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DTE Energy

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NRC-10-0069

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Mr. Hironori Peterson
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Region III
U. S. Nuclear Regulatory Commission
2443 Warrenville Road, Suite 210
Lisle, Illinois 60532-4352

Reference: Fermi 2
NRC Docket No. 50-341
NRC License No. NPF-43

Subject: Fermi 2 Initial License Operator Post-Examination Material

Enclosed, please find the following post-examination material for the Fermi 2 Initial License Examination administered August 30 through September 4, 2010.

- A summary of substantive comments made by the applicants after the written examination including the basis for proposed changes to be made to the original written examination answer keys. This includes a discussion on whether the comment was accepted or rejected by the facility.

- Copies of Form ES-201-3 "Examination Security Agreement", as completed to date with final closeout signatures.

This material is being submitted in accordance with the requirements of NUREG 1021, Revision 9.

Should you have any questions in this matter, please contact Mr. David Coseo, General Supervisor, Operations Training at (734) 586-4055.

Sincerely,

Rodney W. Johnson
Manager, Nuclear Licensing

Enclosures:

1. Comments on SRO Questions
2. Comments on RO Questions
3. Exam Security Agreement to Date

cc: (w/o Enclosures)
NRC Project Manager
Reactor Projects Chief, Branch 4, Region III
NRC Resident Office
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Enclosure 1 to NRC-10-0069

Comments on SRO Questions

SRO Question 3

Numerous fire detection alarms for the Control Center complex have been received. The Control Room has been evacuated as a result of heavy smoke quickly filling the control room. Control Room personnel were unable to shutdown the unit prior to the evacuation and heavy smoke has prevented access to the Relay Room. The fire brigade has been dispatched.

Given the above conditions, the actions for shutting down the reactor are contained in _______(1)_______; and verification that the reactor is shutdown may be determined by _______(2)_______.

a. 1) 20.000.18, Control Of The Plant From The Dedicated Shutdown Panel
   2) verifying that the APRMs indicate power decreasing to less than 3%.

b. 1) 20.000.19, Shutdown From Outside The Control Room
   2) verifying that the APRMs indicate power decreasing to less than 3%.

c. 1) 20.000.18, Control Of The Plant From The Dedicated Shutdown Panel
   2) verifying that RPV injection rates and reactor pressure trends are consistent with decay heat levels.

d. 1) 20.000.19, Shutdown From Outside The Control Room
   2) verifying that RPV injection rates and reactor pressure trends are consistent with decay heat levels.

The correct answer given is C.

Candidate Comment:

The candidates contend that this question has two correct answers (A or C). Both A and C contain 20.000.18 procedure. Answer C is identified as being correct. Answer A should also be accepted due to being able to look at IPCS APRM power in any one of various locations such as the Technical Support Center, Emergency Operations Facility, Research Tagging Center, etc.

Facility Comment:

The station staff supports the candidates in that this question has two correct responses, namely responses A and C for the following reasons:

This question has two parts.

The first part requires the candidate to recognize that 20.000.18, Control of the Plant from the Dedicated Shutdown Panel is the correct procedure to use for the given plant conditions. This is true for both response A and C.

The second part requires the candidate to determine how to verify that the reactor is shutdown given the fact that both the Control Room and the Relay Room are inaccessible due to heavy smoke in each area. A review of the procedure (20.000.18, Control of the Plant from the Dedicated Shutdown Panel) does not indicate any method (preferred or otherwise) for this determination.

Response C is listed as the correct response in that the candidate must realize that an alternate means must be used to verify the reactor is shutdown such as lowering RPV injection rates and reactor pressure trends. The candidates and the station staff agree that these would be acceptable verification methods given the fact that the Control Room and Relay Room are inaccessible, and C is in fact a correct response, however this would not be the only method to determine that the reactor is shut down.

As such, the station staff contends that response A is also correct based on the fact that APRM power can be verified using computers located around the facility by monitoring our Integrated Process Computer System (IPCS). The exam key's explanation for response A being incorrect states the following:

- Correct procedure but APRM verification requires access to the Relay Room which is within the fire zone.

The second part of this response fails to take into consideration the operator's ability to remotely monitor APRM power (outside of the Control Room and Relay Room) using IPCS. Given the fact that the governing procedure (20.000.18) does not provide direction either way, both responses A and C give acceptable methods for determining that the reactor is shutdown. The SOP for IPCS (23.615) provides the following:

- The IPCS provides the capability of monitoring, recording, and displaying plant parameters via strategically located display consoles. In addition, the IPCS has the capability of being accessed via various LAN servers.

Recommendation:

The station staff contends that the answer key be modified to give credit for both responses A and C per NUREG 1021, ES-403; paragraph D.1.b, bullet 3 since we have provided "newly discovered technical information that supports a change in the answer key."

References:

20.000.18, 23.615.

SRO Question 5

Due to an error in the calibration procedure for the RPS Drywell Pressure instruments, the high pressure trip setpoint for all four channels were adjusted such that the channels would not trip until Drywell Pressure reaches 2.2 psig. Which one of the following would satisfy the required Technical Specification?

Readjust the trip setpoint for channels:

a. A and C to 2.0 psig within 6 hours.

b. B and D to 1.8 psig within 1 hour.

c. A and D to 1.8 psig within 1 hour.

d. B and C to 2.0 psig within 6 hours.

The correct answer given is B.

It is apparent that the key provided with the exam meant for C to be the correct answer (not B) as is evident by the explanation provided for each response (see below):

To restore trip capability for Drywell Pressure, at least one of the trip systems must be restored (trip setpoint less than 1.88 psig) within 1 hour. To restore a trip system the two channels for that trip system [(A or C) and (B or D)] must be restored.

a. Wrong combination and setpoint, incorrect time

b. Wrong combination, incorrect time

c. Correct combination and time

d. Correct combination, Wrong time and pressure

The station staff is in agreement with the above explanations and contends that the key had a typographical error and C is in fact the correct answer.

Candidate Comment:

The candidates contend that answers B or C are correct based on the following: The question asks about RPS logic (A or C) and (B or D). Per TS Bases on Page B 3.3.1.1-23, Condition C.1, states "A Function is considered to be maintaining RPS trip capability when sufficient channels are OPERABLE or in trip (or the associated trip system is in trip), such that both trip systems will generate a trip signal from the given Function on a valid signal. For the typical Function with one-out-of-two taken twice logic and the IRM and APRM Functions, this would require both trip systems to have one channel OPERABLE or in trip (or the associated trip system in trip)". Looking further at 23.601, Instrument Trip Sheets, pages 43 and 44, for High Drywell Pressure instruments and Table 3.3.1.1, Function #7, we could implement the TS Bases both ways, either restoring B and D within an hour then placing Trip System A in Trip, or restore A and D within an hour.

