UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION III
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LISLE, IL 60532-4352

October 29, 2010

Mr. Jack M. Davis
Senior Vice President and
Chief Nuclear Officer
Detroit Edison Company
Fermi 2 - 210 NOC
6400 North Dixie Highway
Newport, MI 48166

SUBJECT: FERMI POWER PLANT, UNIT 2
NRC INITIAL LICENSE EXAMINATION REPORT 05000341/2010301(DRS)

Dear Mr. Davis:

On September 17, 2010, the U. S. Nuclear Regulatory Commission (NRC) completed an initial operator licensing examination at your Fermi Power Plant, Unit 2. The enclosed report documents the results of the examination. Preliminary observations and findings were discussed on September 3, 2010, with Mr. J. Davis, Manager Nuclear Training and others of your staff. On September 17, 2010, the NRC received licensee's post-examination comments on the written examination. An exit meeting was conducted by telephone, on October 14, 2010, between Mr. J. Conner, Plant Manager, and other members of your staff and Mr. D. Reeser, Chief Examiner, to review the resolution of the station's post-examination comments and the proposed final grading of the written examination for the license applicants.

The NRC examiners administered an initial license examination operating test during the week of August 30, 2010. The written examination was administered by NRC examiners and Fermi Power Plant, Unit 2 training department personnel on September 4, 2010. Four (4) Senior Reactor Operator (SRO) and three (3) Reactor Operator (RO) applicants were administered license examinations. The results of the examination were finalized on October 14, 2010. One RO applicant failed the written examination and was issued a proposed license denial letter. Five applicants (3 SRO and 2 RO) passed all sections of their respective examinations and were issued applicable operator licenses. The remaining SRO applicant passed all sections of their respective examinations, but because their final written examination grade was less than 82 percent, their license is being withheld, in accordance with NRC policy, pending the outcome of any written examination appeal that may be initiated by the applicant that failed the written examination.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter, its enclosure, and your response (if any), will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records System (PARS) component of NRC's Agencywide Documents Access and Management System (ADAMS), accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

We will gladly discuss any questions you have concerning this examination.

Sincerely,

/RA/

Hironori Peterson, Chief
Operations Branch
Division of Reactor Safety

Docket No. 50-341
License No. NPF-43

Enclosures:
1. Operator Licensing Examination Report 05000341/2010301(DRS)
2. Simulation Facility Report
3. Post Examination Comments and Resolutions
4. Written Examinations and Answer Keys (RO/SRO)

cc w/encl: Distribution via ListServ

U. S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket No:	50-341
License No:	NPF-43
Report No:	05000341/2010301
Licensee:	Detroit Edison Company
Facility:	Fermi Power Plant, Unit 2
Location:	Newport, MI
Dates:	August 30, 2010 through September 17, 2010
Examiners:	D. Reeser, Operations Engineer and Chief Examiner C. Zoia, Operations Engineer/Examiner N. Valos, Senior Risk Analyst/Examiner
Approved by:	Hironori Peterson, Chief Operations Branch Division of Reactor Safety

SUMMARY OF FINDINGS

ER 05000341/2010301(DRS); 08/30/2010 – 09/17/2010; Fermi Power Station, Unit 2; Initial License Exam Report.

The announced initial operator licensing examination was conducted by regional Nuclear Regulatory Commission examiners in accordance with the guidance of NUREG-1021, "Operator Licensing Examination Standards for Power Reactors," Revision 9, Supplement 1.

Examination Summary:

One applicant failed the written examination and was issued a proposed license denial. Six of seven applicants passed all sections of their respective examinations. Three applicants were issued senior operator licenses and two applicants were issued operator licenses. One senior operator license may be issued pending the outcome of any written examination appeal. (Section 4OA5.1)

REPORT DETAILS

4OA5 Other Activities

.1 Initial Licensing Examinations

a. Examination Scope

The U. S. Nuclear Regulatory Commission (NRC) examiners conducted an announced operator licensing initial examination during the week of August 30, 2010. The NRC examiners and members of the facility licensee's staff used the guidance prescribed in NUREG-1021, "Operator Licensing Examination Standards for Power Reactors," Revision 9, Supplement 1, to develop, validate, administer, and grade the written examination and operating test. The NRC examiners prepared the outline and developed the written examination and operating test. The licensee validated the proposed examination during the week of August 9, 2010. During the on-site validation week, the examiners audited four license applications for accuracy. The NRC examiners administered the operating test, consisting of job performance measures and dynamic simulator scenarios, during the period of August 30, 2010 through September 3, 2010. A NRC examiner with the assistance of members of the facility licensee's training staff administered the written examination on September 4, 2010.

b. Findings

(1) Written Examination

All changes made to the proposed examination, were made in accordance with NUREG-1021, "Operator Licensing Examination Standards for Power Reactors," and documented on Form ES-401-9, "Written Examination Review Worksheet," which is available electronically in the NRC Public Document Room or from the Publicly Available Records component of NRC's document system (ADAMS).

On September 17, 2010, the licensee submitted post-examination comments, on 17 questions, for consideration by the NRC examiners when grading the written examination. The results of the NRC's review of the comments are documented in Enclosure 3, Post Examination Comments and Resolutions.

