3.3

Objective: CCP02C

Comments:

Question:

The plant is stable at 5% power when the train A steam dump interlock switch is taken to "OFF-RESET". Which one of the following describes how RCS temperature would be controlled? Assume all systems are normal and no operator actions are taken.

A. Steam dump operation would be blocked- atmospheric dump valves will lift and maintain Tc at the atmospheric dump valve setpoint.

B. Train B steam dump valves will modulate to maintain the selected setpoint.

C. When Tavg-Tref reaches the high deviation setpoint, the trip solenoid will energize and 1/2 of the dump valves will fully open.

D. Tavg will increase adding negative reactivity to the core. Power will drop to just above the point of adding heat and stabilize.

Answer:

A. Steam dump operation would be blocked- atmospheric dump valves will lift and maintain Tc at the atmospheric dump valve setpoint.

Reference: SDS041T

Justification: System Design

K/A: 041020K3.02 3.8/3.9

Objective: SDS04C

Comments:

Question:

Which one of the following EHC circuits acts to limit the turbine-generator speed increase during a large loss-of-load transient?

A. Speed Control Unit

B. Speed Matching Circuit

C. Throttle Pressure Limiter

D. Power/Load Unbalance

Answer:

D. Power/Load Unbalance

Reference: Lesson Plan EHC048C
Text EHC048T

Justification: Power/Load Unbalance circuit compares reheat pressure and generator current and quickly closes the control & intercept valves if there is a > 40% difference. Throttle pressure limiter circuit protects the turbine from abnormally low steam pressure. Speed Control Unit compares actual turbine speed to desired speed and produces a speed error signal to send to the low value gate. Speed Matching circuit aids in synchronizing the turbine by matching its speed with grid frequency.

K/A: 045000A3.05 2.6/2.9

Objective: EHC04C

Comments: RO #50

Question:

The following conditions exist:

- The unit is operating at 100% power, steady state, all rods out.
- All control systems are in automatic.
- Rod bottom lights for rods D4 and M12 energize along with receiving the "Rod at Bottom" and "Two Rods at Bottom" alarm.
- RCS temperature, pressure, and Pressurizer level are all decreasing.

Which one of the following correctly states the required operator response in accordance with AOP 3552 "Malfunction of the Rod Drive System".

A. Notify reactor engineering and insure remaining control rods are above the rod insertion limit.

B. Verify control systems are operating to restore pressurizer level, pressure, and RCS temperature; place rod control system in manual.

C. Notify reactor engineering and place control rods in manual.

D. Trip the reactor and go to E-0 "Reactor Trip or Safety Injection".

Answer:

D. Trip the reactor and go to E-0 "Reactor Trip or Safety Injection".

Reference: AOP 3552 "Malfunction of the Rod Drive System".

Justification: The abnormal procedure for a dropped control rod is very specific in directing a reactor trip for multiple (>1) dropped rods. This invalidates all other choices, leaving 'D' as the correct answer.

K/A: 000003G007 3.4/3.6

Objective: ROD07C

Comments: RO #67

Question:

Given the following conditions:

- The unit is in Mode 1 at 50% power with all loops in operation.
- Control Rod H8 in bank 'D' will not move and indicates 156 steps, all other control bank 'D' rods indicate 180 steps.
- The affected rod is trippable.

Assuming the rod cannot be repaired within the next week, which one of the following correctly describes the actions required by technical specifications for the stuck rod?

A. Verify shutdown margin requirements are met within 1 hour.

B. Verify QPTR within 1 hour and apply L.C.O. 3.2.4 if applicable.

C. Reduce thermal power level to ≤ 50% within the next hour and reduce the neutron flux high trip set point to ≤ 60% within the next four hours.

D. Be in Hot Standby within the next 6 hours.

Answer:

A. Verify shutdown margin requirements are met within 1 hour.

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

Justification: Technical specification 3.1.3.1 outlines the actions to be taken for a single stuck but trippable control rod. Selection 'D' is incorrect as it outlines the steps to be taken for an untrippable rod. Selection 'B' is incorrect as a flux map is required once, within 72 hours. Selection 'C' refers to the actions to be taken with three loops operating and is thus incorrect.

K/A: 000005GO04 2.5/3.6

Objective: ROD08C

Comments: NRC change- revise distractor B and justification.

Question:

The following conditions exist:

- A large break LOCA has occurred.
- A safety injection actuation occurred and the operators are presently in E-1 "Loss of Reactor or Secondary Coolant".
- Area radiation levels within containment have been steadily rising and are presently at 9 R/hr.
- Containment temperature peaked at approximately 220°F and is presently at 165°F.

Which one of the following correctly describes when the operators may stop utilizing 'Adverse Containment' values for this event?

A. When an evaluation of the effects of the high radiation on instrumentation within the containment is complete.

B. When containment temperature decreases to below 200°F.

C. When the effects of the accident on the instrumentation wiring has been evaluated by the I&C department.

D. With temperature in containment below 180°F, the operators may immediately stop using 'Adverse Containment' numbers.

Answer:

D. With temperature in containment below 180°F, the operators may immediately stop using 'Adverse Containment' numbers.

Reference: Millstone Unit 3 EOP E-0 "Reactor Trip or Safety Injection

Justification: As per the EOP referenced, adverse containment values use may be stopped when containment temperature returns below 180°F. Only when the use of adverse containment numbers is required due to high radiation, is an evaluation by other organizations required.

K/A: 000011A2.08 3.4/3.9

Objective: E0003C

Comments: NRC change- revise distractor B.

Question:

In accordance with E-0 "Reactor Trip or Safety Injection" the criteria for tripping reactor coolant pumps is:

A. One charging pump AND one SI pump running, RCS pressure less than 1650 psia

B. One charging pump OR one SI pump running, RCS pressure less than 1500 psia

C. One charging pump AND one SI pump running, RCS pressure less than 1500 psia

D. One charging pump OR one SI pump running, RCS pressure less than 1650 psia

Answer:

B. One charging pump OR one SI pump running, RCS pressure less than 1500 psia

Reference: E-0 "Reactor Trip or Safety Injection"

Justification: E-0 specifically states the RCP trip criteria as being one charging pump or one SI pump running, RCS pressure less than 1500 psia

K/A: 000011A1.03 4.0/4.0

Objective: E0003C

Comments:

Question:

Which one of the following correctly describes when the Reactor Coolant Pump #1 seal leakoff valve must be closed following a failure of the #1 seal?

A. when total seal leakoff flow from the affected RCP exceeds 8 gpm

B. 2 minutes after exceeding 8 gpm and just prior to tripping the affected reactor coolant pump

C. 2 minutes after the affected reactor coolant pump has been tripped

D. when #1 seal leakoff flow drops below 0.8 gpm from the affected RCP

Answer:

C. 2 minutes after the affected reactor coolant pump has been tripped.

Reference: AOP 3554 "RCP Trip or Stopping a RCP at Power"

Justification: AOP 3554 "RCP Trip or Stopping a RCP at Power" states that the seal leakoff valve should not be closed until the pump has stopped rotating.

K/A: 000015A1.07 3.5/3.4

Objective: RCS03C

Comments: NRC change- check numbers used in distractors
RO #56

Question:

The following conditions exist:

- An Anticipated Transient Without Scram (ATWS) event occurs on Unit-3 from 100% power.
- The crew is presently reducing RCS pressure via manual pressurizer PORV operation per FR-S.1 "Response to Anticipated Transient Without Trip" step 4.

Which one of the following describes why RCS pressure is reduced in this condition?

A. prevents challenging pressurizer code safety valves

B. increases charging flow to the RCS

C. decreases RCS pressure to the safety injection automatic actuation setpoint

D. increases negative reactivity added via the moderator temperature coefficient

Answer:

B. increases charging flow to increase boration flow to the RCS

Reference: FR-S.1 "Response to Anticipated Transient Without Trip"

Justification: WOG ERG basis specifically states this.