Facility Comment:

The station staff does not support the candidates on this basis and this question does, in fact, have only one correct answer. Answer C is correct, not B, for the following reasons:

- Response B would restore only RPS B functionality for the High Drywell Pressure trip Function. Per the Tech Spec bases, this would not satisfy the requirement to exit Condition C (1 hour ACTION to restore functionality). Per TS bases: "A Function is considered to be maintaining RPS trip capability when sufficient channels are OPERABLE or in trip (or the associated trip system is in trip), such that both trip systems will generate a trip signal from the given Function on a valid signal." Restoring B and D would only restore functionality to RPS B. RPS A would still not be functional for the High Drywell Pressure trip Function. The candidates stated that performing this action (restoring B and D) and placing the A trip system in trip would satisfy the LCO. Although this is true, this option was not provided in the question and, therefore, response B is not correct.

- Response C is correct because its actions would restore functionality to both RPS A and RPS B for the High Drywell Pressure trip Function. Restoring the A trip setpoint would restore one instrument to OPERABLE status for RPS A and restoring the D trip setpoint would restore one instrument to OPERABLE status for RPS B. This would satisfy the Tech Spec bases for Condition C in that both RPS trip systems would then have sufficient OPERABLE channels such that both trip systems would generate a trip signal for the High Drywell Pressure Function on a valid initiation signal.

Recommendation:

The station staff concludes that the key should be revised to reflect C as the correct answer in accordance with NUREG 1021, ES-403, paragraph D.1.b, bullet 2 since SRO Question 5 contained "unintended typographical errors in a question or on the answer key."

References:

23.601, Technical Specifications (LCO 3.3.1.1 and bases).

SRO Question 6

The plant was operating at 100% power. A failure of the governor / pressure regulator occurred, causing the turbine control valves to slowly close without a corresponding opening of the bypass valves.

The RPS actuated (first hit) on "High Neutron Flux". The "P603" operator stated that the RPS system failed to operate properly. Which one of the following correctly describes the operability of the RPS system?

a. The RPS system functioned as expected and is therefore operable.

b. The RPS system should have first tripped on "High Reactor Pressure" with the "High Neutron Flux (APRM)" trip as a backup.

c. The "Turbine Control Valve Closure" should have resulted in a Recirculation Pump Runback and the RPS system should NOT have tripped.

d. The "Turbine Control Valve Closure" should been the first RPS trip signal in anticipation of "High Neutron Flux" and/or "High Reactor Pressure".

The correct answer given is B.

Candidate Comment:

The candidates contend that both A and B are technically correct and should be accepted as possible answers with the following explanation: There is no information in the question that states that Reactor pressure did or did not reach 1093 psig. If there was more information on the rate of pressure change or the pressure that was reached, then a determination could have been made. But with the current information, it is not possible to distinguish. In order to determine OPERABILITY you would have to know if the Tech Spec pressure recorders actually reached 1093 psig before reaching 118% on Flux. The stem of the question asks "which one of the following correctly describes the OPERABILITY of the RPS system?" The candidates contend that nowhere in answer B is the question of OPERABILITY addressed. Lastly, the second part of answer B implies that the High Neutron Flux (APRM) trip is a backup to the High Reactor Pressure trip and this is not true.

Facility Comment:

The station has concluded that this question has no technical basis and should be removed from the exam. Below is the exam key's explanation for the basis for each response:

a. While the "High Reactor Pressure" trip is not credited in the analysis for protecting the fuel or RPV and is meant to backup the "High Neutron Flux" trip, it is expected to occur before the "High Neutron Flux" trip, especially on a slow moving transient.

b. The "High Reactor Pressure" trip is expected to occur first, especially on a slow moving transient, even though the analysis credits the "High Neutron Flux" trip for protecting the fuel and RPV.

c. No such runback exists.

d. TCV Closure trip is triggered by low hydraulic oil supply pressure when dump valves open, and not on valve position.

The exam key contends that B is the correct response based on the statement that the High Reactor Pressure trip is expected to occur first. The station staff could not find any information to support this either in Technical Specifications, the UFSAR or the Student Text (note: TS Bases 3.3.1.1 and Student Text ST-OP-315-0027, RPS, were listed as references on the exam key). Chapter 15 of the Fermi 2 UFSAR, and TS Bases for the High RPV Pressure Trip, do not include an analysis for a slow reactor pressurization transient as was described in the stem of this question.

The following excerpt is from TS Bases 3.3.1.1 for Function 3 (Reactor Steam Dome Pressure High):

> An increase in the RPV pressure during reactor operation compresses the steam voids and results in a positive reactivity insertion. This causes the neutron flux and THERMAL POWER transferred to the reactor coolant to increase, which could challenge the integrity of the fuel cladding and the RCPB. No specific safety analysis takes direct credit for this Function. However, the Reactor Vessel Steam Dome Pressure—High Function initiates a scram for transients that results in a pressure increase, counteracting the pressure increase by rapidly reducing core power. The overpressurization protection analysis of Reference 4 conservatively assumes scram on the Average Power Range Monitor Neutron Flux—Upscale signal, not the Reactor Vessel Steam Dome Pressure—High signal. Along with the SRVs, the reactor scram limits the peak RPV pressure to less than the ASME Section III Code limits.

From the above excerpt, the following facts can be gleaned:

1. No specific safety analysis takes credit for the High RPV Pressure Trip, which explains why little information can be found in the UFSAR regarding transients and this RPS Function. The Fermi 2 UFSAR Chapter 5 (Reactor Coolant System), Chapter 15 (Accident Analysis) and Tech Spec Bases were all reviewed for this purpose.

2. The High RPV Pressure Trip does function to initiate a scram on high RPV pressure.

3. The APRM Neutron Flux – Upscale Trip Function is the primary Function that is assumed to protect that reactor from over pressurization events. It is not the backup for the High Reactor Pressure Trip as was stated in response B of this question.

The following excerpt is from the Fermi 2 UFSAR, Chapter 15; Section 15.2.1.2.1.2 (The Effect of Single Failures and Operator Errors on a failure of the Pressure Regulator in the Closed Direction).

> The nature of the first assumed failure produces a slight pressure increase in the reactor until the backup regulator gains control. Because no other action is significant in restoring normal operation if the backup regulator fails at this time (the second assumed failure), the control valves will start to close, raising reactor pressure to the point where a flux or pressure scram trip will be initiated to shut down the reactor. At rated power, this event is less severe than the turbine trip where stop valve closure occurs.

As can be seen in the above excerpt, the UFSAR does not delineate between which RPS Function will cause the reactor to scram when both pressure regulator fails. It simply states that "a flux or pressure scram trip will be initiated to shut down the reactor."

The following excerpt is from TS Bases 3.3.1.1 for Function 2c (APRM Neutron Flux Upscale):

> The Average Power Range Monitor Neutron Flux—Upscale Function is capable of generating a trip signal to prevent fuel damage or excessive RCS pressure. For the overpressurization protection analysis of Reference 4, the Average Power Range Monitor Neutron Flux—Upscale Function is assumed to terminate the main steam isolation valve (MSIV) closure event and, along with the safety relief valves (SRVs), limits the peak reactor pressure vessel (RPV) pressure to less than the ASME Code limits. The control rod drop accident (CRDA) analysis (Ref. 5) takes credit for the Average Power Range Monitor Neutron Flux—Upscale Function to terminate the CRDA.