The NRC examiners completed the final grading of the written examination on October 14, 2010, and conducted a review of each missed question to determine the accuracy and validity of the examination questions. In accordance with current NRC policy the release of this examination will be delayed until October 2012.

(2) Operating Test

All changes made to the proposed examination, were made and documented in accordance with NUREG-1021, "Operator Licensing Examination Standards for Power Reactors." Changes were documented in a document titled, "Operating Test Comments," which is available electronically in the NRC Public Document Room or from the Publicly Available Records component of NRC's document system (ADAMS). The NRC examiners completed operating test grading on October 14, 2010.

The following issues were identified during the conduct of the operating test and resulted in changes to the administered exam:

a) During validation of the operating test simulator scenarios, a change to the initial conditions for the first scenario was identified and incorporated into the scenario write-up, but was not incorporated into the simulator setup file. The pre-validation version of the scenario was intended to have the applicants perform a reactivity manipulation (control rod withdrawal) to increase power. A special setup Initial Condition (IC) was created to support that evolution by inserting control rods from a controlled IC. The initial power level of the resultant IC was therefore lower to allow for the control rod withdrawal. The reactivity manipulation was deleted during on-site validation to shorten the scenario run time. The revised scenario was meant to be run using the original (i.e., higher powered) controlled IC. During the first run of the scenario for the actual exam, the simulator operator loaded the lower power IC and as a result the Reactor Feed Pump Min Flow Valve failure did not require any operator actions as the control system was able to compensate without any operator intervention. The expected operator response to the component malfunction was not observable. The initial setup was corrected for subsequent runs of the scenario, and the expected operator responses were observed. Since all applicants participated in three simulator scenarios, the applicants were observed on sufficient number of instrument/component failures such that an appropriate licensing decision could be made.

b) Also during validation of the first scenario, Emergency Operating Procedure (EOP) entry, following a partial scram failure, was incorrectly identified. Because the resultant reactor power was less than 3 percent and reactor pressure vessel (RPV) water level did not drop below Level 3, no EOP entry conditions were met. The partial anticipated transient without scram (ATWS) was mitigated in accordance with the SCRAM Abnormal Operating Procedure (AOP). Because no EOP entry was made, the associated SRO applicants had to be evaluated during a third scenario, in the SRO position, to assess their implementation of the EOPs.

c) The third exam scenario was validated with the expectation that the operators would perform a controlled power reduction in response to a degradation of the Main Generator Stator Cooling Water heat exchanger. The first indication of the malfunction was an alarm associated with the turbine building closed cooling water (TBCCW) system differential pressure control valve, which required entry into the Loss of TBCCW Abnormal Operating Procedure. The validation process failed to identify applicability of a procedure override in the Loss of TBCCW Abnormal Operating Procedure which required tripping the Main Turbine Generator (preceded by shutdown of the reactor) at a stator inlet temperature of 131°F. This temperature was reached before any Stator Water system alarm condition was reached. Fortunately, the premature reactor shutdown did not have a major impact on the remainder of the scenario. The expected reactivity manipulation was not observed. All three applicant crews responded in a similar manner and no additional scenarios were required as a result of the unexpected response.

All three issues were result of poor validation of the simulator exam scenarios. However, none of the issues identified, prevented the ability to make valid licensing decisions.

(3) Examination Results

Four applicants at the Senior Reactor Operator (SRO) level and three applicants at the Reactor Operator (RO) level were administered written and operating tests. Five applicants (three SROs and two ROs) passed all portions of their examinations and were issued applicable operating licenses. One RO applicant failed the written examination and was issued a proposed license denial. One SRO passed all portions of the license examination, but received a written test grade less than or equal to 82 percent. In accordance with NRC policy, that applicant's license will be withheld until any written examination appeal possibilities by the RO applicant has been resolved. If the applicant's grade is still equal to or greater than 80 percent after any appeal resolution, the SRO applicant will be issued an operating license. If the applicant's grade has declined below 80 percent, the applicant will be issued a proposed license denial letter.

.2 Examination Security

a. Scope

The NRC examiners briefed the facility contact on the NRC's requirements and guidelines related to examination physical security (e.g., access restrictions and simulator considerations) and integrity in accordance with 10 CFR 55.49, "Integrity of Examinations and Tests," and NUREG-1021, "Operator Licensing Examination Standard for Power Reactors." The examiners reviewed and observed the licensee's implementation and controls of examination security and integrity measures (e.g., security agreements) throughout the examination process.

b. Findings

The NRC examiners observed, during the conduct of the operating test, several occurrences in which the facility licensee's staff failed to ensure that all place-keeping marks, in procedures used by the applicants, were erased, or applicable pages replaced between resets of the simulator. Additionally, there were a few occurrences where self-checking flagging tools were not removed from panels between resets of the simulator. Each of the above occurrences was evaluated for potential exam security compromise. The examiners concluded that the applicants did not receive any significant benefit, from the failure to clean-up the procedures or remove the flagging tools; therefore, no exam compromise occurred.