K/A: 000024K3.02 4.2/4.4

Objective: FS103C

Comments: NRC change- replace distractor "C"
RO #54

Question:

Given the following conditions:

- The unit is in mode 1 and a turbine runback has occurred.
- Control bank 'D' rods are below the low-low rod insertion limit.
- Emergency borate valve 3CHS*MV8104 failed to open.

Which one of the following actions should be taken to initiate boration in accordance with AOP 3566 "Immediate Boration"?

A. Open one charging pump suction valve from the RWST, open at least one gravity feed boration valve, and completely open charging flow control valve.

B. Open at least one gravity feed boration valve, close at least one VCT outlet valve and limit charging flow to less than 100 gpm.

C. Open one charging pump suction valve from the RWST, close at least one VCToutlet isolation valve, and limit charging flow to 100 gpm.

D. Open at least one gravity feed boration valve and completely open the charging flow control valve.

Answer:

B. Open at least one gravity feed boration valve, close at least one VCT outlet valve and limit charging flow to less than 100 gpm.

Reference: AOP 3566 "Immediate Boration

Justification: Step 1 of AOP 3566 "Immediate Boration directs the operators to open at least one gravity feed boration valve and limit charging flow to less than 100 gpm. This makes selection 'B' the correct answer.

K/A: 000024A1.20 3.2/3.3

Objective: RO #61

Question:

Which one of the following correctly describes the affect a Safety Injection Signal has on the RPCCW system?

A. Starts the standby RPCCW pumps.

B. Isolates RPCCW Ctmt supply and return header isolation valves 3CCP*MOV49A/B, 48A/B, 45A/B.

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

D. Isolates containment header cross-connect valves 3CCP*AOV179A&B and 3CCP*AOV180A&B.

Answer:

D. Isolates containment header cross-connect valves 3CCP*AOV179A&B and 3CCP*AOV180A&B.

Reference: RPCCW Lesson Plan, P&ID 121

Justification: The only actuation on the RPCCW system that occurs during an SIS/CIA actuation is isolation of the containment header cross-connect valves 3CCP*AOV179A&B and 3CCP*AOV180A&B.

K/A: 000026K3.02 3.6/3.9

Objective: RO #62

Question:

Which one of the following correctly describes the setpoints, coincidences and time delays required to actuate the AMSAC system?

A. 2/2 Turbine impulse pressures > 40% with 3/4 S/G levels < 5% for >25 seconds

B. 1/2 Turbine impulse pressures > 40% with 3/4 S/G levels < 5% for >25 seconds

C. 2/2 Turbine impulse pressures > 40% with 2/4 S/G levels < 5% for >25 seconds

D. 1/2 Turbine impulse pressures > 40% with 2/4 S/G levels < 5% for >25 seconds

Answer:

A. 2/2 Turbine impulse pressures > 40% with 3/4 S/G levels < 5% for >25 seconds

Reference: AMS-01-C Lesson plan.

Justification: System design.

K/A: 000029A1.15 4.1/3.9

Objective: AMS04C

Comments: RO #76

Question:

The following conditions exist:

- A safety injection actuation has occurred due to a plant transient.
- All four SG's are depressurizing.
- The crew is performing actions in accordance with the EOP network.

Which one of the following would require an immediate transition to a functional restoration procedure?

A. The crew identifies a valid RED path for Integrity status tree after completing the immediate action steps of E-0 "Reactor Trip or Safety Injection".

B. The crew identifies a valid ORANGE path on Subcriticality after the crew has completed E-0 and transitioned to E-2 "Faulted SG Isolation".

C. At E-0 step 11, containment pressure is verified to be 25 psia and CDA has failed to actuate.

D. The crew identifies a valid RED path for the integrity status tree while at step 16 "Verify ECCS Flow" of E-0.

Answer:

B. The crew identifies a valid ORANGE path on Subcriticality after the crew has completed E-0 and transitioned to E-2 "Faulted SG Isolation".

Reference: OP 3272 EOP User's Guide

Justification: The EOP rules of usage will not allow a transition to a functional restoration procedure via the status trees until E-0 step 29 is performed or when told to monitor the status trees by E-0. Thus unless a transition from E-0 has been made an entry into a valid functional restoration procedure cannot occur. Therefore, selection 'B' is the correct answer.

K/A: 000029GO11 4.4/4.6

Objective: E0003C

Comments: NRC change- replace distractor C

Question:

Which one of the following is the basis for tripping the main turbine in the immediate action steps of FR-S.1 "Response to Nuclear Power Generation/ATWS"?

A. results in an RCS heatup which adds negative reactivity to the core

B. provides another reactor trip signal to open the reactor trip breakers

C. prevents an uncontrolled cooldown of the RCS

D. causes turbine impulse pressure to decrease to the AMSAC actuation setpoint

Answer:

C. prevents an uncontrolled cooldown of the RCS

Reference: FR-S.1 "Response to Nuclear Power Generation/ATWS" Basis Document

Justification: FR-S.1 "Respons. to Nuclear Power Generation/ATWS" background document specifically states the overriding concern is uncontrolled cooldown.

K/A: 000029K2.06 2.9/3.1

Objective: FS103C

Comments: RO #77

Question:

Given the following situation:

- A safety injection actuation has occurred due to a faulted SG upstream of the MSIV.
- The operators have isolated the affected SG in accordance with E-2 "Faulted SG Isolation".

Which one of the following correctly describes the plant response once the affected steam generator is empty? (Assume no further operator actions)

A. Pressurizer pressure, level, and RCS hot leg temperatures stabilize.

B. Pressurizer pressure, level, and RCS hot leg temperatures continue to decrease.

C. Pressurizer pressure, level increase, RCS hot leg temperatures continue to decrease.

D. Pressurizer pressure, level, and RCS hot leg temperatures increase.

Answer:

D. Pressurizer pressure, level, and RCS hot leg temperatures increase.

Reference: WOG ERG E-2.

Justification: Once the faulted SG has blown dry, ECCS injection flow will cause an increase in pressurizer pressure and level. Decay heat will cause RCS hot leg temperatures to increase. As a result, 'd' is the correct answer.

K/A: 000040K1.03 3.8/4.2

Objective: E0203C

Comments:

Question:

The following conditions exist:

- A main steam line break inside containment has occurred.
- All MSIVs have failed open.
- The crew has just transitioned to ECA-2.1 "Uncontrolled Depressurization of All Steam Generators".
- All SG NR levels indicate off-scale low.

Which one of the following correctly describes what actions will be taken with regard to Auxiliary Feedwater Flow?

A. Continue at least 100 gpm flow to one steam generator, isolate flow to all other steam generators.

B. Throttle total AFW flow to all steam generators to 525 gpm.

C. Maintain a minimum flow of 100 gpm to each steam generator.

D. Continue at least 300 gpm flow to one steam generator, isolate flow to all other steam generators.

Answer:

C. Maintain a minimum flow of 100 gpm to each steam generator.

Reference: ECA-2.1 "Uncontrolled Depressurization of All Steam Generators".

Justification: ECA-2.1 "Uncontrolled Depressurization of All Steam Generators" specifically states that a minimum of 100 gpm AFW flow should be established to all steam generators.

K/A: 000040K1.07 3.4/4.2

Objective: FWA06C

Comments:

Question:

Which one of the following correctly describes why AOP 3559 "Loss of Condenser Vacuum" places limits on how LOW turbine power can be reduced (360 Mwe) with a degraded vacuum (5"Hg)?