This supports the fact that the APRM Neutron Flux Upscale trip is not the backup to the High Reactor Pressure Trip Function for any transient.

Recommendation:

The station staff concludes that the key should be changed to remove this question from the exam in accordance with NUREG 1021, ES-403; paragraph D.1.b, bullet 3 since we have provided "newly discovered technical information that supports a change in the answer key."

References:

Technical Specifications (LCO 3.3.1.1 and Bases), Fermi 2 UFSAR (Chapters 5 and 15).

SRO Question 10

The plant is operating at full power with no equipment out of service when the following alarms are received:

- 2D82, REAC BLDG TORUS SUMP LEVEL HI-HI / LO-LO
- 7D72, MOTOR TRIPPED

Given that the above alarms have been validated to be the result of a high Reactor Building Torus Sump level, which one of the following correctly identifies the procedure(s) to be utilized and the next expected action to be directed?

a. Primary Containment Control; Terminate all discharges into the Torus except for those needed for EOP Response.

b. Secondary Containment and Rad Release; isolate all systems discharging into the area except those needed for EOP response or damage control.

c. Primary Containment Control; Operate available Torus Water Management System Pumps to restore and maintain Torus Water level between – 2" and + 2".

d. Secondary Containment Control and Rad Release; Operate available Reactor Building Torus Sump Pumps to restore and maintain sump level below the Maximum Normal Operating level.

The correct answer given is D.

Candidate Comment:

The candidates contend that this question has two possible correct answers, B and D. The stem of the question states that the above alarms have been validated. At Fermi 2 this means that we would follow ARP 2D82, which directs a check that both sump pumps are running on a control room back panel (at the setpoint of 39 inches increasing) since we do not have sump level indication available in the control room. This would satisfy step SCL-2 of Sheet 5 (which is Answer D). So the next expected action to be directed would be to isolate per SC-2 (which is Answer B). Deciding between Answers B and D is a matter of interpreting the information provided in the stem of the question to determine which actions have already been taken versus those yet to be taken.

Facility Comment:

The station staff concurs with the candidates that this question has two possible correct answers, B and D, as described below:

The stem of the question states:

- The plant is operating at full power with no equipment out of service when the following alarms are received:

  - 2D82, REAC BLDG TORUS SUMP LEVEL HI-HI / LO-LO
  - 7D72, MOTOR TRIPPED

  Given that the above alarms have been validated to be the result of a high Reactor Building Torus Sump level, which one of the following correctly identifies the procedure(s) to be utilized and the next expected action to be directed?

Response D is given as the correct answer. Response D stated:

- Secondary Containment Control and Rad Release; Operate available Reactor Building Torus Sump Pumps to restore and maintain sump level below the Maximum Normal Operating level.

The basis for D is as follows:

- Correct – Alarms are indicative of a condition (high sump/area water level) requiring entry into Secondary Containment Control and the first directed action is to operate available sumps to restore and maintain level.

The justification provided is correct. However, based on the information provided in the stem of the question, the candidates needed to interpret what actions had already been taken and what the next expected action to be directed should be.

Based on the nature of this particular sump and the indications available for this sump, the only way of validating (as stated in the question stem) that the alarm was the "result of a high sump level" is to verify that both sump pumps are running on the back of the control room P602 panel. Once this verification is made, EOP entry is required since both Torus sump pumps are verified to be running (Reference Table 13 of 29.100.01 Sheet, Secondary Containment Control and Rad Release). Since the stem of the question stated that "the above alarms have been validated to be the result of a high Reactor Building Torus Sump level" the candidates could reasonably conclude either of the following:

1. The sump pumps are running, which is the EOP entry condition for this Sump, and it is now time to Execute and Control parameters concurrently per Step SC-1 of 29.100.01, Sheet 5.

2. The sump pumps are running and the EOPs have been entered, Step SCL-1 is complete (this step is synonymous with the EOP entry condition), Step SCL-2 is complete (also synonymous with the EOP entry condition) and the next step to be directed is SC-2 (isolate systems).

For the other sumps on Sheet 5, after the alarm is received and verification (sump level) of EOP entry is made, then the next course of action would be to verify all sump pumps were running (Step SCL-2). For this particular sump, it could be assumed that step SCL-2 is already complete (due to verification of EOP entry conditions) or SCL-2 needs to be completed (verified) prior to continuing.

Based on the information provided in the stem of the question, there is no way for the candidate to determine what actions have been taken or what action(s) are to be directed next per the EOP flowcharts. Both actions (Operate available sump pumps to restore water level less than Max Norm and Isolate systems discharging into the area) are acceptable and appropriate given the way the EOPs are implemented for this particular sump.

Recommendation:

The station staff concurs with the candidates and concludes that the key should be revised to accept both responses B and D as correct answers in accordance with NUREG 1021, ES-403, paragraph D.1.b, bullet 1 since this is "a question with an unclear stem that confused the applicants or did not provide all the necessary information."

References:

2D82 (REAC. BLDG. TORUS SUMP LEVEL HI-HI/LO-LO), 29.100.01, Sheet 5 (Secondary Containment Control and Radiation Release EOP).

SRO Question 13

A reactor startup is in progress with Intermediate Range Monitor (IRM) Channel A INOPERABLE (circuit board removed for repair) and BYPASSED, when the following occurs:

- IRM Channel C indicates upscale at 125/125, irrespective of Range Switch position.
- IRM Channels B, D, E, F, G, and H indicate 15-32/40 on Range 7.
- ALL Average Power Range Monitors (APRMs) are DOWNSCALE.

Which ONE of the following addresses the Technical Specification REQUIRED ACTION?

a. Within 12 hours, ensure IRM Channel C in a TRIPPED condition; the Reactor Startup may continue, including entry into MODE 1.

b. SHUTDOWN, to MODE 3 within 12 hours, per GOP 22.000.04, Plant Shutdown from 25% power; less than the REQUIRED Intermediate Range Monitors are OPERABLE.

c. BYPASS the IRM Channel C ROD BLOCK using the joystick per 23.603, Intermediate Range Monitors; RESET the Half Scram, and continue the Reactor Startup.

d. BYPASS IRM Channel C ROD BLOCK by placing the Reactor Mode Switch in RUN per GOP 22.000.02, Plant Startup from 25% power; RESET the Half Scram, and continue the Reactor Startup.

The correct answer given is A.

Candidate Comment:

The candidates contend that this question has two correct answers, A and B, for the following reasons: Since IRM C cannot be bypassed, answer A could not be correct since control rod withdrawal could not continue and entry into MODE 1 cannot occur. We enter all applicable Actions for the LCO with only 2 of the 4 IRM's on the RPS A side, we enter Condition A (which is answer A), and we also enter Condition D for Condition A, B, or C, not being met, which causes us to enter Condition G (the actions of which are answer B).