The above events were entered into the facility licensee's corrective action program and actions were initiated to prevent reoccurrence.

4OA6 Management Meetings

.1 Debrief

Mr. H. Peterson, Operations Branch Chief, presented the examination team's preliminary observations and findings on September 3, 2010, to Mr. J. Davis, Nuclear Training Manager, and other members of the Fermi 2 staff. No proprietary or sensitive information was identified.

.2 Exit Meeting

The chief examiner conducted an exit meeting on October 14, 2010, with Mr. J. Conner, Plant Manager, and other members of your staff, by telephone. The NRC's proposed disposition of the station's post-examination comments was disclosed and the revised preliminary written examination results were provided during the telephone discussion.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee
J. Conner, Plant Manager
G. Strobel, Manager Nuclear Operations
J. Davis, Manager Nuclear Training

NRC
H. Peterson, Chief Operations Branch
D. Reeser, Chief Examiner
N. Valos, Examiner
C. Zoia, Examiner

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened, Closed, Discussed

None

LIST OF ACRONYMS USED

AC	Alternating Current
ADAMS	Agency-Wide Document Access and Management System
AOP	Abnormal Operating Procedure
ATWS	Anticipated Transient Without Scram
CFR	Code of Federal Regulations
DRS	Division of Reactor Safety
EOP	Emergency Operating Procedure
ER	Examination Report
IC	Simulator Initial Condition
JPM	Job Performance Measures
NCV	Non-Cited Violation
NRC	U.S. Nuclear Regulatory Commission
QA	Quality Assurance
RO	Reactor Operator
RPV	Reactor Pressure Vessel
SDP	Significance Determination Process
SRO	Senior Reactor Operator
TBCCW	Turbine Building Closed Cooling Water
URI	Unresolved Item

SIMULATION FACILITY REPORT

Facility Licensee: Fermi Power Plant, Unit 2

Facility Docket No: 50-341

Operating Tests Administered: August, 30, 2010 through September 3, 2010

The following documents observations made by the NRC examination team during the initial operator license examination. These observations do not constitute audit or inspection findings and are not, without further verification and review, indicative of non-compliance with 10 CFR 55.45(b). These observations do not affect NRC certification or approval of the simulation facility other than to provide information which may be used in future evaluations. No licensee action is required in response to these observations.

During the conduct of the simulator portion of the operating tests, the following items were observed:

ITEM	DESCRIPTION
1	One or more control rod select pushbuttons exhibited sticking contacts which caused delays in the selection and insertion of control rods when required for ATWS mitigation and AOP (RR Pump Trip) response.
2	Unexpected/unplanned actuation of a Bypass Valve Unitized Actuator Fault alarm caused a distraction and prolonged the scenario while applicants responded to the alarm. A CRIS dot was placed on the alarm window for later runs of the scenario.
3	Unexplained lack of position indication for the Condensate Pump Discharge valves delayed start of scenario (low power scenario) while problem was researched.

POST EXAMINATION COMMENTS AND RESOLUTIONS

I. 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.

NOTE: The following information was provided by the facility in response to a follow-up question to confirm that the procedurally directed actions to initiate a reactor scram would also de-energize the APRMs.

Procedure 20.000.18, Subsequent Action B.1 directs the following to ensure that the reactor is shutdown:

> Depress Manual Scram pushbuttons.
>
> OR
>
> Open C71B-CB3A, H11-P608 Power Range Neutron Monitoring System A (RPS MG Set A Output Panel C71P001A in DC MCC Area AB3 G12)
>
> AND
>
> C71B-CB3B, H11-P608 Power Range Neutron Monitoring System B (RPS MG Set B Output Panel C71P001B in DC MCC Area AB3 G12)

From 20.000.18 Bases for Action B.1

> Depressing the Manual Scram Pushbuttons is the only manual action performed in the control room that is given credit in the NRC guidance document Generic Letter 86-10, Implementation of Fire Protection Requirements. Equivalent actions are given to account for the condition the SCRAM pushbutton wiring may be damaged in the fire.

Opening RPS breakers CB3A and B de-energizes neutron monitoring cabinets. APRM power loss would result. APRMs would indicate 0% power on IPCS due to the loss of power, which also consequently causes a full reactor scram by de-energizing the Voters. This indication would validate that the APRMs lost power and the Operator would conclude that his actions were successful in de-energizing RPS (and causing the associated shutdown). An operator could further validate reactor shutdown by also looking at the control rod positions.