A. prevents damage to the low pressure turbine blading

B. ensures the steam dumps are still operable to handle the decay heat following the reactor and turbine trip

C. prevents development of large rotor bows due to excessive rubbing

D. To prevent damage to the condensate pumps from cavitation due to decreased condensate subcooling

Answer:

A. To prevent damage to the low pressure turbine blading.

Reference: General Electric Turbine Generator Tech Manual

Justification: Tech manual states that the turbine will be damaged from buffeting due to degraded vacuum conditions, making 'A' the correct choice. 'B' is a credible distractor but is not the stated reason for the limit. Rotor bows are a function of improper rotor rotation during shutdown conditions. The reduction of condenser vacuum (and resultant increase in condenser saturation temperature) will decrease slightly the amount of condensate depression available, but not to the point of causing condensate pump caviatation.

K/A: 000051A2.02 3.9/4.1

Objective: A5903C

Comments: RO #55

Question:

Which one of the following correctly states the **worst** case long-term effect of a sustained loss of all A.C. power?

A. depletion of the station batteries in approximately 2 hours, resulting in a loss of accident instrumentation

B. loss of RCS inventory to containment and core uncovery due to Reactor Coolant Pump seal failure

C. depletion of the DWST resulting in loss of heat sink and resultant core cooling challenges

D. loss of natural circulation cooling due to steam bubble in the reactor vessel head

Answer:

B. loss of RCS inventory to containment and core uncovery due to Reactor Coolant Pump seal failure

Reference: WOG ERG ECA-0.0

Justification: WOG ERG for ECA-0.0 page 3 specifically states the greatest long-term threat to plant safety from a loss of all A.C.power as being the RCS inventory loss from the failed RCP seals.

K/A: 000055GO07 3.6/3.7

Objective: A0003C

Comments: NRC change- time in distractor "A" to 2 hours
RO #52

Question:

Which one of the following statements is corr ct regarding the effects from a loss of 120 VAC Vital Instrument Bus VIAC-1? (Assume no operator action)

A. Main turbine overspeed protection is lost.

B. Power is lost to the plant process computer (MODCOMP).

C. The 'A' train loads will not start on an ESF actuation.

D. Condenser steam dump valves fail closed.

Answer:

C. The 'A' train loads will not start on an ESF actuation.

Reference: AOP 3564 "Loss of One Protective System Channel".

Justification: A caution statement prior to step 6 of AOP 3564 "Loss of One Protective System Channel" states that the 'A' train loads will not start on an ESF actuation. This makes selection 'C' the correct answer. Distractors A & D occur on a loss of battery bus 5.

K/A: 000057A2.19 4.0/4.3

Objective: 12007C

Comments: NRC change- replace distractors A & D to avoid compromise with question #83.
RO #63

Question:

The following conditions exist:

- Process radiation monitor 3LWS*RE70, Radioactive Liquid Waste Effluent Monitor to Circulating Water System Tunnel is out of service.
- A liquid waste discharge is required from Discharge Waste Test Tank 'A'.

Which one of the following statements is correct regarding the required discharge?

A. Cannot be performed until the inoperable process monitor 3LWS*RE70 is returned to service.

B. 3LWS*RE70 can be bypassed by the shift supervisors provided dual chemistry samples, dual release rate calculations, and dual lineups are performed.

C. 3LWS*RE70 can be bypassed by the shift supervisor provided the Discharge Waste Test Tank 'A' is recirculated for an additional 175 minutes and resampled by chemistry.

D. The discharge may be conducted with 3LWS*RE70 out of service provided health physics uses portable radiation monitors to continuously monitor the discharge.

Answer:

B. 3LWS*RE70 can be bypassed by the shift supervisors provided dual chemistry samples, dual release rate calculations, and dual lineups are performed.

Reference: OP3335D "Radioactive Liquid Waste".

Justification: OP3335D "Radioactive Liquid Waste" specifically states that a liquid release may be performed with 3LWS-RE70 out of service and bypassed by the shift supervisor provided dual chemistry samples, dual release rate calculations, and dual lineups are performed.

K/A: 000059K1.05 2.6/3.6

Objective: LWS04C

Comments:

Question:

Which one of the following describes precautions associated with entering areas protected by CO_2 Fire Suppression Systems?

A. CO_2 discharge will occur 20 seconds after the warning horns sound.

B. Entering areas protected by CO_2 Fire Suppression for prolonged maintenance activities requires disabling the fire protection zone and establishing a fire watch.

C. The "Abort" position for the local CO_2 actuation switch will mechanically lock-out the actuation system.

D. Entry into an area that has had a CO_2 actuation can be done 30 minutes after the actuation without using SCBA equipment.

Answer:

B. Entering areas protected by CO_2 Fire Suppression for prolonged maintenance activities requires disabling the fire protection zone and establishing a fire watch.

Reference: OP3341C Rev. 9

Justification: Procedural Requirement

K/A: 000067K1.01 2.9/3.9

Objective: 60004C

Comments: NRC change- replace original question

Question:

Which one of the following groups of indications are provided on the Auxiliary Shutdown Panel?

A. steam generator pressure, pressurizer level, power range nuclear instrumentation

B. pressurizer level, power range nuclear instrumentation, RCS loop hot leg temperature

C. reactor trip breaker position, RCS loop hot leg temperature, steam generator pressure

D. RCS loop hot leg temperature, steam generator pressure, pressurizer level

Answer:

D. RCS loop hot leg temperature, steam generator pressure, pressurizer level

Reference: Technical Specification 3.3.3.5

Justification: Technical Specification 3.3.3.5 Table 3.3-9 lists the instrumentation provided at the Auxiliary Shutdown Panel.

K/A: 000068K2.01 3.9/4.0

Objective: RCS06C

Comments: NRC change- modify distractor C.
SRO #60

Question:

Which one of the following conditions would require Containment Closure?

A. Rx. Vessel inspection in progress with all fuel in the spent fuel pool.

B. Maintenance and reclosure of an opened containment penetration that will last 2 hours, time to core boiling is 5 hours.

C. RCS is filled, vented, and pressurized to 200 psia.

D. Maintenance and reclosure of an opened containment penetration that will last 2 hours, time to core boiling is 1 hour.

Answer:

D. Maintenance and reclosure of an opened containment penetration that will last 2 hours, time to core boiling is 1 hour.

Reference: OP 3260A "Conduct of Outages"

Justification: Step 1.4.1 of OP 3260A "Conduct of Outages" states that containment closure must be maintained with the following exceptions: 1) Individual penetrations can be closed prior to time of core boiling; 2) RCS is capable of being pressurized; and 3) All fuel is out of vessel and containment. Selections 'A', 'B', and 'C' all meet the exceptions criteria as laid out in section 1.4.1, thus selection 'D' is the correct answer.

K/A: 000069A2.02 3.9/4.4

Objective: 60004C

Comments:

Question:

During implementation of FR-C.1 "Response to Inadequate Core Cooling", steam generator depressurization is initially stopped when S/G pressure is < 140 psig. Which one of the following is the basis for stopping the cooldown here?

A. evaluate reactor vessel pressurized thermal shock considerations

B. shut accumulator isolation valves to prevent nitrogen injection

C. evalute SI and RHR injection flow prior to injecting accumulators

D. prevent exceeding RCP operational limits

Answer:

B. shut accumulator isolation valves to prevent nitrogen injection

Reference: Westinghouse Owners Group Emergency Response Guidelines - FR-C.1 Response to Inadequate Core Cooling
Westinghouse Mitigating Core Damage - Core Cooling

Justification:
A. Not a priority during an ICC situation
B. Correct
C. Accumulators have already injected
D. Specifically not a concern in ICC

K/A: 000074K3.11 4.0/4.4

Objective: FC103C

Comments: RO #53

Question:

The following conditions exist:

- The unit has tripped from 100% power.
- A LOCA has occurred.
- RCS pressure has decreased to 400 psia.
- Containment temperature is 230°F.
- RCS Subcooling is 10°F.

Which one of the following is an indication of degraded core cooling while on natural circulation?