Facility Comment:

The station staff disagrees with the candidates' conclusion that this question has two correct answers (A and B). Instead, it is the station's position that this question has no correct answer for the following reasons:

Response A is listed as the correct answer in the exam key. The explanation (from the exam's answer key) for this being the correct answer is given below:

- "Continued operation is permitted as long as the un-bypassed channel is placed in a tripped condition within 12 hours. Entry into MODE 1 would be permitted since IRMs are not required to be operable in MODE 1."

This justification fails to take into account the fact that, with IRM C failed upscale or placed in the Trip condition, a rod block will exist that will prevent further control rod withdrawal. Although entry into MODE 1 is permitted, as described in the exam key, Reactor Startup could not continue to MODE 1 as is stated in Response A.

Since the stem of the question stated that IRM Channel A was already INOPERABLE and in Bypass, it is impossible to Bypass IRM Channel C once it fails upscale. Therefore, performing the action of placing IRM C in a TRIPPED condition within 12 hours would be the correct action to address Tech Specs (first part of Response A), however the resultant Rod Block would prevent withdrawing control rods as would be necessary to establish 5 to 10% power and/or to clear the APRM Downscale Trip Setpoint, both of which are necessary prior to entering MODE 1 (reference 22.000.02, Step 6.2.28 for the conditions necessary to enter MODE 1). The following CAUTION from 22.000.02 is provided for reference:

CAUTION

The Reactor Mode Switch shall not be placed or maintained in RUN if the minimum number of operable APRM channels cannot be maintained above their downscale setpoints

Since the Tech Spec required Actions of Condition A (first half of response A) could be performed (i.e., place IRM C in a Tripped Condition), then the plant would not have to take the required Actions of LCO 3.3.1.1 Condition D (Enter the condition referenced in Table 3.3.1.1-1 for the associated instrument) and, subsequently would not have to perform Condition G (Be in MODE 3 in 12 hours), which is the basis for the first half of Response B actions.

Regarding the second half of Response B: Since LCO 3.3.1.1 (Table 3.3.1.1-1) requires 3 IRM Channels to be OPERABLE in MODE 2, with IRM A and IRM C both INOPERABLE, the plant is operating with less than the REQUIRED number of IRMs OPERABLE, as is stated in response B.

Recommendation:

Since the second part of response A cannot be performed due to the rod withdrawal block and the first part of response B is not necessary to comply with Technical Specifications, the station concludes that this question should be removed from the examination as per NUREG 1021, ES-403; paragraph D.1.b, bullet 3 since we have provided "newly discovered technical information that supports a change in the answer key."

References:

Technical Specifications (LCO 3.3.1.1), GOP 22.000.02, ARP 3D113 (Control Rod Withdrawal Block).

SRO Question 21

You have been assigned to review an upcoming refueling outage schedule to determine if the criteria for Infrequently Performed Tests or Evolutions (IPTE) are met. Which one of the following qualifies as an Infrequently Performed Tests or Evolution?

a. Performance of a passive surveillance activity (e.g., a visual inspection), that only affects one train of a multi-train safety related system, and that is only performed every other refueling outage.

b. An operational pressure test performed on a valve that was replaced in a safety related system.

c. Performance of a surveillance test that integrates several surveillance test activities, involving multiple systems, which were previously performed independently.

d. Core refueling activities, supervised by an experienced SRO (three consecutive refueling outages), and conducted by contractor crews of experienced fuel handlers.

The correct answer given is B.

Candidate Comment:

The candidates contend that C is the correct answer. Per MOP03, an evolution is an IPTE when it meets two (2) of the criteria listed under paragraph 3.6.1. The only response in the question that appears to meet two of these criteria is response C. The candidates propose that the activity described in response C could meet criteria 1, 3, 4 and/or 5. Response B, on the other hand, does not appear to meet any of the criteria (or 1 at the most) and would therefore not meet the definition of an IPTE.

Facility Comment:

The station staff also believes that C is the correct answer. It appears that the answer key incorrectly listed B as being correct when the explanation for the correct response (provided with the answer key) indicates that C is the correct answer.

Below is the Explanation of the four responses that were provided with this exam:

a. Incorrect, while the periodicity is infrequent, the activity is passive and only affects one train of a single system.
b. Incorrect, the test would involve normal operation of the system.
c. Correct
d. Incorrect since the personnel involved are experienced.

Recommendation:

The station staff concludes that the key should be changed to reflect C as the correct answer in accordance with NUREG 1021, ES-403, paragraph D.1.b, bullet 2 since the question contained "unintended typographical errors in a question or on the answer key."

References:

MOP03.

SRO Question 22

The unit is operating at 50% RTP. The 120 kV offsite circuit is declared inoperable for maintenance activities at 1000 on 9/20/10; all other systems are operable. Subsequently, the 345 kV offsite circuit is declared inoperable 4 hours later. Which one of the following describes the required TS ACTIONS?

a. The unit must be in Mode 3 by 0200, 9/21/2010.

b. The unit must be in Mode 3 by 2200, 9/20/2010.

c. Verify correct breaker alignment and indicated power availability for each offsite circuit within 1 hour and once per 8 hours thereafter.

d. Restore one offsite circuit to Operable status by 1400 on 9/21/2010

The correct answer given is B.

Candidate Comment:

The candidates contend that this question has two correct answers; C and D. Per TS 3.8.1, condition D, ACTION D.1, we Perform SR 3.8.1.1 for the first inoperability (at 1000) of the 120 kV offsite circuit. At Fermi-2 we accomplish this by performing 24.000.01, Attachment 29A, which directs the performance of Attachment 28B. These steps are described by response C of this question. Then, we would be required to perform condition E of TS 3.8.1 for the subsequent inoperability (at 1400) of the 345kV offsite circuit. These steps are described by response D of this question.

Facility Comment:

The station staff reply is as follows:

It appears that the key is incorrect for this question. The key lists B as the proposed correct answer, however the explanation provided clearly indicates that response D is intended as the correct answer. The explanation for each response is provided below:

a. Incorrect, Condition E is applicable. Shutdown required only if required actions and completion times of Condition D or E cannot be met.

b. Incorrect, Condition E is applicable. Shutdown required only if required actions and completion times Condition D or E cannot be met.

c. Incorrect, since action is only required for operable offsite sources.

d. Correct response. Condition E with 2 offsite circuits inop. Action E.1 is not required all other systems are operable (given in stem).

Concerning the candidates' comments for this question; the station staff's position is that this Question has two possible correct answers, namely C and D. The basis for this justification is described as follows:

Response D is one of two correct responses and would be the required Actions to take after the 345 kV offsite circuit was found to be inoperable subsequent to the 120 kV offsite circuit being declared inoperable. This is per TS LCO 3.8.1 Condition E, ACTION E.2.

Regarding response C, the exam key for response C states:

- Incorrect, since action is only required for operable offsite sources.

This statement is true only for the subsequent inoperability of both offsite circuits. However, a four hour gap exists between the first INOPERABILITY and the subsequent INOPERABILITY. Note that the stem of the question did not state if the TS 3.8.1 Condition D Action was completed following the initial INOPERABILITY of the 120 kV offsite circuit at 1000 on 9/20/10. The question did not specify which required Actions the question was looking for, nor did it ask for the most limiting Tech Spec Action.