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

NRC Resolution

AOP 20.000.18, “Control of the Plant from the Dedicated Shutdown Panel,” and AOP 20.000.19, “Shutdown from Outside the Control Room,” both provide guidance for shutting down the reactor if the operators were unable to do so before evacuating the Main Control Room. Because there was a fire in a 3L zone, AOP 20.000.18 was the controlling procedure. AOP 20.000.18 directed the operators to the RPS MG Set Output Panels, located in the DC MCC area of the Auxiliary Building, to open breakers supplying the Power Range Neutron Monitors (i.e., APRMs). If the reactor was shutdown in accordance with the direction given in AOP 20.000.18, power was removed from the APRMs (and LPRMs as well) and the instruments will fail downscale; the second part of response “a” cannot be completed. While observing that the APRM indication on IPCS read 0% may validate that the action to de-energize power to the APRMs was successful, as indicated in the facility’s supplemental response, the action does not validate that the reactor was shutdown and that power was decreasing. Alternate indications will have to be used. As stated in the facility’s response, AOP 20.000.18 does not specify any method (preferred or otherwise) for verification that the reactor is shutdown. The knowledge of how the secondary plant parameters relate to reactor power is fundamental to the ability of the operator to validate that primary indications are reading correctly. This is a skill that operators utilize on a routine basis. The amount of feedwater flow needed to maintain RPV water level, and trends in reactor pressure (which will be dependent on steam flow rates), were directly related to the power output of the reactor. Response “c” was the only response that contained both the correct procedure and an alternate action to verify that the reactor is shutdown.

Response “c” is the only correct answer and the answer key was not changed.

II. 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)

NRC Resolution:

As stated in the facility comments, response “b” would restore only the RPS B Trip System functionality for the High Drywell Pressure trip Function. The RPS A Trip System would still not be functional 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). A Function was considered to be maintaining RPS trip capability when sufficient channels were 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. The candidates stated that performing the action stated in response “b” (restoring B and D) and placing the RPS A Trip System in trip would satisfy the LCO. Although this was true, this option was not provided in the question and, therefore, response B was not correct.

As indicated above, the preliminary answer key incorrectly identified response “b” as the correct answer. The explanation of response choices in the “master exam” correctly identifies response “c” as the only correct answer. The answer key has been corrected to reflect that “c” was the correct answer.

III. 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. he 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 over pressurization 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 over pressurization 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.I.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)

NRC Resolution:

Response “b” was originally identified as the correct answer based on the following excerpt from TS Bases 3.3.1.1 for Function 3 (Reactor Steam Dome Pressure High):

> The over pressurization 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.

This bases statement centers on UFSAR section 5.2.2.3, “Overpressure Protection Analysis,” and the Reactor Protection System’s role as a “complementary pressure protection device” in the protection of the Reactor Pressure Vessel from an overpressure transient. The transient under discussion is the closure of all MSIVs (i.e., the limiting transient).

UFSAR Section 5.2.2.3.4.1 states in part:

> “The analysis hypothetically assumes the failure of the direct isolation valve position scram. The reactor is shut down by the backup, indirect, high-neutron flux scram.”

UFSAR Section 5.2.2.3.5, “Safety/Relief Valve Sizing” states in part:

> “The safety/relief valve capacity required for overpressure protection is determined from the minimum capacity that will provide an adequate margin between the peak vessel pressure and the vessel code limit (1375 psig) in response to the MSIV closure-flux SCRAM event.”
>
> “The MSIV closure-pressure SCRAM event is evaluated as confirmation of the safety/relief valve capacity determined from the safety/relief valve sizing criteria and to demonstrate the overpressure protection capability of the safety/relief valve system at the highest level of indirect SCRAM.”

These statements from the UFSAR seem to support the position that the high RPV Pressure trip was expected to occur before the high Neutron Flux trip during the MSIV closure, and by extension that the Neutron Flux trip backs up both the direct MSIV closure trip as well as the high RPV Pressure scam. However, the exam question was related to a less severe slow pressure transient and all the above discussion is related to the rapidly increasing pressure transient.

The pressure transients analyzed in Section 15.2 of the UFSAR (including the pressure regulator failure) all result in rapid pressure changes. The NRC was unable to find any documentation in the UFSAR, Technical Specification Bases, or the facility’s training materials, related specifically to the slow moving transient described by the exam question, which would support the position that the high RPV Pressure scram would always occur before the high Neutron Flux scram. As a result, answer choice “b” cannot be substantiated to be true under all conditions.

The NRC was also unable to find any documentation which would support the position that the high Neutron Flux scram would always occur before the high RPV Pressure scram. Thus, answer choice “a” cannot be substantiated to be true under the conditions specified in the question stem.

Since neither answer choice can be substantiated to be true for the transient in question, the NRC agrees with the facility that SRO Question 6 should be removed from the exam due to the lack of technical justification for implying that the Reactor Protection System was not operating properly if the high neutron flux trip occurs before the high RPV pressure trip.

IV. 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:

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-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.I.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)

NRC Resolution:

Given the information provided in the stem of the question, it was valid to assume that both Torus Area Floor Drain Sump pumps would be running, pending verification. The NRC disagrees with the facility comment that the only way to validate that the 2D82 alarm was the result of high level was to verify that both sump pumps were running on the back of the control room P602 panel. The concurrent actuation of the 7D72 alarm (trip of TWMS pumps) would provide some assurance that the alarm was the result of high level, as well as reports from the field. Verification that both sump pumps were running would probably be the quickest method. Since the stem of the question stated that the alarm condition had been validated to be the result of a high Reactor Building Torus Sump level (as opposed to a low level), entry into the Secondary Containment and Rad Release EOP (which was provided – with entry conditions removed – to the applicants) was indicated and concurrent execution to monitor and control the parameters of Step SC-1 implemented. No information was provided in the stem of the question to indicate any challenges to Secondary Containment temperatures or radiation levels, therefore the actions associated with monitoring and controlling Secondary Containment water levels would receive priority. As stated by the facility, with entry into the EOP, Step SCL-1 is complete. Therefore the next action to be directed upon entry into the EOP is SCL-2 (response “d”).