A. Core exit thermocouples indicate 750°F.

B. Reactor coolant system wide range cold leg temperatures indicate 500°F.

C. RVLMS plenum indicates 82%.

D. Reactor coolant system wide range hot leg temperatures indicate 500°F.

Answer:

A. Core exit thermocouples indicate 750°F.

Reference: Critical safety function status tree F-0-2

Justification: Procedural requirement

K/A: 000074GO12 4.3/4.4

Objective: EOU10C

Comments: RO #57

Question:

The following conditions exist:

- A shutdown is in progress from 100% power due to RCS specific activity being outside the technical specification limits.
- To comply with technical specifications the plant must be cooled down to less than 500°F.

What is the basis for maintaining the RCS temperature less than 500°F?

A. This temperature minimizes thermal stresses on the leaking fuel pins, thus limiting the fission product release to the reactor coolant.

B. This temperature prevents the release of activity should a steam generator tube rupture, since the saturation pressure of the RCS is below the lift pressure of the atmospheric steam relief valves.

C. This temperature limits the further release of fission products to the reactor coolant system, thus limiting the projected off-site dose rates to within the limitations of 10CFR100 following a DBA LOCA.

D. This temperature is below the solubility limit for elemental iodine, thus allowing the predominant fission product to accumulate on interior RCS components.

Answer:

B. This temperature prevents the release of activity should a steam generator tube rupture, since the saturation pressure of the RCS is below the lift pressure of the atmospheric steam relief valves.

Reference: Technical Specification 3.4.8 basis.

Justification: Technical Specification 3.4.8 basis specifically states that the 500°F temperature prevents the release of activity should a steam generator tube rupture, since the saturation pressure of the RCS is below the lift pressure of the atmospheric steam relief valves.

K/A: 000076GO04 2.1/3.7

Objective: RCS08C

Question:

The following situation exists:

- A reactor trip and safety injection have occurred from 100% power.
- The operators have just completed step 4 of E-0 "Reactor Trip or Safety Injection".
- The crew notes that total AFW flow cannot be increased above 300 gpm.

Which one of the following describes when the crew should transition from E-0 to FR-H.1 "Response to Loss of Secondary Heat Sink" (Assume SG levels are below minimum required).

A. Once the immediate actions of E-0 are completed.

B. When directed to transition to FR-H.1 "Response to Loss of Secondary Heat Sink" from E-0 based on SG levels/AFW flow.

C. After performing E-0 step 29 "Initiate Monitoring of Critical Safety Functions"

D. Immediately.

Answer:

B. When directed to transition to FR-H.1 "Response to Loss of Secondary Heat Sink" from E-0 based on SG levels/AFW flow.

Reference: E-0 "Reactor Trip or Safety Injection"

Justification: Referring to the rules of usage of EOPs, a transition to a Functional Restoration procedure via status trees is not to be performed until a transition out of E-0 is directed, thus making selections 'A' and 'D' invalid. Prior to reaching the diagnostic steps of E-0 (selection 'C') step 17 RNO directs a transition to FR-H.1, making selection 'B' the correct answer.

K/A: 000007G012 3.8/3.9

Objective: E0003C

Comments: RO #68

Question:

Which one of the following correctly describes the Technical Specification leakage type and operational requirements if a Pressurizer PORV is leaking at a rate of 1 gpm to the PRT while at 100% power?

A. IDENTIFIED leakage that requires a shutdown.

B. UNIDENTIFIED leakage that requires a shutdown.

C. IDENTIFIED leakage that does not require a shutdown.

D. UNIDENTIFIED leakage that does not require a shutdown.

Answer:

C. IDENTIFIED leakage that does not require a shutdown.

Reference: Technical Specification 3.4.6.2 Operational Leakage

Justification: Technical Specifications definitions states that IDENTIFIED leakage is leakage "into closed systems, such as pump seal or valve packing leaks that are captured and conducted to a sump or collecting tank", thus making both selections 'A' and 'C' as possible correct answers. The LCO for operational leakage places a limit of 10 gpm on IDENTIFIED leakage, thus the unit can continue operating, making selection 'C' the correct answer.

K/A: 000008GO03 3.1/3.5

Objective: RCS08C

Comments: RO #69

Question:

An inadvertant SI has occurred and E-0, Reactor Trip or Safety Injection, has been implemented. The following conditions exist:

- Pressurizer pressure 2300 psia and increasing
- One charging pump has been manually stopped
- Both SI pumps are operating

Subsequently, one PORV opens and fails to reclose. Attempts to close the PORV block valve fail. For this event, which one of the following conditions will result in the operators manually restarting the non-operating charging pump?

A. Pressurizer level decreases < 15%

B. Subcooling decreases < 32°F

C. RCS pressure decreases < 1650 psia

D. any Steam Generator narrow range level decreases < 6%

Answer:

B. Subcooling decreases < 32°F

Reference: E-0 Step 16 Caution

Justification: Distractor A is incorrect as pressurizer level will be increasing with ECCS in operation and a pressurizer steam space break. Distractor C is the RCS pressure above which charging pumps can be stopped in E-0. Distractor D is a point at which ECCS cannot be stopped, but feed flow must be maximized and monitored.

K/A: 000008K1.01 3.2/3.7

Objective: E0003C

Comments: NRC change- replace original question with another.
RO #70

Question:

The following conditions exist:

- A small break LOCA has occurred.
- The crew is depresurzing the RCS in accordance with ES-1.2 "Post-LOCA Cooldown and Depressurization".
- Pressurizer level has just rapidly increased from 5% to 60%.

Which one of the following describes the cause for this rapid increase in PZR level?

A. Rx. vessel head void has formed.

B. ECCS flow is refilling the pressurizer.

C. Reference leg flashing on the pressurizer level instruments.

D. Break flow transitioned from two phase to single phase flow.

Answer:

A. Rx. vessel head void has formed.

Reference: ES-1.2 "Post-LOCA Cooldown and Depressurization".

Justification: A CAUTION just prior to the RCS depressurization step in ES-1.2 "Post-LOCA Cooldown and Depressurization" states that "Voiding may occur in the RCS during RCS depressurization. This will result in a rapidly increasing PZR level. This makes selection 'A' correct.

K/A: 000009EK3.10 3.4/3.6

Objective: RCS06C

Comments:

Question:

Which one of the following correctly states the basis for the RCP trip criteria?

A. Prevents RCP pump damage from overspeeding in the event of a large break LOCA.

B. Retains the RCPs operable in the event of a challenge to the core cooling status tree.

C. Limits the amount of heat added to the RCS during accident conditions.

D. Limits RCS inventory depletion and possible core uncovery during a small break LOCA.

Answer:

D. Limits RCS inventory depletion and possible core uncovery during a small break LOCA.

Reference: WOG ERG Generic Issue "RCP Trip/Restart".

Justification: WOG ERG Generic Issue "RCP Trip/Restart" specifically states the basis for the RCP trip criteria to limit RCS inventory depletion and possible core uncovery that would result from an RCP trip during a small break LOCA. Thus selection 'D' is the correct answer.

K/A: 000009K3.23 4.2/4.3

Objective: RCS05C

Comments: RO #71

Question:

FR-H.1 "Response to Loss of Secondary Heat Sink" step 1 directs the operators to check if the RCS pressure is greater than any non-faulted steam generator. If RCS pressure is less than non-faulted SG pressures, the operator is directed to transition to E-1 "Loss of Reactor or Secondary Coolant". Which one of the following correctly describes why FR-H.1 is not performed in this situation?

A. E-1 is the only procedure that verifies the injection alignment of ECCS systems

B. The steam generators are no longer removing heat from the RCS, thus restoration of the steam generators is not required.

C. Because a transition to ES-1.3 "Transfer to Cold Leg Recirculation" will shortly be required and can only be entered from E-1.

D. The LOCA is of a higher priority than the red path on heat sink.

Answer:

B. The steam generators are no longer removing heat from the RCS, thus restoration of the steam generators is not required.

Reference: FR-H.1 "Response to Loss of Secondary Heat Sink"

Justification: WOG ERG states the background reason for this step as being that with SG pressure greater than RCS pressure then the SGs are no longer a heat sink and are not required for decay heat removal. As a result FR-H.1 need not be performed and E-1 would be the appropriate procedure to guide plant recovery actions. Thus 'B' is the correct answer.

K/A: 000009K3.11 4.4/4.5

Objective: FH103C

Comments:

Question:

The following conditions exist:

- VCT is at 40% level with an automatic makeup in progress.
  VCT level transmitter LT-112 fails LOW.

Which one of the following correctly describes the immediate system response?

A. 3CHS-LCV112D & E (RWST suction valves) will open

B. 3CHS*AOV71 (Letdown divert to degasifiers) diverts flow to the VCT

C. 3RCS-LCV459 & 460 (Letdown Isolation valves) will shut

D. automatic VCT makeup will stop

Answer:

B. 3CHS*AOV71 (Letdown divert to degasifiers) fails to divert flow to the VCT

Reference: CHS004T

Justification: Level transmitter LT-112 controls automatic makeup to the VCT and AOV 71 position. A failure low would result in a continuous demand for automatic VCT makeup, thus making selection D incorrect. It takes both VCT level channels failing low to cause a suction swap over to the RWST. LT-112 has no input to the letdown isolation valves.

K/A: 000022A1.08 3.4/3.3

Objective: CHS07C

Comments: NRC change- change question from 112 failing high to 112 failing low
RO #74

Question:

Which of the following correctly describes the reason for maintaining a hot leg vent path while operating in reduced inventory conditions in accordance with OP 3270A "Reduced Inventory Operation Mode 5"?

A. Prevents RCS pressurization from heatup which could limit RCS makeup flowrate.

B. Prevents forming a "hard" gas bubble in the upper reactor vessel head that could inhibit natural circulation cooling.

C. Prevents the formation of a "loop seal" that can cause an erroneously low RVLMS level indication.

D. Allows for feed and bleed cooling if the RHR train is lost.

Answer:

A. Prevents RCS pressurization from heatup which could limit RCS makeup flowrate.

Reference: OP 3270A "Reduced Inventory Operation Mode 5", WOG ARG on Loss of Shutdown Cooling in Reduced inventory Conditions.

Justification: WOG ARG on Loss of Shutdown Cooling in Reduced inventory Conditions states that the hot leg vent path is maintained to prevent RCS pressurization which would limit RCS makeup flow. Selection 'A' is the correct answer.

K/A: 000025K1.01 3.9/4.3

Objective: RCS03C

Comments: NRC change- reword distractor C
RO #75

Question:

Which one of the following describes the actions required by technical specifications if the Reactor Coolant System pressure exceeds 2750 psia while the unit is in Mode 3?

A. Reduce Reactor Coolant System pressure to less than 2750 psia within 15 minutes. Notify the NRC Operations Center within 15 minutes.

B. Reduce Reactor Coolant System pressure to less than 2750 psia within 5 minutes. Notify the NRC Operations Center within 1 hour.

C. Reduce Reactor Coolant System pressure to less than 2750 psia within 1 hour. Notify the NRC Operation Center within 1 hour.

D. Reduce Reactor Coolant System pressure to less than 2750 psia within 1 hour. Notify the NRC Operations Center within 15 minutes.

Answer:

B. Reduce Reactor Coolant System pressure to less than 2750 psia within 5 minutes. Notify the NRC Operations Center within 1 hour.

Reference: Technical Specification 2.1.2 and 6.7

Justification: Technical Specification 2.1.2 specifically directs RCS pressure to be reduced to less than 2750 psia within 5 minutes. Thus making 'B' the only correct choice. In addition, neither specification 2.1.2 or section 6.7 of Technical Specifications require a mode change to occur after a violation of a safety limit; thus further disqualifying choices 'C' and 'D'.

K/A: 000027K3.03 3.7/4.1

Objective: RCS08C

Comments: RO #52

Question:

The following conditions exist:

- Normal plant cooldown is in progress in accordance with OP3208 "Plant Cooldown"
- RCS wide range cold leg temperature is 300°F.
- RCS pressure is 400 psia.

Which one of the following describes the plant response if RCS wide range pressure detector PT-403 fails HIGH?

A. PORV PCV-456 will open.

B. PORV PCV-456 will open only if PT-405 also reads high.

C. PORV PCV-456 block valve 3RCS*MV8000B will auto open.

D. PORV PCV-456 opening signal will shift to PT-405.

Answer:

A. PORV PCV-456 will open.

Reference: Pzr Lvl and Pressure Control Lesson Plan.

Justification: PORV 456, when armed in the COPS mode, will open whenever RCS pressure as sensed by PT-403 exceeds a variable setpoint determined by RCS wide range cold leg temperature. If PT-403 fails high with the PORV 456 armed the PORV will fail open, making selection 'A' correct.

K/A: 000027A2.18 3.4/3.5

Objective: PPL07C

Comments: NRC change- distractors B & D

Question:

Given the following conditions:

- Unit 3 is in Mode 6 with core alterations in progress.
- Power Range NI N-44 is out of service for calibration.
- Both Source Range channels indicate 4 cps and steady.

Which one of the following correctly describes the actions to be taken in the event that Power Range NI channel N-41 were to fail high?

A. Verify stable Source Range counts and continue with core alterations.

B. Trip affected Power Range NI channel bistables within 6 hours, continue with core alterations.

C. Continue with core alterations, repair either N-41 or N-44 prior to mode change to Mode 5.

D. Immediately suspend all operations involving core alterations or positive reactivity changes.

Answer:

D. Immediately suspend all operations involving core alterations or positive reactivity changes.

Reference: Technical Specification 3.9.2

Justification: The failing of two power range channels will enable the P-10 permissive, de-energizing high volts to the source range channels resulting in a loss of both N31 and N32 source range channels. Technical Specification 3.9.2 states that with less than two source range channels operable, you must immediately suspend all operations involving core alterations or positive reactivity changes. Thus, selection 'd' is the correct answer.

K/A: 000032GO11 3.1/3.4

Objective: NIS08C

Comments: NRC change- modify choice D
RO #78

Question:

Given the following conditions:

- plant startup in progress
- reactor power is 7% and steady
- intermediate range (IR) channel N35 fails as indicated by "IR Loss of Detector Voltage" and "IR Loss of Compensation Voltage" annunciators alarming

Which one of the following actions are required?

A. Reduce power to below P-6 until the IR channel operability is restored.

B. Reduce power to below 5% until the IR channel operability is restored.

C. Restore IR channel operability prior to exceeding 10% power.

D. Place the level trip bypass switch for intermediate range channel N35 to BYPASS position.

Answer:

C. Restore IR channel operability prior to exceeding 10% power.

Reference: AOP 3571 Rev. 3 Attachment E

Justification: Procedural requirement

K/A: 000033K3.02 3.6/3.9

Objective: NIS07C

Comments:

Question:

The following conditions exist:

- A steam generator tube rupture has occurrred.
- The operators are presently at step 4 of E-3 "Steam Generator Tube Rupture" verifying ruptured SG level greater than 6% (42% adverse) level.

Which one of the following correctly describes the basis for maintaining the ruptured SG level greater than 6% (42% adverse) level?

A. Provides filtering of the elemental iodine present in the ruptured steam generator.

B. Maintains a thermal stratification layer over the ruptured SG U-tubes, allowing for depressurization of the RCS to the ruptured SG pressure.

C. Maintains an adequate heat sink available.

D. Limits the leak rate of reactor coolant into the affected steam generator.

Answer:

B. Maintains a thermal stratification layer over the ruptured SG U-tubes, allowing for depressurization of the RCS to the ruptured SG pressure.

Reference: 000037K3.07 4.2/4.4

Justification: WOG ERG background document for E-3 "Steam Generator Tube Rupture" states the reason for a minimum level in the SG is to maintain a thermal stratification layer over the top of the SG U-tubes. This will insulate the affected SG steam bubble and prevent the affected SG from depressurizing during the RCS cooldown. This makes selection 'B' correct.

K/A: 000037K3.07 4.2/4.2

Objective: E0303C

Comments: RO #79

Question:

Which one of the following correctly describes the basis for checking ruptured SG pressure greater than 420 psig in procedure E-3 "Steam Generator Tube Rupture" step 13 just prior to commencing RCS cooldown?

A. Ensures the affected steam generator U-tubes do not become uncovered in the subsequent cooldown.

B. Ensures an automatic main steamline isolation will not occur during the cooldown, thus avoiding a release due to atmospheric dump valve operations.

C. Ensures that the subsequent RCS cooldown does not result in any Pressurized Thermal Shock concerns.

D. Minimizes the RCS-to-ruptured SG pressure difference, thus minimizing the primary to secondary leak rate during the cooldown.

Answer:

C. Ensures that the subsequent RCS cooldown does not result in any Pressurized Thermal Shock concerns.

Reference: WOG ERG E-3 "Steam Generator Tube Rupture"

Justification: WOG ERG specifically states the basis for verifying ruptured SG pressure greater than 420 psig is to prevent any PTS concerns during the subsequent RCS cooldown. Thus, selection 'C' is the correct answer.

K/A: 000038A2.15 4.2/4.4

Objective: E0303C

Comments: NRC change- check setpoints
RO #81

Question:

Which one of the following groups contains instruments, which if failed, each would directly impact main feedwater pump speed control?

A. main steam header pressure, turbine impulse pressure, feed pressure

B. feed flow, power range nuclear instrument, S/G pressure

C. steam flow, S/G pressure, main steam header pressure

D. narrow range S/G level, feed flow, steam flow

Answer:

C. steam flow, S/G pressure, main steam header pressure

Reference: AOP 3571 Rev. 3

Justification: System design

K/A: 000054GO11 3.4/3.3

Objective: FWS07C

Comments:

Question:

The unit is at 100% power. An electrical failure has caused the following:

- Feedwater Regulating and Regulating Bypass valves shut.
- MSIV's shut.
- Thermal barrier cooling to the A and D RCPs is lost.

A loss of which of the following buses would cause these indications?

A. Loss of DC Bus 301B-1 (Battery Bus 2).

B. Loss of 120 VAC bus VIAC-1.

C. Loss of 120 VAC bus VIAC-2.

D. Loss of DC Bus 301A-1 (Battery Bus 1).

Answer:

D. Loss of DC Bus 301A-1 (Battery Bus 1).

Reference: AOP 3563 "Loss of DC Bus Power"

Justification: These indications provided are from Attachment A of the listed reference, making selection 'D' the correct answer.

K/A: 000058A2.01 3.7/4.1

Objective: 12507C

Comments: RO #83

Question:

Which one of the following groups of pressurizer parameters generate an RPS or ESFAS signal?

A. high pressurizer level, low pressurizer level, low pressurizer pressure

B. high pressurizer level, high pressurizer pressure, low pressurizer level

C. high pressurizer level, low pressurizer pressure, high pressurizer pressure

D. low pressurizer level, low pressurizer pressure, high pressurizer pressure

Answer:

C. high pressurizer level, low pressurizer pressure, high pressurizer pressure

Reference: Technical Specifications LCO 3.3.1, & LCO 3.3.2

Justification: Plant Design

K/A: 000028A1.01 3.8/3.9

Objective: RPS04C

Comments:

Question:

The following situation exists:

- Refueling operations are in progress in accordance with OP 3210B "Refueling Operations".
- A spent fuel assembly is in the fuel transfer car within the fuel transfer tube.
- The transfer car drive assembly has failed such that an emergency retrieval using the emergency pull out cable is required.

Which one of the following describes the retrieval location of the transfer car and what crane is utilized for this retrieval?

A. Retrieved into the containment using the Sigma refueling machine auxiliary hoist.

B. Retrieved into the containment using the polar crane.

C. Retrieved into the fuel building using the new fuel handling crane.

D. Retrieved into the fuel building using a portable hoist.

Answer:

C. Retrieved into the fuel building using the new fuel handling crane.

Reference: OP3303C "Fuel Transfer System"

Justification: Procedural requirement

K/A: 000036K2.01 2.9/3.5

Objective: FHS06C

Comments: RO #85

Question:

Assuming an SI did **NOT** occur, which one of the following correctly describes the response of the emergency diesel generator system to loss of offsite power from 100% power?

A. Buses 34(C) and 34(D) are stripped, normal loads resequence on after the diesel output breaker closes, the safety injection and RHR pumps are not started.

B. Buses 34(C) and 34(D) are stripped, emergency loads including the safety injection and RHR pumps are started after the diesel output breaker closes.

C. Buses 34(C) and 34(D) are stripped, all previously running loads with the exception of the safety injection and RHR pumps restart when the diesel output breaker closes.

D. Buses 34(C) and 34(D) are stripped, all previously running loads including the safety injection and RHR pumps restart when the diesel output breaker closes.

Answer:

A. Buses 34(C) and 34(D) are stripped, normal loads resequence on after the diesel output breaker closes, the safety injection and RHR pumps are not started.

Reference: EDG Sequencer Lesson Plan

Justification: With a loss of offsite power without any other safeguards signals present, the vital buses will be stripped and the diesel output breaker will close in when the EDG is at running speed and voltage. The sequencer will then sequence on loads with the exception of the SI and RHR pumps that would start if a SIS was present. Thus selection 'A' is the correct answer.

K/A: 000056A2.47 3.8/3.9

Objective: EGS07C

Comments: NRC change- distractors C & D

Question:

A PEO is directed to perform an independent verification of a system valve lineup. During the verification, the PEO finds a valve closed which is required to be open. Which one of the following actions is required?

A. Document that the valve was in the wrong position on the verification sheet and continue with the verification.

B. Open the valve and notify the Shift Supervisor when the independent verification is complete.

C. Stop the independent verification and notify the Shift Supervisor.

D. Open the valve and continue the independent verification process.

Answer:

C. Stop the independent verification and notify the Shift Supervisor.

Reference: WC-6 Rev. 0 Step 1.3.8

Justification: Procedural requirement

K/A: 194001K1.01 3.6

Objective:

Comments: RO #88

Question:

An AWO involves a clearance which requires a pneumatically-operated valve to be used as an isolation component. The valve fails open on a loss of air or power. Assuming the valve is shut, which one of the following must be done to use the valve as an isolation component?

A. nothing- pneumatic valves cannot be used as isolation components

B. tag shut the air supply to the valve operator

C. tag the power supply to the valve control solenoid

D. block the valve in position

Answer:

D. block the valve in position

Reference: WC-2 Rev.2 Attachment 2

Justification: Procedure requirements

K/A: 194001K1.02 3.7

Objective:

Comments: RO #89

Question:

The breaker for the "B" service water pump is racked down and danger tagged. Maintenance has decided to remove the breaker assembly from the cubicle for repair. Which one of the following describes how will the danger tag on the breaker be handled?

A. After the breaker is removed from the cubicle, a danger tag will be attached to the cubicle.

B. Prior to removing the breaker, the danger tag will be moved from the breaker to the cubicle.

C. Prior to removing the breaker, a danger tag will be attached to the control room switch.

D. After removing the breaker with the danger tag still attached, an additional danger tag will be hung on the breaker cubicle.

Answer:

B. Prior to removing the breaker, the danger tag will be moved from the breaker to the cubicle.

Reference: WC-2

Justification: Procedural requirement

K/A: 194001A1.12 3.1/4.1

Objective:

Comments: NRC Change- replace distractor D
RO #90

Question:

The HR-3 key, assigned for controlling access to the Unit 3 MIDs area, can only be given out with the permission from which of the following?

A. Shift Supervisor or Duty Officer

B. Duty Officer or Unit Director

C. HP Manager

D. RPS Supervisor

Answer:

B. Duty Officer or Unit Director

Reference: RPM 5.1.3

Justification: Procedural requirement

K/A: 194001K1.04 3.5

Objective:

Comments: RO #91

Question:

Which one of the following people must authorize divers to enter the intake structure bay to perform maintenance?

A. Security Shift Supervisor

B. Shift Supervisor

C. Duty Officer

D. Operations Manager

Answer:

B. Shift Supervisor

Reference: ODI 3-OPS-1.05

Justification: ODI requirement

K/A: 194001K1.05 3.1/3.4

Objective: 60003C

Comments: RO #92

Question:

Which one of the following groups contains cautions which **all** apply to local breaker operations?

A. Do not attempt to rack a breaker with the breaker closed.
Stand aside when operating breakers manually.
Control board switches must be tagged prior to local breaker operations.

B. Flash gear must be worn when manually racking down a 6900V breaker.
Breakers have no protective trips when control power is removed.
Do not leave a breaker in the TEST position if testing is not being done.

C. Control power must be available during manual breaker operation.
Manual 4160V breaker racking is used as a last resort.
Stand aside when operating breakers manually.

D. Breakers have no protective trips when control power is removed.
Control board switches must be tagged prior to local breaker operations.
Flash gear must be worn when manually racking down a 6900V breaker.

Answer:

B. Flash gear must be worn when manually racking down a 6900V breaker.
Breakers have no protective trips when control power is removed.
Do not leave a breaker in the TEST position if testing is not being done.

Reference: OP 3370A Rev. 10

Justification: Procedural requirements

K/A: 194001K1.07 3.6/3.7

Objective:

Comments: RO #93

Question:

An emergency entry is required into a heat stress environment. Which one of the following people must approve this task?

A. Shift Supervisor

B. Duty Officer

C. Unit Director

D. Operations Manager

Answer:

A. Shift Supervisor

Reference: SH 4

Justification: Procedural requirement

K/A: 194001K1.08 3.4

Objective:

Comments: NRC change- replace distractor C
RO #94

Question:

Which one of the following actions is required if main generator hydrogen purity cannot be maintained above 90% while the generator is in operation at 100% load?

A. Reduce generator load 20 Mwe for each 1% purity is below 90%.

B. Reduce generator load to less than 50% until purity can be maintained > 90%

C. Shutdown the main generator and purge the hydrogen with carbon dioxide.

D. Continuously monitor purity and shutdown the generator if purity drops below 50%.

Answer:

C. Shutdown the main generator and purge the hydrogen with carbon dioxide.

Reference: OP 3324C Rev. 8 Precaution 6.4

Justification: Procedural requirement

K/A: 194001K1.15 3.8

Objective:

Comments: RO #95

Question:

Which one of the following represents the minimum Unit 3 Fire Brigade composition as required by WC-7, Fire Protection Program?

A. 1 Fire Brigade Leader from an unaffected unit and 4 Fire Brigade members from the unaffected units

B. 1 Fire Brigade Leader from an unaffected unit and 3 Fire Brigade members from the unaffected and/or affected units

C. 1 Fire Brigade Leader from the affected unit and 4 Fire Brigade members from the unaffected and/or affected units

D. 1 Fire Brigade Leader from the affected unit and 3 Fire Brigade members from the affected unit

Answer:

C. 1 Fire Brigade Leader from the affected unit and 4 Fire Brigade members from the unaffected and/or affected units

Reference: WC-7

Justification: Procedural requirement

K/A: 194001K1.16 3.5

Objective:

Comments: RO #96

Question:

"Emergency Boration", "Refueling Operations", and "Emergency Diesel Generator A Operability Test" are all procedures which are characterized as which one of the following?

A. General Level of Use

B. Continuous Level of Use

C. Information Level of Use

D. Urgent Level of Use

Answer:

B. Continuous Level of Use procedure

Reference: DC 4

Justification: Procedural requirement

K/A: 194001A1.02 3.9

Objective:

Comments: NRC change- substitute description of procedures with plant activities in stem RO #97

Question:

In accordance with EOP 3505 "Loss of Shutdown Cooling", which of the following actions are designed to protect personnel working inside the containment?

A. Fully open containment purge supply and exhaust dampers.

B. Perform a plant page announcement to evacuate personnel in containment.

C. Sound containment evacuation alarm and establish containment closure.

D. Have security evacuate non-essential personnel from containment.

Answer:

B. Perform a plant page announcement to evacuate personnel in containment.

Reference: EOP 3505 "Loss of Shutdown Cooling"

Justification: EOP 3505 "Loss of Shutdown Cooling" directs the operators to make a plant page announcement to evacuate personnel in containment. Selection 'B' is the correct answer.

K/A: 194001A1.04 3.0

Objective:

Comments: NRC change- replace distractor A
RO #98

Question:

Which one of the following describes when a non-intent procedure change must be reviewed by PORC?

A. prior to implementation

B. within 7 days of implementation

C. within 14 days of implementation

D. within 30 days of implementation

Answer:

C. within 14 days of implementation

Reference: DC-1

Justification: Procedural requirement

K/A: 194001A1.13 4.3/4.1

Objective:

Comments: RO #99

Question:

Unit 3 is running at 100% and the latest chemistry samples indicate RCS pH is high, but still in the acceptable range. Which one of the following actions is prescribed to lower RCS pH?

A. add boric acid to the RCS

B. add Hydrazine to the RCS

C. align letdown flow through the delithiating demineralizer

D. purge the VCT with hydrogen

Answer:

C. align letdown flow through delithiating demineralizers

Reference: CP 3802A
OP 3304A

Justification: Boric acid will have a detrimental effect on reactivity control. Hydrazine and purging the VCT with hydrogen will act to raise pH. High pH is caused by a buildup of lithium, which will be removed by the demineralizer.

K/A: 194001A1.14 2.5

Objective: RCS06C

Comments: NRC change- distractor C, purification vice delithiating. Will keep delithiating as it is a plant-specific identifier.

Question:

The shift supervisor has decided to evacuate the control room due to a plant fire. Which one of the following are **NOT** required actions in accordance with EOP 3503 "Shutdown Outside the Control Room"?

A. Control room operators close the MSIVs.

B. Primary plant equipment operators verify proper MCC Rod Control area ventilation.

C. Secondary plant equipment operators locally verify turbine stop valves are closed.

D. Outside plant equipment operators verify proper TDAFW pump operation.

Answer:

D. Outside plant equipment operators verify proper TDAFW pump operation.

Reference: EOP 3503 "Shutdown Outside the Control Room

Justification: Procedural requirement

K/A: 194001A1.10 3.9

Objective: 60003C

Question:

Which one of the following groups contains individuals, all of which can give authorization to startup the reactor after a plant trip?

A. Station Senior Vice President, Unit Director, Operations Manager

B. Unit Director, Operations Manager, Duty Officer

C. Operations Manager, Duty Officer, Station Senior Vice President

D. Shift Supervisor, Unit Director, Operations Manager

Answer:

A. Station Senior Vice President, Unit Director, Operations Manager

Reference: COP 200.1 pg. 30.

Justification: The reference listed states the authorization requirements to startup the reactor following a plant trip. The shift supervisior and Duty Officer are the only individuals on this list who cannot make that authorization.

K/A: 194001A1.11 4.1

Objective: 60003C

Comments: NRC change- change stem & distractors to eliminate NOT

Question:

The following conditions exist:

- A 10CFR50.72 report has been issued.
- Classification was an Unusual Event Delta-Two.
- Twenty minutes later it is determined that the report is incorrect and needs to be retracted.

Which of the following actions are required regarding the retraction of the report?

A. Reports associated with an Unusual Event or higher may not be retracted.

B. Retraction is not necessary as long as the classification was higher than actual conditions required.

C. Cannot retract any 10CFR50.72 reports. Another report will have to be issued to clarify the initial report.

D. Document reason for the retraction, notify the NRC of the retraction, and enter notification of the retraction in the control room log.

Answer:

D. Document reason for the retraction, notify the NRC of the retraction, and enter notification of the retraction in the control room log.

Reference: EPIP 4400 pg12.

Justification: EPIP 4400 states that in order to retract a report you must Document reason for the retraction, notify the NRC of the retraction, and enter notification of the retraction in the control room log. Selection 'D' is the correct answer.

K/A: 194001A1.16 4.4

Objective: E0005C

Comments:

Question:

You are exiting a contaminated area inside the RCA and a PCM is unavailable. There are no instructions from Health Physics posted in the area. Which one of the following identifies the MINIMUM requirements for performing a whole body frisk using a portable detector?

A. Background must be less than or equal to 200 cpm and survey results must be less than 100 cpm above background.

B. Background must be less than or equal to 200 cpm and survey results must be less than 200 cpm above background.

C. Background must be less than or equal to 300 cpm and survey results must be less than 100 cpm above background.

D. Background must be less than or equal to 300 cpm and survey results must be less than 200 cpm above background.

Answer:

C. Background must be less than or equal to 300 cpm and survey results must be less than 100 cpm above background.

Reference: RPM 5.3.1

Justification: Procedure requirement

K/A: 194001K1.03 2.8/3.4

Objective: G0003C

Comments:

SENIOR REACTOR OPERATOR

Candidate's Name: KEY

ANSWER SHEET

Multiple Choice: Circle or **X** your choice. If you change your answer, write your selection in the blank.

001	(A)	B	C	D	____	021	A	(B)	C	D	____
002	A	B	C	(D)	____	022	(A)	B	C	D	____
003	A	(B)	C	D	____	023	A	(B)	C	D	____
004	A	(B)	C	D	____	024	(A)	B	C	D	____
005	(A)	B	C	D	____	025	A	(B)	C	D	____
006	A	B	C	(D)	____	026	(A)	B	C	D	____
007	(A)	B	C	D	____	027	A	(B)	C	D	____
008	A	B	C	(D)	____	028	(A)	B	C	D	____
009	(A)	B	C	D	____	029	A	(B)	C	D	____
010	A	B	(C)	D	____	030	(A)	B	C	D	____
011	A	B	C	(D)	____	031	A	B	(C)	D	____
012	A	B	(C)	D	____	032	A	B	(C)	D	____
013	A	(B)	C	D	____	033	A	B	(C)	D	____
014	A	(B)	C	D	____	034	A	B	(C)	(X)	post exam change 11/29/95
015	A	B	C	(D)	____	035	A	B	(C)	D	____
016	A	B	C	(D)	____	036	A	(B)	C	D	____
017	A	B	(C)	D	____	037	A	(B) or	(C)	D	post exam change 11/29/95
018	A	B	C	(D)	____	038	A	B	(C)	D	____
019	A	B	C	(D)	____	039	A	(B)	C	D	____
020	(A)	B	C	D	____	040	(A)	B	C	D	____

Candidate's Name: _____

ANSWER SHEET

Multiple Choice: Circle or **X** your choice. If you change your answer, write your selection in the blank.

041	A	B	C	(D)	____	061	A	B	C	(D)	____
042	A	B	C	(D)	____	062	A	(B)	C	D	____
043	(A)	B	C	D	____	063	(A)	B	C	D	____
044	A	B	C	(D)	____	064	A	(B)	C	D	____
045	A	(B)	C	D	____	065	A	(B)	C	D	____
046	A	B	(C)	D	____	066	A	B	(C)	D	____
047	A	(B)	C	D	____	067	A	(B)	C	D	____
048	A	(B)	C	D	____	068	(A)	B	C	D	____
049	A	B	C	(D)	____	069	A	B	C	(D)	____
050	(A)	B	C	D	____	070	A	(B)	C	D	____
051	A	(B)	C	D	____	071	A	(B)	C	D	____
052	A	B	(C)	D	____	072	(A)	B	C	D	____
053	A	B	C	(D)	____	073	A	(B)	C	D	____
054	A	B	(C)	D	____	074	(A)	B	C	D	____
055	(A)	B	C	D	____	075	A	B	C	(D)	____
056	A	(B)	C	D	____	076	A	B	(C)	D	____
057	A	B	(C)	D	____	077	A	(B)	C	D	____
058	A	(B)	C	D	____	078	A	B	(C)	D	____
059	A	(B)	C	D	____	079	A	B	(C)	D	____
060	A	B	C	(D)	____	080	A	B	C	(D)	____

Candidate's Name: _____

ANSWER SHEET

Multiple Choice: Circle or **X** your choice. If you change your answer, write your selection in the blank.

081	A	B	(C)	D	____
082	A	B	(C)	D	____
083	(A)	B	C	D	____
084	A	B	(C)	D	____
085	A	B	C	(D)	____
086	A	(B)	C	D	____
087	A	(B)	C	D	____
088	A	(B)	C	D	____
089	A	(B)	C	D	____
090	(A)	B	C	D	____
091	A	B	(C)	D	____
092	A	B	(C)	D	____
093	A	(B)	C	D	____
094	A	(B)	C	D	____
095	A	B	(C)	D	____
096	A	B	(C)	D	____
097	A	B	C	(D)	____
098	(A)	B	C	D	____
099	A	B	C	(D)	____
100	A	B	(C)	D	____

ATTACHMENT 3

MP3 WRITTEN EXAMINATION COMMENT SUMMARY AND NRC RESOLUTION

Facility comments in regard to two questions were submitted to the NRC chief examiner prior to his review of the written examination results on November 29, 1995. The following is a summary of each of those comments along with its respective resolution by the NRC.

#RO 38/SRO 34

Facility Comment: With the synchroscope going slow in the slow direction, incoming (main generator) is going slower than running (grid). Main generator speed must be increased to get the synchroscope going slow in the fast direction and for the generator to pick up load when it is phased.

Recommendation: Change correct answer to "C".

NRC Resolution: Agree with facility in that correct answer should be "C" instead of "D".

#RO 43/SRO 37

Facility Comment: Answer key provides correct answer as "C". "B" is also correct, because the air ejector rad monitor (ARC-21) will close the blowdown isolation valves on high radiation.

Recommendation: Credit given for both answers "B" and "C".

NRC Resolution: Agree with facility in that credit should be given for both answers "B" and "C".

ATTACHMENT 4

SIMULATION FACILITY REPORT

Facility: Millstone 3

Facility Docket No: 50-423

Operating Tests Administered: November 27-December 1, 1995

This form is used only to report simulator observations. These observations do no constitute audit or inspection findings and are not, without further verification and review, indicative of noncompliance with 10 CFR 55.45(b). These observations do not affect NRC certification or approval of the simulation facility, other than to provide information that may be used in future evaluations. No licensee action is required in response to these observations.

The simulator, for the most part, correctly modeled the Millstone Unit 3 plant. During the simulator examination, the simulator froze up once during one of the scenarios; however, it was early enough that the examiners were able to continue on with the scenario, once the problem was identified and corrected. It was later determined that the freeze was attributable to a power surge to the entire training facility. Also, the NRC was informed that a simulator software upgrade and overall general maintenance was scheduled to follow shortly after the completion of the examination.