A candidate could logically answer that, at time 1000, he had 1 hour to perform Action D.1 (which is response C) and then would be required to perform Action E.2 (which is response D) after the second inoperability occurred at 1400.

Recommendation:

The station staff concludes that the key should be changed to reflect C and D as possible correct answers in accordance with NUREG 1021, ES-403; paragraph D.1.b, bullet 1 since this was "a question with an unclear stem that confused the applicants or did not provide all the necessary information."

References:

Technical Specifications (LCO 3.8.1 and Bases).

Enclosure 2 to NRC-10-0069

Comments on RO Questions

<u>RO Question 3</u>

The plant is operating at full power, with no equipment out of service, when the following group of annunciators actuated:

- 10D68, DIV II ESS 130V BATTERY 2PB TROUBLE
- 2D24, DIV II CSS LOGIC POWER FAILURE
- 2D30, RHR LOGIC B 125V DC BUS POWER FAILURE

Assuming that <u>no other alarms were received</u>, the above group of annunciators indicates that:

a. ONE Division 2 battery ONLY has been lost.

b. BOTH Division 2 batteries ONLY have been lost

c. ONE Division 2 battery and the associated Battery charger have been lost.

d. BOTH Division 2 batteries and associated battery chargers have been lost.

The correct answer given is C.

Candidate Comment:

The candidates stated there would be several other alarms associated with the condition stated in answer C. In addition, there is no singular loss of any battery (or battery and charger combinations) that would result in the conditions listed in the stem.

Facility Comment:

The station staff supports the candidate's position that this is an invalid question.
Loss of 2B-2 results in the following ADDITIONAL alarms:

- 1D31, ADS DRYWELL PRESS BYPASS TIMER INITIATE A/B LOGIC
- 1D56, RCIC LOGIC BUS POWER FAILURE
- 1D57, ADS/SRV/EECW TCV POWER SUPPLY FAILURE
- 1D62, STM LK DET HPCI LOGIC POWER FAILURE
- 2D5, TESTABILITY DIV II ECCS LOGIC/POWER FAILURE
- 2D27, REACTOR PRESSURE LOW
- 2D48, HPCI/RCIC SUCTION TRANS CST LEVEL LOW

Loss of 2B-1 results in the following alarms:

- 10D1, EDG 13/14 STARTING AIR TANK PRESSURE LOW
- 10D68, DIV II ESS 13OV BATTERY 2PB TROUBLE

- 1D74, RCIC VALVES MTR OVERLOAD /LOSS OF POWER
- 2D50, HPCI LOGIC BUS POWER FAILURE
- 2D54, HPCI INVERTER CIRCUIT FAILURE
- 2D60, HPCI TURBINE EXH/EXH DRN VAL NOT FULLY OPEN

The station staff concludes that this question is faulty for two reasons:

- The statement, "assume no other alarms were received" does not represent an accurate alarm status for the candidates to make an assessment of the impact of this event. As can be seen above, a loss of either of the Division 2 batteries would result in several more alarms than those listed in the stem of the question. Providing an incomplete list added confusion to this question.

- At Fermi 2, logic systems can be powered from either of the two Division 1 or two Division 2 batteries. It is not expected that an operator would be required to distinguish between which of the two Division 1 (or Division 2 as in the case of this question) batteries supplies power to a particular control circuit. The ability to use a list of alarms to distinguish between losing either, or both, of a Division's batteries does not discriminate between a competent and non-competent operator.

The second bullet above was brought up, during exam validation, when this question received the following comment from the Licensed Operator who validated this exam:

- This question needs a reference or its Level of Difficulty (LOD) is too high (i.e.>5 on a 5-point scale).

Recommendation:

The exam key should be modified to remove this question from the exam per NUREG 1021, ES-403; paragraph D.1.b, bullet 1 "a question with an unclear stem that confused the applicants or did not provide all the necessary information."

References:

20.300.260V ESF BUSES, 20.300.260V ESF, SD-2530-11, SD-2531-13, I-2215-02, I-2215-03, I-2215-05, I-2205-05, I-2205-07, I-2205-12 and Simulator.

RO Question 6

The Dedicated Shutdown System transfer switches are designed to:

a. Bypass all interlocks and breaker protective functions for the equipment to be operated from the Dedicated Shutdown System panels.

b. Transfer all power supplies for Dedicated Shutdown System related equipment to sources powered from the Division II ESF busses.

c. Transfer Dedicated Shutdown System related equipment:

- instrument and control power to sources powered from the Division II ESF busses; and
- controls to locations external to the Main Control Room.

d. Transfer Dedicated Shutdown System related equipment:

- instrumentation and control power to sources external to the Control Center Complex; and
  controls to locations external to the Control Center Complex.

The correct answer given is D.

Candidate Comment:

The candidates stated the question has no right answer based the use of the phrase "transfer of control power to sources external to the Control Center Complex."

Facility Comment:

The station staff does not agree with the candidates. Surveillances are required to verify operability of the Dedicated Shutdown panel. Within these surveillances, instrumentation and control power for several pieces of equipment is verified.

Recommendation:

No changes to the exam or answer key are recommended.

References:

24.321.05, 24.321.06, and 24.321.08

<u>RO Question 14</u>

A small break LOCA has resulted into entry of EOP 29.100.01 Sheet 2, "Primary Containment Control". Drywell temperature has increased to 150°F despite efforts to maintain temperature less than 145°F by operating the Drywell Cooling System per the System Operating Procedure (SOP 23.415). You are now directed to operate ALL available Drywell Cooling per 29.ESP.08.

Select the response below that best summarizes how the Drywell Cooling system is operated per 29.ESP.08 to <u>maximize</u> cooling of the Drywell.

	TWO SPEED DW CLG FANS	SINGLE SPEED DW CLG FANS	RBCCW/EECW TO DW COOLERS
a.	4 fans in SLOW	10 fans in RUN	ISOLATED
b.	4 fans in FAST	10 fans in RUN	UNISOLATED/ RBCCW Supplying
c.	4 fans in FAST	10 fans in RUN	ISOLATED
d.	4 fans in FAST	10 fans in RUN	UNISOLATED/ EECW Supplying

The correct answer given is B.

Candidate Comment:

The candidates stated there is no procedural guidance to differentiate between which section of 29.ESP.08 will maximize cooling. Additionally, the stem of the question does not provide enough information to distinguish between B and D. Therefore, there are two correct answers, B and D.

Facility Comment:

The station staff supports the candidates' position. In accordance with 29.ESP.08, three options are available to the Operator to restore Drywell Cooling; the option to restore (1) RBCCW, (2) EECW, or (3) a combination of both. No direction is given in 29.ESP.08 on how to maximize cooling and the Operators are left with the ability to use the procedure, based on plant conditions, to control Drywell Temperature with any of the above 3 options.

The explanation given for distracter D states:

- EECW doesn't have the capacity to supply ECCS related load and Drywell coolers also.