The facility contends that SCL-2 was complete, based solely on the fact that both sump pumps are running, and that it is appropriate to move on to step SC-2 (response “b”). The NRC disagrees; Step SCL-2 was not complete until either the sump water level was restored and maintained below the “MAX NORM” level, or the determination is made that the sump level cannot be restored and kept below the “MAX NORM” level. If operation of both sump pumps restores the sump level below the point that the pumps cycle off, no further action was required by the EOP, although location and isolation of the source of in-leakage to the sump would certainly be expected. No information was provided to support the assumption that the sump level cannot be restored and kept below the “MAX NORM” level, therefore moving on to Step SC-2 was not warranted given the information provided in the stem of the question and response “b” was not an acceptable answer.

There were no validation comments related to the sequencing of the EOP steps. One commenter stated that the EOP flow chart needed to be provided, and it was. The only other comment suggested replacing the 7D72 alarm with the statement that the TWMS pumps were tripped. This comment was not incorporated. Two of the four applicants correctly answered the question.

Response “d” remains the only correct answer and no change was made to the answer key.

V. 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 <u>not</u> 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 <u>not</u> 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)

NRC Resolution:

The explanation of response choices in the “master exam” clearly stated why responses “c” and “d” are incorrect, and the facility does not disagree with those explanations. The facility also agreed that response choice “b” was incorrect since the affected channel can be placed in a trip condition within the specified completion time of 12 hours. The facility also agreed that the first part of response “a” was correct. The only remaining contention was whether operation may continue, including entry into Mode 1.

The question asked: given the conditions specified in the stem of the question, which ONE of the responses addresses the Technical Specification REQUIRED ACTION. The question did not ask whether it was feasible to enter MODE 1, but simply whether the selected action was permissible under the requirements specified in the Technical Specifications. The Technical Specifications do not prohibit continuation of the startup and entry into MODE 1.

The limitation for continuation of the startup rests with the ability to override the control rod withdrawal block and enable control rod withdrawal. The Technical Specifications do not specify how to place the IRM channel in a tripped condition. As long as IRM channel A or C was placed in a trip condition (associated RPS trip relay was tripped with the other IRM channel bypassed), the Technical Specification required action was satisfied. The Technical Requirements Manual (TRM) requires that the control rod block functions for six (out of eight) IRM channels are operable. Since any single trip function can initiate a control rod block, there were no divisional restrictions. The affected IRM channel (A or C) could be placed in a trip condition to satisfy TS 3.3.1.1 while at the same time ensuring that there were no control rod block outputs. The control rod block functions for the affected IRM would be inoperable, but requirements of TR 3.3.2.1 would still be satisfied. This would permit continuation of the startup.

The actions, described in the previous paragraph, were not specifically identified in an approved procedure and would require approval in accordance the facility's plant modification and/or maintenance administrative control procedures, and it was unlikely that they would be approved in a timely enough manner to support staying at power under the conditions postulated in the question.

The NRC disagrees with the applicant's contention that there were two correct answers, but concurs with the facility's contention that there was no correct answer. The question was removed from the exam.

VI. 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.I.b, bullet 2 since the question contained "unintended typographical errors in a question or on the answer key."

References:

MOP03.

NRC Resolution:

As indicated above, the preliminary answer key incorrectly identified response “b” as the correct answer. The explanation of response choices in the “master exam” correctly identifies response “c” as the correct answer. The answer key has been corrected to reflect that “c” was the correct answer.

VII. 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.I is not required all other systems are operable (given in stem).

Concerning the candidates' comments for this question; the station staffs 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.I.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)

NRC Resolution:

The NRC disagreed with the facility's contention that response "c" was also a correct answer in that there was no information in the question to support the conclusion that Action D.1 (response "c"), which was required to be completed within one hour, had not been completed prior to the inoperability of the second offsite source. Even if it had not been completed, with the loss of the second off-site source, it was no longer required.

The facility's recommendation contends that the stem of the question may have confused the applicants or did not provide all of the necessary information. The applicable Technical Specification LCO (3.8.1), associated action statements, and associated surveillance requirements were provided to the applicants. No questions, asking for clarifying information on this question, were asked by any of the applicants during the exam. Additionally, only one of the four SRO applicants missed this question. It was not apparent to the NRC that the stem of the question was confusing or that incomplete information was provided.

As indicated above, the preliminary answer key incorrectly identified response "b" as the correct answer. The explanation of response choices in the "master exam" correctly identified response “d” as the correct answer. The answer key has been corrected to reflect that response “d” was the only correct answer.

VIII. RO Question 3

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 no other alarms were received, 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 130V 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, 1-2215-02, 1-221503, 1-2215-05, 1-2205-05, 1-2205-07, 1-2205-12 and Simulator

NRC Resolution:

At Fermi, the Divisional ESF DC Distribution System consisted of two 130 VDC batteries connected in series with 130 VDC loads being split between each of the 130 VDC batteries (and/or associated chargers) and 260 VDC load being connected across the series connected batteries.
The question was designed to test the applicants' knowledge that the Divisional ESF 130 VDC loads were not all supplied from the same voltage source (i.e., a battery and charger combination) and that the loads can be supplied by either the battery or its associated charger.

The facility's assertion, that the assumption stated in the question stem – that "no other alarms were received" – does not represent an accurate alarm status for the postulated event, was technically correct. The list of active alarms in the stem of the question was intentionally limited to simplify the evaluation of whether one or both DC distribution sub-systems (a battery, associated charger, and associated loads) were lost. The first alarm listed was common to both DC distribution subsystems, but the second two alarms were associated with ECCS subsystems that were both supplied by the same DC distribution subsystem. The alarms were chosen with the expectation that they would be easily identifiable given the importance of the systems. The NRC author of the question believed that providing a complete list of alarms would have added a greater degree of difficulty due to the similarities in alarm titles between the two sub-systems.

The facility response stated that a reviewer felt that the question was too difficult without a reference being provided. The reviewer did not provide any explanation as to why he thought the question was too difficult. The NRC had reviewed the facility's learning objectives for the High Pressure Coolant Injection System (ST-OP-315-0039-001), Core Spray System (ST-OP-315-0040-001), Residual Heat Removal System (ST-OP-315-0041-001), Automatic Depressurization System (ST-OP-315-0042-001), and the Reactor Core Isolation Cooling System (ST-OP-315-0043-001), and concluded that the question could be answered with knowledge related to a learning objective (A014), found in each of those lesson plans, stating that the student be able to "Describe the normal and alternate power supplies to [the associated system] components." While each of the above lesson plans listed the power supplies at the DC distribution sub-system level, the facility, in their comments above, had indicated that it does not expect the applicants to memorize the power sources beyond the divisional level (i.e., Division 1 DC or Division 2 DC).

Given this clarification of the facility's expectation for knowledge of the associated DC power supplies, the NRC agreed with the facility's contention that the question level of difficulty was beyond that expected for a question on an initial license examination.

The question was removed from the exam.

IX. 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

NRC Resolution:

The following information was extracted from the facility's training guide for the Dedicated Shutdown System (ST-OP-315-0099-001).

> The purpose of the Dedicated Shutdown System is to provide safe shutdown capability separate and remote from the control center complex (Control Room, Relay Room and Cable Spreading Room) and other sensitive fire zones when a fire in any of these zones is assumed to significantly damage the equipment and/or cabling in these zones.
>
> The Dedicated Shutdown Panel (H21-P623) is located in the BOP Switchgear Room on the Radwaste Building second floor. The Dedicated Shutdown Panel receives power for its instrumentation from an internal DC-to-AC Inverter that is powered from 125 VDC panel 2PC3-5 (BOP Battery).
>
> The six Local Dedicated Shutdown Panels are located in the area of their associated bus or motor control center. Power for the six Local Dedicated Shutdown Panels is taken from the MCC that it serves. Panel H21-P632 has an alternate power source of 72M-3B for Dedicated Shutdown operation that is connected by Transfer Switch R1600-S148.
>
> The DSS transfer switches are designed to perform 2 essential functions. The switches allow control of selected components from the Dedicated Shutdown Panel or the Local Dedicated Shutdown Panels. The switches also provide electrical separation between Control Room controls and Dedicated Shutdown Panel controls. The Local Shutdown Panel transfer switches provide electrical separation between the Main Control room, the Division I Remote Shutdown Panel, and the Local Panel controls.

The locations of the Dedicated Shutdown Panel and the six local panels, all external to the Control Center Complex, were chosen to ensure that the requirements for separation of circuits and equipment were satisfied.

NRC concurs with the facility that no changes to the exam answer key were required.

X. RO Question 14

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, ES403; paragraph D. .b, bullet 3 "newly discovered technical information that supports a change in the answer key."

References:
29.ESP.08

NRC Resolution:

The focus of the question was whether the applicant could differentiate between capabilities of the RBCCW and EECW systems and not on whether the procedure permitted operation of one system over the other. The response originally identified as the correct answer in the “master exam” was based on statements, in the facility’s training materials (ST-OP-315-0017-001), related to the basis for the automatic isolation of the RBCCW/EECW loads located within the Drywell. As stated in the training materials the basis for the automatic isolation was to ensure that EECW was not operated beyond the system’s thermal capacity. This can potentially compromise operation of other safety-related equipment located outside the Drywell. Isolation of the Drywell portion of EECW also assures that the other loads served by EECW will receive design cooling water flows.

Upon further review of the capabilities of the RBCCW and EECW systems, the NRC agreed that there was insufficient basis to state that one system was more capable of removing heat from the Drywell over the other. While the RBCCW system overall capacity for heat removal was significantly greater, it was based on being able to remove heat from non-essential loads as well as the essential loads. The flow rates through the Drywell coolers will be approximately the same regardless of which system was in operation, the difference in heat removal will be largely dependent upon the cooling water supply temperature. With the information supplied in the question, there was insufficient information to make a judgment on which system will remove more heat.

Based upon additional review of the available technical references provided, the answer was revised to accept either response “b” or “d” as correct answers. It was decided to accept both answers, rather than delete the question, since both answers are effectively the same.

XI. RO Question 21

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 additional15 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, ES403; paragraph D.I.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

NRC Resolution:

As indicated in the facility’s comments above, the Main Turbine Bypass Valves will receive a closure signal approximately 15 seconds before the MSIVs receive an isolation signal. In order to answer the question, the applicant would need to be able to determine whether reactor pressure would reach the High Reactor Pressure scram setpoint before the MSIVs begin to close resulting in a MSIV Closed (not full open) scram signal or vice-versa.

Based the facility’s demonstration on the simulator, the High Reactor Pressure scram occurred first but only four (4) seconds before the MSIV closure. As stated, the rate of increase in reactor pressure, and therefore the time to reach the High Reactor Pressure Scram Setpoint, would be largely dependent upon how much steam was being used by plant auxiliaries (Sealing Steam, number of Steam Jet Air Ejectors in service, whether feedwater heating was in-service, whether both feedwater pumps were in service or not, etc.) as well as the actual power level at the time of the event. The NRC agreed that four seconds was not a significant enough margin to state conclusively that one scram signal will always occur before the other.

During exam validation the answer was changed from “d” to “a” based on the facility reviewer’s feedback that closure of the Main Turbine Bypass valves would never cause a scram under the conditions specified in the stem. The validity of the statement was not verified. Additionally no written documentation could be located to support the position that either answer “a” or “d” was correct.

Based on the information provided by the facility and the inability to locate a written reference to support the position that either response “a” or “d” would likely occur in any predictable sequence, the question was deleted from the exam.

XII. 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

NRC Resolution:

As pointed out in the facility's comment above, the explanation of response choices incorrectly identified the Division 1 Wide Range RPV Level instruments as N692A and N692B, as opposed to N692A and N692C. The NRC verified this information using the facilities reference materials. The answer key and the explanation of response choices in the "master exam" were corrected to indicate that the correct answer was "c."

XIII. 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 1-2145-50 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, (SD2530-18). 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:

1-2145-50, ST-OP-315-0023-001

NRC Resolution:

As indicated by the facility, all IRM recorders are powered from the same circuit. This clearly supports that response “a” was the correct answer as the IRM recorders are the only IRM indication on P603. While it was true that each of the other response choices was a subset of response “a,” only response “a” identified that “all” recorder indications would be lost. None of the other responses provides a complete description of “the impact on Intermediate Range Neutron Monitoring indication at P603.” There was a significant difference between losing “all” IRM indication and losing only half of the indications.

Based the incorrect responses provided by the applicants (all five who missed the question, selected response “d”), it was readily apparent that the applicants believed the power to the recorders was divisionally separated. Since all of the IRM recorders were powered from the same power source and the question states that the power source was lost, no additional information was needed to answer the question. No questions were asked by the applicants during the exam to indicate that they needed additional information to answer the question or that they were confused about what the question was asking. The NRC disagreed with both the applicants and the facility that the applicants were confused by the information provided, or that not enough information was provided to answer the question.

The question will be retained as is and the answer key was not changed.

XIV. 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 candidate's 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 H1 -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 HI 1-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.I.b, bullet 3 "newly discovered technical information that supports a change in the answer key."

References:

29.100.01, SH 6, Curves, Cautions & Tables

NRC Resolution:

The NRC does not disagree with the applicants or the facility that Torus Water Temperature can be monitored over the entire indicated range of Torus Water Level. However, as pointed out by both the applicants and the facility, and as stated in EOP **CAUTION** 6 (emphasis added), below -11" the Torus Temperature shall be monitored using specific recorders. The author of the question assumed that, if there was no apprehension that the wrong indication would be used, there would be no need for EOP CAUTION 6.

Given the apparent confusion over the intended meaning, or significance, of the word "concern," the NRC has agreed to accept both responses "b" and "c." The answer key was changed accordingly.

XV. 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 > 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 > 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 staffs 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.I.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

NRC Resolution:

The question was written the test the applicants knowledge of the effect that a loss or malfunction of the Fuel Pool Cooling and Cleanup system – specifically a pipe break in the supply line – will have on Fuel Pool water level (K/A 233000.K3.02), by testing the applicant's knowledge of the physical relationship between key components in the Fuel Pool Cooling and Cleanup system and does not specifically ask for the purpose function of the vacuum relief, although this information was indirectly tested by the question. The question asked where the pool water level will stabilize when the transient was over.

The facility staff stated that it was their position that the question will not distinguish between a competent and non-competent operator and was not linked to a licensed operator's job requirements. The facility's training guide on for Fuel Pool Cooling and Cleanup System (ST-OP-315-0015-001) includes the following learning objectives:

- A007. Describe how the Fuel Pool Cooling and Cleanup System assist in maintaining the critical safety functions.
- A008. Discuss the function and purpose of Fuel Pool Cooling and Cleanup System components, including their importance to nuclear safety.
- A009. Describe the characteristics and locations of major Fuel Pool Cooling and Cleanup System components.
- C002. Discuss potential modes of Fuel Pool Cooling and Cleanup System component failures and any industry operating experience related to the failure.

All of the above learning objectives were relevant to the question being asked. A competent operator must be able to determine if the vacuum relief feature was/has functioning properly in the event of a pipe break in the supply line. As stated in the facility's response above and as discussed in the training materials, water level will continue to lower after the vacuum relief line is uncovered. An observer would not know if the vacuum relief is open, and not plugged, until water level stops lowering. The facility stated that the nine inch difference due to the pipe diameter was not operationally significant. However, if level continued to lower beyond the bottom of the supply line penetration, this would possibly indicate that the vacuum relief opening was plugged and that immediate attention would be warranted to ensure that the spent fuel remains adequately covered with water.

The NRC finds that the question was technically accurate, operationally valid, and was supported by the facility's training materials and learning objectives. There was only one correct answer and the answer key was not changed.

XVI. 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 H20 Header Pressure High/Low, 7D23, Gen Service H20 Header Pressure High/Low.

NRC Resolution:

The stem of the question stated that GSW pump pit level was lowering and not lake level as stated by the applicant. Response “d” was not required unless lake level was also dropping. The question was retained as is with only one correct answer.

XVII. 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 bases 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. .b, bullet 3 "newly discovered technical information that supports a change in the answer key."

References:

29.100.01, Sheet 1

NRC Resolution:

The question was written to test Generic K/A 2.4.9, "Knowledge of low power/shutdown implications in accident (e.g., loss of coolant accident or loss of residual heat removal) mitigation strategies. The question tested the applicants' knowledge of: ECCS response under the conditions given in the question stem; injection source availability; and how decay heat was removed from the reactor, with the RPV head removed, in the event that the normal decay heat removal method was lost. Question responses "a" and "c" were ruled out due to the fact that RPV water level had not lowered enough to automatically start the ECCS systems. Response to the question then rested on selection of either "b" or "d."

The facility staff's comment that no **[direct]** procedural guidance exists was partially true, but procedural guidance does exists to support an informed response to the proposed event. Additionally, the applicant was not expected to be able to answer the question based on memorized procedural guidance, but to be able answer the question by evaluating the plant conditions and arriving at a logical response based on his knowledge of decay heat removal methods.

Initial event response would be, as indicated in the facility response, in accordance with EOP 29.100.01 Sheet 1, “RPV Control.” EOP 29.100.01 provides guidance to restore and maintain water level within a band of 173 – 255.” Not mentioned in the facility’s response, was that AOP 20.205.01, “Loss of Shutdown Cooling,” would also be applicable. AOP 20.205.01 provides guidance (Step C) to restore water level to greater than 220.” This RPV water level was high enough to promote natural circulation within the RPV. While this will promote natural circulation, this will not establish a heat sink for decay heat removal. The level band stated in question response “b” was chosen, as an intended distractor, to coincide with a RPV level that would be consistent with using Operating Procedure 23.800.05, “Alternate Reactor Coolant Circulation and Decay Heat Removal Core Spray or RHR,” as directed by 20.205.01. Using this method would only be applicable if the plant was in MODE 4 such that raising level with the RPV head installed would raise pressure high enough to open one or more SRVs to establish a flow path to the Torus. If RPV level was to be maintained in any of the ranges specified above, the RPV would heat up until boiling commenced. This would not be a preferred heat removal method (because of containment issues), but adequate core cooling would be maintained and thus response “b” would technically be a correct answer.

The RPV level band specified in response “d” is consistent with the RPV level needed to establish the alternate decay heat removal method outlined in Operating Procedure 23.800.07, “Reactor Coolant Natural Circulation and Decay Heat Removal.” While this procedure is not specifically mention in AOP 20.205.01, it is consistent with the action specified in Step I of that procedure. This level range would support removal of the Fuel Pool Gates and establishment of heat removal using the Fuel Pool Cooling and Cleanup System.

The NRC disagreed with the facility’s conclusion that the question is invalid since, as pointed out above, procedural guidance does exist to support the answer choices. The NRC does agree with the applicants that, while not a preferred choice, response “b” was a technically correct answer. Therefore the answer key was revised to accept both “b” and “d” as correct answers. Additionally the explanation of responses in the “master exam” will be updated to reflect the justification for the correct answers.

WRITTEN EXAMINATIONS AND ANSWER KEYS (RO/SRO)

RO/SRO Initial Examination ADAMS Accession Number ML102990386.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter, its enclosure, and your response (if any), will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records System (PARS) component of NRC's Agencywide Documents Access and Management System (ADAMS), accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

Hironori Peterson, Chief
Operations Branch
Division of Reactor Safety

Docket No. 50-341
License No. NPF-43

Enclosures:
1. Operator Licensing Examination Report 05000341/2010301(DRS)
2. Simulation Facility Report
3. Post Examination Comments and Resolutions
4. Written Examinations and Answer Keys (RO/SRO)

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