29.ESP.08 provides procedural guidance to allow EECW to be unisolated and supplying Drywell loads as long as Drywell Temperature is less than 242°F. Thus, procedurally an operator given the direction to Operate ALL Drywell Cooling system per 29.ESP.08 would have the option to use either RBCCW or EECW and the procedure does not direct a preferred method. This makes answers both answers B and D correct.

There is no specific direction as to whether EECW or RBCCW would be the preferred method to maximize cooling. This is due to the environmental variables associated with both lake and ultimate heat sink temperatures. (Cooling water flow to the drywell is independent of the cooling source.) Therefore, it is not possible to conclusively determine which cooling medium would maximize cooling.

The following comment was provided during the validation process:

- B and D are possibly correct. The ESP does not differentiate.

This comment was not incorporated into the final administered version of this exam question.

Recommendation:

The exam key should be modified give credit for answers B and D per NUREG 1021, ES-403; paragraph D.1.b, bullet 3 "newly discovered technical information that supports a change in the answer key."

References:

29.ESP.08

<u>RO Question 21</u>

Synchronization and initial "block" loading of the Main Generator has just been completed. N30-R824, Main Cond Vacuum Pressure Recorder, indicates 0.5 psia.

At time zero, condenser vacuum begins to degrade (rising at 0.2 psi per minute).

Assuming that condenser vacuum degrades at a constant rate and that no operator action is taken in response to the degrading vacuum, which one of the following will be the first to result in actuation of a reactor scram?

a. Closure of the MSIVs

b. Trip of the Main Turbine

c. Trip of the Reactor Feed Pump

d. Closure of the Turbine Bypass Valves

The correct answer given is A.

Candidate Comment:

The candidates state there are complex thermodynamics involved that do not allow them to choose between MSIV closure and RPS High Pressure Scram. Therefore, either answer A or D is correct.

This question was asked of the exam proctor:

Q: *Is additional dynamics required to determine if at 6.8 psia , closing of the Bypass Valves will cause a High Pressure Reactor Scram at this power because MSIV's will not close for an additional 15 seconds.*

The following response was provided by the exam proctor:

A: *Information provided in the stem of the question is sufficient to answer the question.*

Facility Comment:

The station staff supports the candidates' position that there are two correct answers.

Based on the set points in 20.125.01, Loss of Condenser Vacuum, for degrading vacuum and the rate of the vacuum leak (0.2 psi/min), Bypass valves would be closed for 15 seconds prior to MSIV closure. The event which causes a scram becomes a matter of which will happen first, vacuum degrades to 6.85 psia (MSIV closure) or Reactor Pressure High Scram 1093 psig (due to Bypass Valve closure).

The explanation for answer D as a distracter is as follows:

- Main Turbine Bypass Valve Closure will occur at 6.8 psia, approximately 31.5 Minutes into the event, but because of the low power level, the resultant pressure transient will not result in a scram before the MSIV closure. Explanation supported by simulation.

The station was unable to recreate an exact 0.2 psi/min condenser leak. The station staff was able to create a simulation that provided data to determine how long it took to receive a High Pressure Reactor Scram following a Main Steam Bypass Valve closure.

Initial conditions prior to Main Steam Bypass Valve closure:

- Main Turbine block loaded and then subsequently tripped (this would happen at 2.7 psia).
- MS Bypass valves approximately 55% open per 22.000.02, Section 7.0.
- North Reactor Feed Pump, Gland Sealing Steam, and Steam Jet Air Ejectors in service.
  - T-0 seconds Bypass Valves Tripped
  - T-11 seconds Reactor Scram due to Reactor High Pressure

Note: MSIV's will not close until T-15 seconds per stem of the question.

The question is asking the candidates to determine, within 4 seconds, which would happen first. Depending on variables within the plant's house loads for steam, this time difference could be even less.

Thus, as shown above, choosing answer A over answer D does not discriminate between a competent and non-competent operator.

Recommendation:

The exam key should be modified give credit for answers A and D per NUREG 1021, ES-403; paragraph D.1.b, bullet 1 "a question with an unclear stem that confused the applicants or did not provide all the necessary information."

References:

20.125.01, and Site Simulator

RO Question 30

A calculation error resulted in all Division 1 Wide Range Reactor Water Level channels being calibrated to read higher than the Division 2 Wide Range Reactor Water Level Channels.

During a level transient the following RPV levels are observed:

- Division 1 Wide Range level instruments 115 inches
- Division 2 Wide Range level instruments 110 inches

Assuming NO operator action, which of the following conditions would the operator expect to see?

a. The MSIVs would be isolated

b. ALL Containment Isolation Valves would be closed

c. HPCI and RCIC would be actuated

d. HPCI and RCIC would NOT be actuated

The correct answer given is D.

Candidate Comment:

The candidates state C is the correct answer based on the RPV level instruments that supply HPCI and RCIC start logic and the fact that, even with the given instrument inaccuracies, enough instruments would sense the level reduction to cause a start of these two systems.

Facility Comment:

The station staff supports the candidates.

The exam stated D is correct based on the following:

- Both Division 1 Wide Range level instruments (N692A and B) read above the setpoint (110.8 inches), the 1 out of two logic (N692A or B AND N692C or D) is not satisfied.

Division 1 Wide Range level instruments are N692A and N692C. They are not N692A and B as stated in the exam key explanation. The Division 2 Wide Range level instruments N692B and N692D are below the actuation setpoint of 110.8 inches thus the 1 out of 2 logic (N692A or B AND N692C or D) is satisfied and both HPCI and RCIC would automatically actuate as stated in answer is C.

Recommendation:

The exam should be modified to change the correct answer to C per NUREG 1021, ES-403; paragraph D.1.b, bullet 3 "newly discovered technical information that supports a change in the answer key."

References:

23.601 (Rev 33) page 17, 6M721-5701-2.

RO Question 34

What is the impact on Intermediate Range Neutron Monitoring indication at P603 if there is a loss of power to 120 VAC UPS Distribution Cabinet B?

a. A loss of all IRM channel indications.

b. A loss of indication on IRM channels A, B, C, and D

c. A loss of indication on IRM channels E, F, G, and H

d. A loss of indication on IRM channels B, D, F, and H

The correct answer given is A.

Candidate Comment:

The candidates believe all answers are correct because each one of the responses would be true for the information provided in the stem of the question. There was not enough information provided in the responses to allow the candidates to discriminate an incorrect response.

Facility Comment:

The station staff agrees with the candidates' that all answers are technically correct.

Based on print I-2145-50 and ST-OP-315-0023-001, Intermediate Range Monitoring (IRM); Table 2 – IRM Power Supplies, all IRM recorders are powered from UPS B, ckt 9, (SD-2530-18). This question's responses were poorly worded which resulted in responses B through D all being subsets of response A. Questions similar to this used in this exam, included words such as ALL or ONLY to help the candidates to distinguish between the responses. For this reason, all of this question's responses are correct for the given loss of power to 120 VAC UPS Distribution Cabinet B.

Recommendation:

This question should be removed from the exam per NUREG 1021, ES-403; paragraph D.1.b, bullet 1 "a question with an unclear stem that confused the applicants or did not provide all the necessary information."

References:

I-2145-50, ST-OP-315-0023-001

RO Question 60

Which one of the following is **NOT** a concern with lowering Torus water level?

a. Air entrainment in the ECCS pumps

b. Ability to monitor Torus Water Temperature

c. Damage to SRV Tail Pipes and supporting structures

d. The ability to adequately suppress/condense steam discharged from the RPV

The correct answer given is C.

Candidate Comment:

The candidates state this question has two correct answers B and C. Their basis is that response B is not considered a concern because clear guidance is provided in 29.100.01, Curves, Cautions & Tables; Caution 6 which states:

- With Torus water level less than -11 inches Torus water temperature must be obtained from Drywell and Torus Air and Water Temperatures, Division I and Division II Recorders, points 11 and 12.

Therefore, the candidates contend, Torus Water Temperature can be monitored, as Torus level lowers, and the question has two correct answers (B and C).

Facility Comment:

The station staff supports the candidates assessment that there are two correct answers (B and C). EOP Caution 6 (see above) exists in the EOPs because the T23-R800 (Torus Water Temperature Recorder) on the H11-P601 becomes unreliable at -11 inches Torus Water Level. The Caution serves as a reminder to the Operators and provides direction to transfer Torus Water Temperature monitoring to the T50-R800A/B (Div 1/2 Primary Containment Air and Water Temperature Recorder) on the H11-P601 and P602 panels. One of these recorders is on the same Control Room panel as the T23-R800 and the other recorder is on the adjacent panel. Therefore the ability to monitor Torus Water Temperature is never lost; it just shifts from one location to another.

The following comment was provided during the written exam validation process (on 7/21/10) by Licensed Operators who took this exam:

- B and C are acceptable because TWT can be indicated to bottom of Torus.

Recommendation:

The exam key should be modified to allow two correct answers (B and C) per NUREG 1021, ES-403; paragraph D.1.b, bullet 3 "newly discovered technical information that supports a change in the answer key."

References:

29.100.01, SH 6, Curves, Cautions & Tables

RO Question 61

A pipe break at the juncture of the FPCCU supply to the Spent Fuel Pool and RHR FPCCU Cooling Assist return line will result in Spent Fuel Pool Level stabilizing at a level equal to the elevation ________.

a. of the line break.

b. of the lowest weir setting.

c. where the vacuum relief line openings are uncovered.

d. of the bottom edge of the supply lines at their highest point.

The correct answer given is D.

Candidate Comment:

The candidates believe there are two correct answers (C or D) based on the function of the vacuum relief line.

Facility Comment:

The station staff supports the candidates position that both C and D should be accepted as correct answers based on the function (and operation) of the vacuum relief line and the fact that it is this function that causes the level drop to stop in both responses C and D of this question.

The Functional System Description, student text, and EDP 10109 each correctly state that the vacuum relief lines on the supply lines for FPCCU will stop siphoning the fuel pool as soon as the relief line is uncovered (just above the Technical Specification allowable level of $\geq$ 22 feet above irradiated fuel in the fuel pool, and just below the lowest weir setting). The purpose of the vacuum relief lines is not to maintain the water level $\geq$ 22 feet above the spent fuel. The purpose of the vacuum relief lines is to prevent siphoning the water from the fuel pool and uncovering the spent fuel. The student text provides some additional insight that can only be gained from piping drawings.

The return lines penetrate the Fuel Pool below the normal water line, travel horizontally, then downward to an elevation near the bottom of the fuel pool with no check valves to stop the reverse flow during a pipe break. A break of the return line outside the Fuel Pool will cause pool level to lower until the siphon break is uncovered. Water level will then continue to drain the fuel pool (an additional 9 inches), until level in the fuel pool reaches the bottom of the pipe as it passes through the wall of the fuel pool. This information supports D as being a correct response.

However, the ability to choose between responses C and D does not distinguish between a competent and non-competent operator and is not linked to a licensed operator's job requirements.

It is the station staff's position that a competent operator is required to know that a pipe break in the FPCCU system will affect Spent Fuel Pool Water Level as follows:

- Pipe break at the pool outlet (pump suction line) will not result in draining the Spent Fuel Pool due to the elevation of the lowest weir setting.

- Pipe break at the pool inlet (return line) will not result in draining the Spent Fuel Pool due to the return line including a vacuum relief system that prevents siphoning the pool.

The only difference between responses C and D in this question is the fact that after the vacuum relief lines are uncovered, the siphon is broken, and water level will naturally (due to gravity) continue to drain to the bottom of the return pipe elevation inside the pool. The design feature that actually prevents draining (siphoning) the Fuel Pool is the vacuum relief system, which is required operator knowledge and would be verified by a candidate who chose C as the correct response. The fact that Fuel Pool level will continue to lower (approximately 9 inches due to pipe diameter) is of little operational significance which results in minimal discrimination value between responses C and D.

Recommendation:

The exam key should be modified to allow two correct answers (C and D), per NUREG 1021, ES-403; paragraph D.1.b, bullet 4 "a question that is at the wrong license level (RO versus SRO) or not linked to job requirements."

References:

EDP 10109, M-5712-01, M-2048, M-3356-1, M-3369-1, ST-OP-0015-001 Fig 4, and FSD - G41-00-SD.

RO Question 65

With the plant operating at full power, the following alarms are received:

- 7D22, GEN SERV H2O SCREEN A H2O LEVEL HIGH/LOW
- 7D23, GEN SERV H2O SCREEN B H2O LEVEL HIGH/LOW

GSW Pump Pit level is 8.9 feet (568' 11") as indicated on P41-R802, GSW Pump Pit Level Recorder and is decreasing at a rate of 1 inch/hour. Which of the following actions is appropriate for these conditions?

a. Shutdown CW Makeup and Decant Pumps.

b. Operate Circ Water Makeup pumps to raise Circ Pond level to the high end of the band.

c. Place the Electric and Diesel Fire Pumps to OFF/RESET and the Diesel Fire Pump Controller to OFF.

d. Transfer GSW suction from the lake to the CW Reservoir.

The correct answer given is B.

Candidate Comment:

The candidates state there are two correct (B and D) answers based on procedural guidance. AOP 20.131.01 directs transferring suction from GSW to the CW reservoir due to lowering lake level (Condition J). Since the stem of the question states that lake level is lowering (1 inch/hour), the candidates contend that the AOP Action described in response D (transfer suction) is warranted. They add that the AOP does not provide a specific lake level at which to take this Action.

Facility Comment:

The station staff does not agree with the candidates. 20.131.01, Loss of General Service Water System BASES does not support a transfer of suction to the Circ water pond with the conditions given in the stem of the question and therefore B is the only correct answer.

Recommendation:

No changes to the exam or answer key are recommended.

References:

20.131.01, Loss of General Service Water System, 7D22, Gen Service H2O Header Pressure High/Low, 7D23, Gen Service H2O Header Pressure High/Low.

Question 74

The unit is in MODE 5 with Division 2 RHR operating in the Shutdown Cooling Mode. The RPV head has just been removed and fill of the Reactor cavity is ready to begin.

A rupture in the common suction line to RHR loops occurs resulting in a Group 4 Containment Isolation. RPV water level dropped to 95 inches (wide range level) before the associated Containment Isolation Valves fully closed.

Select the answer that (1) describes how RPV water level will be restored; and (2) the appropriate RPV level necessary to establish adequate decay heat removal.

a. (1) Available ECCS will automatically start and align to inject
   (2) Raise RPV water level to just below the Main Steam Lines (260" to 275" indicated on Flood-Up Level Instrument)

b. (1) An injection system (e.g., CS, LPCI, SBFW, Cond/FW) must be manually started and aligned for injection
   (2) Raise RPV water level to just below the Main Steam Lines (260" to 275" indicated on Flood-Up Level Instrument)

c. (1) Available ECCS will automatically start and align to inject
   (2) Raise water level until level in the Reactor Cavity is just below top of the Reactor Cavity Weirs (635" to 645" indicated on Flood-Up Level Instrument)

d. (1) An injection system (e.g., CS, LPCI, SBFW, Cond/FW) must be manually started and aligned for injection
   (2) Raise water level until level in the Reactor Cavity is just below top of the Reactor Cavity Weirs (635" to 645" indicated on Flood-Up Level Instrument)

The correct answer given is D.

Candidate Comment:

The candidates state there are two correct answers (B and D) that would provide adequate decay heat removal.

Response B is acceptable because RPV level must be restored to >220 inches in order to allow for natural circulation to occur within the RPV. Then an alternate decay heat removal method could be established. Response B provides conditions that would allow for these actions.

Response D is acceptable because RPV level must be restored to >220 inches in order to allow for natural circulation to occur within the RPV. Then an alternate decay heat removal method could be established. Response D provides conditions that would allow for these actions.

Either of the level bands provided in Responses B and D would, therefore, allow for further actions to be taken to establish adequate decay heat removal.

Facility Comment:

The station staff does not agree with the candidates that B and D are both correct. Rather, the station staff contends that no procedural guidance could be found that would support any of the answers. Thus, the station's position is there is no correct answer.

Procedural guidance to mitigate this event is provided in 29.100.01, Sheet 1 (EOPs) due to a Level 3 EOP entry set point being reached when level drops below 173 inches. Therefore, Level Override 3 states:

> **IF** SDC is required **THEN** Keep RPV water level 173 IN. to 255 IN.

Adequate Decay Heat Removal would be provided (Adequate Core Cooling assured) by restoring and maintaining RPV level within the EOP prescribed level band (for being in the SDC mode) of 173 to 255 inches.

During the exam validation, the licensed operators who took this exam responded that this question was a question with no direct procedural guidance.

No explanation or technical basis for any of the answers were included with the exam answer key.

Recommendation:

The station staff concludes that this is an invalid question since no procedural guidance exists to support any of the answers. Therefore, the question should be removed from the exam per NUREG 1021, ES-403; paragraph D.1.b, bullet 3 "newly discovered technical information that supports a change in the answer key."

References:

29.100.01, Sheet 1.

Enclosure 3 to NRC-10-0069

Exam Security Agreement

1. Pre-Examination

I acknowledge that I have acquired specialized knowledge about the NRC licensing examinations scheduled for the week(s) of 8/30/2010 as of the date of my signature. I agree that I will not knowingly divulge any information about these examinations to any persons who have not been authorized by the NRC chief examiner. I understand that I am not to instruct, evaluate, or provide performance feedback to those applicants scheduled to be administered these licensing examinations from this date until completion of examination administration, except as specifically noted below and authorized by the NRC (e.g., acting as a simulator booth operator or communicator is acceptable if the individual does not select the training content or provide direct or indirect feedback). Furthermore, I am aware of the physical security measures and requirements (as documented in the facility licensee's procedures) and understand that violation of the conditions of this agreement may result in cancellation of the examinations and/or an enforcement action against me or the facility licensee. I will immediately report to facility management or the NRC chief examiner any indications or suggestions that examination security may have been compromised.

2. Post-Examination

To the best of my knowledge, I did not divulge to any unauthorized persons any information concerning the NRC licensing examinations administered during the week(s) of 8/30/2010. From the date that I entered into this security agreement until the completion of examination administration, I did not instruct, evaluate, or provide performance feedback to those applicants who were administered these licensing examinations, except as specifically noted below and authorized by the NRC.

	PRINTED NAME	JOB TITLE / RESPONSIBILITY	SIGNATURE (1)	DATE	SIGNATURE (2)	DATE	NOTE
1.	TONY ROBERTS	SR. INSTRUCTOR / EXAM DEV.	Tony Roberts	2/25/10	Tony Roberts	9/8/10	
2.	Timothy J Barrett	Operations Training Supervisor	[illegible]	2/25/10	[illegible]	9-8-10	
3.	William L. Wade	Simulator Specialist	William L. Wade	2/25/10	William L. Wade	9-13-10	
4.	Penny R. Watkins	Clerical	Penny R Watkins	4-8-10	[illegible]	9-13-10	
5.	S. Roth	Human Performance Superintendent	[illegible]	5/11/10			
6.	PAT TARNACKI	SR INST / EXAM DEVELOPMENT	[illegible]	6/9/10	[illegible]	9/7/10	
7.	Gordon Richard	NSO / EXAM VALIDATION	[illegible]	7-21-10			
8.	[illegible] PARENT	NSO / EXAM VALIDATION	[illegible]	7/21/10			
9.	WILLIAM KUTTIG	NSO / EXAM VALIDATION	[illegible]	7/21/10			
10.	JEFF CLEMENTS	NSO / EXAM VALIDATION	[illegible]	7-21-10			
11.	Jeff Groff	SRO - Exam Validation	[illegible]	7-21-10			
12.	Bruce Moss	SRO - Exam Validation	[illegible]	7-21-10			
13.	STEVE CLARK	NSO - EXAM VALIDATION	[illegible]	8-2-10	[illegible]	9-7-10	
14.	George Drayton	SRO - ILO Exam Validation	[illegible]	8/9/10			
15.	WILLIAM RICHARDS	NSO / EXAM VALIDATION	[illegible]	8/9/10			

NOTES:

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	PRINTED NAME	JOB TITLE / RESPONSIBILITY	SIGNATURE (1)	DATE	SIGNATURE (2)	DATE	NOTE
1.	D. Coseo	OPS TRG SUPV	[signature]	8/12/10	[signature]	9/8/10	
2.	B. Ager	OPS SNR INST	[signature]	8/31/10	[signature]	9/13/10	
3.	E Tarisius	SR OPS INST	[signature]	8/31/10	[signature]	9/13/10	
4.	S. Schmus	ILO Prog. Supv.	[signature]	8/31/10	[signature]	9/5/10	
5.	W.A. Conboy	" "	[signature]	8-31-10	[signature]	9/8/10	
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NOTES: