UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION III
2443 WARRENVILLE ROAD, SUITE 210
LISLE, ILLINOIS 60532-4352

September 19, 2019

Mr. Paul Fessler, Senior VP
and Chief Nuclear Officer
DTE Energy Company
Fermi 2 – 260 TAC
6400 North Dixie Highway
Newport, MI 48166

SUBJECT: FERMI POWER PLANT, UNIT 2—NRC INITIAL LICENSE EXAMINATION REPORT 05000341/2019301

Dear Mr. Fessler:

On July 18, 2019, the U.S. Nuclear Regulatory Commission (NRC) completed the initial operator licensing examination process for license applicants employed at your Fermi Nuclear Power Plant. The enclosed report documents the results of those examinations. Preliminary observations noted during the examination process were discussed on June 26, 2019, with Mr. Ertman L. Bennett, Director–Nuclear Operations, and other members of your staff. An exit meeting was conducted by telephone on August 9, 2019, with Mr. Ertman L. Bennett, other members of your staff, and Mr. J. DeMarshall, Chief Operator Licensing Examiner, to review the final grading of the written examination for the license applicants. During the telephone conversation, NRC resolutions of the station's post-examination comments, received by the NRC on July 18, 2019, were discussed.

The NRC examiners administered an initial license examination operating test during the weeks of June 17, 2019, and June 24, 2019. The written examination was administered by Fermi Power Plant training department personnel on June 28, 2019. Five Senior Reactor Operator and nine Reactor Operator applicants were administered license examinations. The results of the examinations were finalized on August 8, 2019. Eleven applicants passed all sections of their respective examinations. Three applicants were issued senior operator licenses and eight applicants were issued operator licenses. Three applicants failed one or more sections of the administered examination and were issued Preliminary Results Letters.

The written examination, administered operating test, as well as documents related to the development and review (outlines, review comments and resolution, etc.) of the examination will be withheld from public disclosure until July 18, 2021. However, because three applicants received Preliminary Results Letters due to receiving a non-passing grade on the written examination, the applicants were provided copies of the written examination material. For examination security purposes, your staff should consider the written examination material uncontrolled and exposed to the public.

This letter, its enclosure, and your response (if any) will be made available for public inspection and copying at http://www.nrc.gov/reading-rm/adams.html and at the NRC Public Document Room in accordance with Title 10 of the *Code of Federal Regulations*, Part 2.390, “Public Inspections, Exemptions, Requests for Withholding.”

Sincerely,

/RA/

Robert J. Orlikowski, Chief
Operations Branch
Division of Reactor Safety

Docket No. 50–341
License No. NPF–43

Enclosures:
1. OL Exam Report 05000341/2019301
2. Post-Exam Comments, Evaluations,
   and Resolutions
3. Simulator Fidelity Report

cc: Distribution via LISTSERV®
A. Pullam, Director, Nuclear Training

Letter to Paul Fessler from Robert J. Orlikowski dated September 19, 2019.

SUBJECT: FERMI POWER PLANT, UNIT 2—NRC INITIAL LICENSE EXAMINATION REPORT 05000341/2019301

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REGION III

Docket No:	50-341
License No:	NPF–43
Report No:	05000341/2019301
Enterprise Identifier:	L-2018-OLL-0005
Licensee:	DTE Energy Company
Facility:	Fermi Power Plant, Unit 2
Location:	Newport, MI
Dates:	June 17, 2019, through June 28, 2019
Examiners:	J. DeMarshall, Senior Operations Engineer, Chief Examiner B. Bergeon, Operations Engineer, Examiner D. Reeser, Operations Engineer, Examiner
Approved by:	R. Orlikowski, Chief Operations Branch Division of Reactor Safety

SUMMARY

Examination Report 05000341/2019301; 6/17/2019–6/28/2019; DTE Energy Company, Fermi Power Plant, Unit 2; Initial License Examination 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 11.

Examination Summary

Eleven of fourteen applicants passed all sections of their respective examinations. Three applicants were issued senior operator licenses and eight applicants were issued operator licenses. Three applicants failed one or more sections of the administered examination and were issued Preliminary Results Letters. (Section 4OA5.1)

REPORT DETAILS

4OA5 Other Activities

.1 Initial Licensing Examinations

a. Examination Scope

The U.S. Nuclear Regulatory Commission (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 11, to develop, validate, administer, and grade the written examination and operating test. The written examination outlines were prepared by the NRC staff and were transmitted to the facility licensee's staff. Members of the facility licensee's staff prepared the operating test outlines and developed the written examination and operating test. The NRC examiners validated the proposed examination during the week of May 13, 2019, with the assistance of members of the facility licensee's staff. During the onsite validation week, the examiners audited four license applications for accuracy. The NRC examiners, with the assistance of members of the facility licensee's staff, administered the operating test, consisting of job performance measures and dynamic simulator scenarios, during the period of June 17, 2019, through June 26, 2019. The facility licensee administered the written examination on June 28, 2019.

b. Findings

(1) Written Examination

The NRC examiners determined that the written examination, as proposed by the licensee, was within the range of acceptability expected for a proposed examination. Less than 20 percent of the proposed examination questions were determined to be unsatisfactory and required modification or replacement.

During the validation of the written examination, several questions were modified or replaced. All changes made to the written examination were made in accordance with NUREG-1021, "Operator Licensing Examination Standards for Power Reactors," and were documented on Form ES-401-9, "Written Examination Review Worksheet." The Form ES-401-9, the written examination outlines (ES-401-1 and ES-401-3), and both the proposed and final written examinations, will be available electronically in the NRC Public Document Room or from the Publicly Available Records component of NRC's Agencywide Documents Access and Management System (ADAMS) on July 18, 2021 (ADAMS Accession Numbers ML17215A393, ML17215A384, ML17215A391, and ML17215A381, respectively).

On July 18, 2019, the licensee submitted documentation noting that there were ten post-examination comments for consideration by the NRC examiners when grading the written examination. The post-examination comments are documented in Enclosure 2 to this report.

The NRC examiners completed grading of the written examination on August 2, 2019, and conducted a review of each missed question to determine the accuracy and validity of the examination questions.

(2) Operating Test

The NRC examiners determined that the operating test, as originally proposed by the licensee, was within the range of acceptability expected for a proposed examination.

Following the review and validation of the operating test, minor modifications were made to several job performance measures, and some minor modifications were made to the dynamic simulator scenarios. All changes made to the operating test were made in accordance with NUREG-1021, "Operator Licensing Examination Standards for Power Reactors," and were documented on Form ES-301-7, "Operating Test Review Worksheet." The Form ES-301-7, the operating test outlines (ES-301-1, ES-301-2, and ES-D-1s), and both the proposed and final operating tests, will be available electronically in the NRC Public Document Room or from the Publicly Available Records component of NRC's ADAMS on July 18, 2021, (ADAMS Accession Numbers ML17215A393, ML17215A384, ML17215A391, and ML17215A381, respectively).

The NRC examiners completed grading of the operating test on August 8, 2019.

(3) Examination Results

Five applicants at the Senior Reactor Operator level and nine applicants at the Reactor Operator level were administered written examinations and operating tests.

Eleven applicants passed all portions of their examinations. Eleven applicants were issued their respective operating licenses on August 8, 2019. Two Senior Reactor Operator applicants and one Reactor Operator applicant failed the written examination portion of the administered examination and were issued Preliminary Results Letters.

.2 Examination Security

a. Scope

The NRC examiners reviewed and observed the licensee's implementation of examination security requirements during the examination validation and administration to assure compliance with Title 10 of the *Code of Federal Regulations*, Part 55.49, "Integrity of Examinations and Tests." The examiners used the guidelines provided in NUREG-1021, "Operator Licensing Examination Standards for Power Reactors," to determine acceptability of the licensee's examination security activities.

b. Findings

None

4OA6 Management Meetings

.1 Debrief

The chief examiner presented the examination team's preliminary observations and findings on June 26, 2019, to Mr. Ertman L. Bennett, Director–Nuclear Operations, and other members of the Fermi Power Plant, Unit 2, staff.

.2 Exit Meeting

The chief examiner conducted an exit meeting on August 9, 2019, with Mr. Ertman L. Bennett, Director–Nuclear Operations, and other members of the Fermi Power Plant, Unit 2, staff, by telephone. The chief examiner asked the licensee whether any of the material used to develop or administer the examination should be considered proprietary. No proprietary or sensitive information was identified during the examination or debrief/exit meetings.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee

E. Bennett, Director, Nuclear Operations
A. Pullam, Director, Nuclear Training
W. Conroy, Operations Training General Supervisor
E. Kokosky, Nuclear Oversight Director
J. Haas, Licensing Manager
S. Gatter, Principal Engineer Licensing
M. Donigian, Simulator/Exam Group Program Supervisor
J. Vanbrunt, Exam Developer
S. Schmus, Exam Developer
M. Smith, Exam Developer
B. Ager, ILT Program Supervisor

U.S. Nuclear Regulatory Commission

T. Briley, Senior Resident Inspector
T. Taylor, Resident Inspector
J. DeMarshall, Senior Operations Engineer, Chief Examiner
B. Bergeon, Operations Engineer, Examiner
D. Reeser, Operations Engineer, Examiner

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened, Closed, and Discussed

None

LIST OF ACRONYMS USED

ADAMS	Agencywide Documents Access and Management System
NRC	U.S. Nuclear Regulatory Commission

RO Question 4

After a Main Turbine Trip and the plant is stable 20.109.01, "TURBINE/GENERATOR TRIP" directs the following:

- Open Cutout Switch TCO/6.
- Open 2A and 2B 345kV Disconnects (CI-A and CI-B).
- Close 345kV Breakers CF and CM (23.300.02).
- Shutdown Main Turbine (23.109).

What is the reason for the realignment of the electrical system?

A. (1) Cutout Switch TCO/6 must be opened to allow 345kV Breakers CF and CM to be closed due to a trip signal from the open Main Generator Field Breaker.
(2) 2A and 2B 345kV Disconnects (CI-A and CI-B) must be open to prevent motorizing the generator after its shutdown when CF and CM are reclosed.

B. (1) Cutout Switch TCO/6 must be opened to allow 345kV Breakers CF and CM to be closed due to a trip signal from the open Main Generator Field Breaker.
(2) 2A and 2B 345kV Disconnects (CI-A and CI-B) must be open to satisfy an interlock to allow closure of 345kV Breakers CF and CM.

C. (1) Cutout Switch TCO/6 must be opened so that the Main Turbine Trip can be reset allowing the Shutdown Main Turbine to be completed.
(2) 2A and 2B 345kV Disconnects (CI-A and CI-B) must be open to prevent motorizing the generator after its shutdown when CF and CM are reclosed.

D. (1) Cutout Switch TCO/6 must be opened so that the Main Turbine Trip can be reset allowing the Shutdown Main Turbine to be completed.
(2) 2A and 2B 345kV Disconnects (CI-A and CI-B) must be open to satisfy an interlock to allow closure of 345kV Breakers CF and CM.

Answer: A

<u>Reference(s) provided to NRC</u>:

20.109.01, TURBINE/GENERATOR TRIP (Rev. 20)

20.109.01 BASES (Rev. 0)

ARP 4D65, GENERATOR PROTECTIVE RELAYING OPERATED (Rev. 17)

Applicant Comment:

Per 20.109.01, Condition D:

SUBSEQUENT ACTIONS (continued)

CONDITION	ACTION
	------------ NOTE 1 ------------
D. Plant stable.	❑ ❑ D.1 Open Cutout Switch TCO/6.
	❑ D.2 Open 2A and 2B 345kV Disconnects (CI-A and CI-B).
	------------ NOTE 2 ------------
	❑ D.3 Close 345kV Breakers CF and CM (23.300.02).
	❑ D.4 Shutdown Main Turbine (23.109).

Per 20.109.01, Bases Paragraph Action D.1 - D.3:

> "The normal Ring bus layout of the 345kV mat is designed to prevent loss of a single line or single breaker from causing a loss of power to the Div 2 busses. These steps restore the 345kV mat to its normal lineup. TCO/6 must be opened to allow closure of CM and CF breakers when the Field Breaker is open. Opening the CI-A and CI-B intermediate mat disconnects isolates the main unit transformers 2A and 2B from the 345kV mat allowing closure of the output breakers CF and CM."

Answer B, Part 2, states:

> "2A and 2B 345kV Disconnects (CI-A and CI-B) must be open to satisfy an interlock to allow closure of 345kV Breakers CF and CM."

The wording is practically the same as the bases per Subsequent Action D as the reasoning for opening the disconnects, which allows closure of CF and CM.

Answer A, Part 2, states:

> "2A and 2B 345kV Disconnects (CI-A and CI-B) must be open to prevent motorizing the generator after its shutdown when CF and CM are reclosed."

Per 20.109.01, Condition C:

C.	Main Generator load ≤ 0 MWs.	❑	❑ C.1	Open the following: ❑ Gen Output Breaker CM. ❑ Gen Output Breaker CF. ❑ Generator Field Breaker.

(continued)

Per 20.109.01 Bases, Paragraph Action C.1:

> "Opening Generator Output Breakers CF and CM, allow electrical separation from the Detroit Edison Grid. Failure of the breakers to open could motorize the generator, which may result in damage."

The bases for Subsequent Action C.1 states that the reason CF and CM are open is to prevent motorizing the generator, not CI-A and CI-B disconnects. This may be the correct answer but the wording may be misleading due to the confliction between the answer and the bases behind the action

***Facility Position on Applicant Comment*:**

Fermi station staff supports the applicant's position. Relevant wording in the basis document for 20.109.01 actions D.1 thru D.3 are remarkably similar to those of answer B, part 2):

- Basis statement: "*Opening the CI-A and CI-B intermediate mat disconnects isolates the main unit transformers 2A and 2B from the 345kV mat allowing closure of the output breakers CF and CM.*"
- Answer B part 2): "*2A and 2B 345kV Disconnects (CI-A and CI-B) must be open to satisfy an interlock to allow closure of 345kV Breakers CF and CM.*"

The exam author relied upon the inclusion of the words, "*to satisfy an interlock*", to make this an incorrect choice. However, since the basis document does not specifically elaborate on the reason for opening CI-A & B to allow CM & CF to close, an applicant who thoroughly studied the basis document and recalled these statements could reasonably conclude that answer B part 2) was restating the procedure step basis.

Additionally, although there is no interlock preventing CM & CF from closing while CI-A & B are also closed, if this action were attempted, both CM & CF will immediately trip due to the action of generator protective relaying (reference ARP 4D65). Thus, generator protective relaying is effectively providing an interlock that prevents CM and CF breakers from closing.

Therefore, station staff recommends accepting both A and B as correct per the third bullet of ES-403, Paragraph D.1.b., which allows for post-exam changes for questions with newly discovered technical information.

NRC Evaluation/Resolution:

Recommendation not accepted. The question tests applicant knowledge of the reason for realignment of the electrical system in accordance with 20.109.01, "TURBINE/GENERATOR TRIP," Condition D, "after the plant has stabilized" (emphasis added) following a trip of the Turbine/Generator. Part 2 of the answer specifically tests applicant knowledge of the reason why the 2A and 2B 345kV Disconnects (CI-A and CI-B) must be opened per Condition D, Subsequent Action Step D.2, which states: "Open 2A and 2B 345kV Disconnects (CI-A and CI-B)."

The applicant contends, citing the bases for Condition C, Subsequent Action C.1, that the reason Generator Output Breakers CF and CM are opened (and not the CI-A and CI-B Disconnects) is to prevent motorizing the generator. This is correct relative to Condition C, "Main Generator load ≤ 0 MWs," following a trip of the Turbine/Generator, "before the plant has been stabilized" (emphasis added), when CI-A and CI-B are in the closed position (Note: these disconnects are interlocked to not operate electrically with either the 345kV Breaker CF or CM closed). The CI-A and CI-B Disconnects are not operated under Subsequent Action C.1.

The applicant also contends, citing the bases for Condition D, Subsequent Actions D.1–D.3, and Part 2 of Distractor B, that the wording is "practically the same" in each with respect to the reason for opening the disconnects, i.e., that Disconnects CI-A and CI-B are opened to "allow closure of 345KV Output Breakers CF and CM." While the *"allow closure of"* phrasing is similar between the two, there is a significant difference in the actual wording of Distractor B, Part 2, which makes it incorrect. Distractor B, Part 2 states:

> "*2A and 2B 345kV Disconnects (CI-A and CI-B) must be open to satisfy an interlock to allow closure of 345kV Breakers CF and CM."*

There is "no interlock" preventing the reclosure of Output Breakers CF and CM while Disconnects CI-A and CI-B are also closed, as confirmed by the Facility in their "Position on Applicant Comment" response. In this lineup were attempted, reverse power would be applied to the Generator. The potential for motoring exists because the Generator has not been electrically isolated. Motorizing the Generator "after the plant has stabilized" (emphasis added) following a trip of the Turbine/Generator, is prevented by the conduct of procedurally directed switching activities specifically sequenced to ensure the opening of Disconnects CI-A and CI-B prior to reclosure of the CF and CM Output Breakers when restoring the 345kV mat to its normal lineup. The bases for Condition D, Subsequent Actions D.1 – D.3, states:

> *"Opening the CI-A and CI-B intermediate mat disconnects isolates the main unit transformers 2A and 2B from the 345kV mat allowing closure of the output breakers CF and CM."*

Opening Disconnects CI-A and CI-B accomplishes isolation of the Main Unit Transformers 2A and 2B from the 345kV mat, which electrically isolates the shutdown Generator, allowing closure of Output Breakers CF and CM without challenging protective relaying features. This precludes any possibility for motorization and subsequent reliance on the Generator Reverse Power Protective Relaying scheme to function properly in the event Output Breakers CF and CM are reclosed with Disconnects CI-A and CI-B still closed in; otherwise damage to the Generator could result. The only assurance that motoring can be prevented "after the plant has stabilized" following a trip of the Turbine/Generator, is to ensure complete electrical isolation of the Generator by opening Disconnects CI-A and CI-B prior to reclosing Output Breakers CF and CM when restoring the 345kV mat to its normal lineup in accordance with 20.109.01, Condition D, Subsequent Actions D.1–D.3.

In summary: (1) the applicant's answer choice B is incorrect because there is "no interlock" requiring Disconnects CI-A and CI-B to be in the open position to allow reclosure of 345kV Breakers CF and CM, and (2) "after the plant has stabilized" following a trip of the Turbine/Generator, Disconnects CI-A and CI-B must be open to preclude any possibility of motorizing the shutdown Generator when Output Breakers CF and CM are reclosed during restoration of the 345kV mat to its normal lineup. Therefore, the U.S. Nuclear Regulatory Commission (NRC) concludes that choice A, as annotated on the answer key, is the only correct answer, and that the question is considered acceptable as administered.

RO Question 6

During the implementation of 20.000.19, "Shutdown from Outside the Control Room" the following actions from Condition E are directed:

E.1 Position CMC switches on H21-P100 to match Control Room position.

E.2 Place the following in ON (H21-P100):

- C3500-M130, Div 2 DC Transfer switch.
- C3500-M131, BOP Transfer switch.
- C3500-M134, Swing Bus Transfer switch.
- C3500-M132, Div 1 DC Transfer switch.
- C3500-M133, Div 1 AC Transfer switch.

The reasons for performing these actions are:

A. (E.1) Any misalignment of these switches will prevent equipment operation.
(E.2) Transfers control power from the normal source to an alternate source, for remote service.

B. (E.1) Any misalignment of these switches will prevent equipment operation.
(E.2) Precludes simultaneous operation of the reactor plant from two locations.

C. (E.1) Any mispositioned CMC switches will either start or shutdown equipment unnecessarily.
(E.2) Transfers control power from the normal source to an alternate source, for remote service.

D. (E.1) Any mispositioned CMC switches will either start or shutdown equipment unnecessarily.
(E.2) Precludes simultaneous operation of the reactor plant from two locations.

Answer: D

***Reference(s) provided to NRC*:**

20.000.19, SHUTDOWN FROM OUTSIDE THE CONTROL ROOM (Rev. 43)

20.000.19 BASES (Rev. 1)

***Applicant Comment*:**

The position of the Transfer Switches does NOT prevent control from two locations, just the existence of the transfer switch does. In other words, the function of part (2) is performed by the switch, regardless of its position, so changing the switch POSITION does not preclude simultaneous operation from two locations. Moving the transfer switch, although not changing power supply does change the source of control power, because the flow of control power now routes it through the CMC switches on the RSD, rather than the CMC switches on the Control Room Panels.

***Facility Position on Applicant Comment*:**

The station staff supports the candidate's position.

The purpose of a Transfer Switch is to transfer control. The existence of the switch precludes simultaneous operation of the reactor plant from two locations. The stem was written with the intent of meeting the K/A which required "reasons for the following responses - Disabling control room controls" because of this the stem has the focus on the action vice the switch. The purpose of changing the switch position is to change where the component is controlled from.

While it was not the intent of the distractor, the word "source" can be interpreted to be which CMC switch is the "source" of the signals that is operating the equipment. The question did not specify power supply. Since operation of the transfer switch transfers control from the control room CMC to the RSD CMC, answer C is also correct.

The station staff recommends accepting both C and D as correct per the first bullet of ES-403, Paragraph D.1.b., which allows for post-exam changes for questions with an unclear stem that confused the applicants.

***NRC Evaluation/Resolution*:**

Recommendation not accepted. The question tests applicant knowledge of the reason for placing certain switches on the Remote Shutdown Panel (RSP) in predesignated positions in accordance with 20.000.19, "SHUTDOWN FROM OUTSIDE THE CONTROL ROOM," Condition E. Part 2 of the answer specifically tests applicant knowledge of the reason for placing the Transfer Switches at the RSP to the ON position as directed by Condition E, Subsequent Action Step E.2, which lists the Div 1 DC, Div 2 DC, BOP, Swing Bus, and Div 1 AC Transfer Switches.

The applicant contends: (1) that the position of the RSP Transfer Switches does NOT prevent control from two locations, (2) that changing RSP Transfer Switch position does not preclude simultaneous operation from two locations, and (3) that changing RSP Transfer Switch position does change the source of control power, because the flow of control power now is re-routed through the CMC switches on the RSD, rather than the CMC switches on the Control Room Panels.

The applicant's contention is directly refuted by the Bases statement for Subsequent Action E.2 in the 20.000.19 Bases document, which states:

> *"The requirement for a transfer switch exists to preclude simultaneous operation of the reactor plant from two locations. These switches must be in the ON position for operation of equipment from the Remote Shutdown Panel. Power supplies for the Remote Shutdown System equipment and components shall be consistent with that used by the interfacing system. Control power shall be the normal power serving this equipment or components.*

> *The Remote Shutdown System provides an alternate means for control of equipment and not an alternate means for supplying power to it. As a result, transfer of control power from the normal source to an alternate source is not provided."*

Separately, the Facility, in their "Position on Applicant Comment," response, states: *"The purpose of a Transfer Switch is to transfer control. The existence of the switch precludes simultaneous operation of the reactor plant from two locations."* In addition, the NRC, through independent review of the electrical schematic diagram for Control Rod Drive (CRD) Water Pump A (Detroit Edison Drawing Number 61721-2111-01), a load for the BOP Transfer Switch, was able to confirm that the associated RSP Transfer Switch does preclude simultaneous operation of the CRD pump from two locations and that the transfer of control power from the normal source to an alternate source is not provided.

Based on the above, the NRC concludes that choice D, as annotated on the answer key, is the only correct answer, and that the question is considered acceptable as administered.

RO Question 12

RCIC is in operation per 23.206, "Reactor Core Isolation Cooling System" Section 7.6, "RPV Pressure Control Using RCIC"

3D168, REACTOR PRESSURE HIGH, subsequently alarms for the first time during this event.

Reactor Pressure is rising slowly at approximately 0.1 psi per minute.

RCIC __(1)__ used for RPV Pressure Control __(2)__.

A. (1) can continue to be
(2) by maintaining RCIC discharge pressure approximately 100 psi above reactor pressure by adjusting E41-F011, HPCI/RCIC Test Iso/PCV.

B. (1) can continue to be
(2) by maintaining RCIC discharge pressure approximately 100 psi below reactor pressure by adjusting E41-F011, HPCI/RCIC Test Iso/PCV.

C. (1) is not being
(2) because the RCIC system is tripped and the trip must be reset prior to use.

D. (1) is not being
(2) because Low-Low Set is now controlling pressure.

Answer: A

Reference(s) provided to NRC:

23.206, REACTOR CORE ISOLATION COOLING SYSTEM (Rev. 3)

ARP 3D168, REACTOR PRESSURE HIGH (Rev. 12)

Applicant Comment:

Part (2) of Distractor B: Even though the procedure states maintain RCIC discharge pressure 100 psi above reactor pressure, RPV pressure control would still be possible even with RCIC discharge pressure 100 psi below reactor pressure. The candidate contends that steam would still be drawn from the RPV, to turn the RCIC turbine, so RPV pressure control would still be possible. The candidates contend that knowledge of the specific pressure range for using RCIC for RPV pressure control does not necessarily preclude it from controlling RPV pressure.

Facility Position on Applicant Comment:

The station staff supports the candidate’s position.

The flow path is through the F022 MOV isolation and F011 air operated pressure/flow control valve. F011, HPCI/RCIC Test Return to CST Isolation Valve, is a Drag (modified globe) valve with a custom trim built into the valve body to allow the appropriate differential pressure to be developed for RCIC (or HPCI) flow. During flow testing or pressure control, an opening pressure setpoint and valve position setpoint are dialed into the controller. When RCIC pump discharge pressure exceeds the opening pressure F011 will stroke open and control pressure at the setpoint. The valve will stroke at a rate of 0-100 % in 6 seconds this delay in opening keeps the pump discharge piping full to avoid water hammer. When pump discharge pressure drops below 75 psig F011 will auto close.

Because the stem of the question does not indicate the current setting of the E41-F011, HPCI/RCIC Test Iso/PCV, nor the current RCIC discharge pressure, Distractor B can be correct. If adjusting the setting of the E41-F011, HPCI/RCIC Test Iso/PCV causes turbine speed to increase, pressure reduction can occur, even if discharge pressure is approximately 100 psi below reactor pressure (This has been verified using the simulator model). Without a reference point, however it is unknown if the RCIC turbine will be speeding up or slowing down. The operator will adjust E41-F011 to change speed of RCIC Turbine to control reactor pressure.

The station staff recommends accepting both A and B. as correct per the third bullet of ES-403, Paragraph D.1.b., which allows for post-exam changes for questions with newly discovered technical information.

***NRC Evaluation/Resolution*:**

Recommendation not accepted. The question tests applicant knowledge of how to properly operate RCIC to control RPV pressure in accordance with 23.206, “REACTOR CORE ISOLATION COOLING SYSTEM,” Section 7.6, “RPV Pressure Control Using RCIC.” Part 2 of the answer specifically tests applicant knowledge of what value to procedurally maintain RCIC discharge pressure at relative to reactor pressure when operating RCIC in the pressure control mode (i.e., 100 psi above reactor pressure).

The applicant contends: (1) that even though the procedure states to maintain RCIC discharge pressure 100 psi above reactor pressure, RPV pressure control would still be possible even with RCIC discharge pressure 100 psi below reactor pressure, and (2) that knowledge of the specific pressure range for using RCIC for RPV pressure control does not necessarily preclude it from controlling RPV pressure. The Facility, in their “Position on Applicant Comment” response states: *“If adjusting the setting of the E41-F011, HPCI/RCIC Test Iso/PCV causes turbine speed to increase, pressure reduction can occur, even if discharge pressure is approximately 100 psi below reactor pressure (This has been verified using the simulator model).”*

While some degree of RPV pressure control may be effective under certain plant conditions with RCIC Pump discharge pressure 100 psi below reactor pressure (as noted by the Facility in at least one instance using the simulator model), no bases, reference, specific data, or evidence of any kind has been provided that would: (1) support the conclusion that this would be an acceptable operational practice under all plant conditions for which the established procedural guidance in 23.206, Section 7.6, was specifically intended, (2) warrant the operation of RCIC in a manner that would knowingly deviate from the established procedural guidance in 23.206, Section 7.6, for controlling reactor pressure (i.e., procedural non-compliance), (3) warrant a change to the established procedural guidance in 23.206, Section 7.6, or (4) substantiate prior observance of this operational practice at the actual plant.

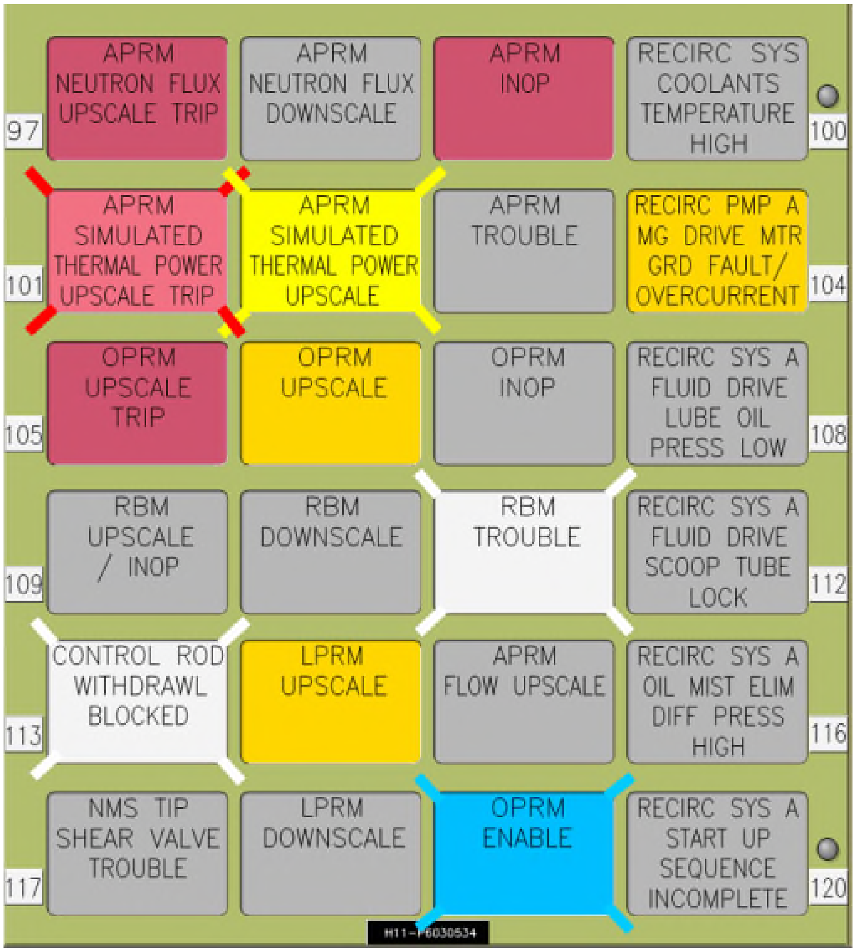
The question is not one of whether RPV pressure control using RCIC is possible at some reduced value of RCIC Pump discharge pressure, but rather what value of RCIC Pump discharge pressure, by procedure, will be maintained when using RCIC for RPV pressure control in accordance with 23.206. The stem specifically states: *"RCIC is in operation per 23.206, "Reactor Core Isolation Cooling System," Section 7.6, "RPV Pressure Control Using RCIC."* Procedure 23.206, Section 7.6, provides the following guidance:

> *Step 7.6.13* – *"Maintain RCIC discharge pressure approximately 100 psi above reactor pressure by adjusting E41-F011, HPCI/RCRC Test Iso/PCV, with the pulsar knob for E41-K820, Test Iso/PCV E41-F011 Ctrlr.*
>
> *Step 7.6.14* – *"*Adjust *E41-K820, Test Iso/PCV E41-F011 Ctrlr, to establish a discharge pressure between reactor pressure plus 100 psi and reactor pressure plus 150 psi as indicated on E51-R609, RCIC Pump Pressure Indicator."*

If the RCIC System discharge pressure were to be maintained as described in choice B, then operation would not be in accordance with the approved station procedure. Based on the above, the NRC concludes that choice A, as annotated on the answer key, is the only correct answer, and that the question is considered acceptable as administered.

RO Question 39

With the plant operating at 100% CTP, the following alarms were received on the H11-P603:



Monitoring the H11-P603 panel reveals:

- B31-R617, Recirc A Loop • Flow Indicator 39,153 GPM.
- B31-R613, Recirc B Loop Flow Indicator 39,156 GPM.
- B31-R614, Recirc Loops Flow Recorders Loop A 0.0 GPM, Loop B 39,156 GPM.

Which of the following annunciator actions has the highest priority now?

A. Bypass APRM #1 per 3D111, RBM Trouble.

B. Verify Reactor Scrams per 3D101, APRM Simulated Thermal Power Upscale Trip.

C. Verify the Reactor is not exhibiting Thermal Hydraulic Instability per 3D119, OPRM Enable.

D. Reduce reactor power by inserting in-sequence Control Rods per 3D102, APRM Simulated Thermal Power Upscale.

Answer: A

Reference(s) provided to NRC:

ODE – 10, OPERATIONS DEPARTMENT EXPECTATIONS (Rev. 26)

Applicant Comment:

Upon initial receipt of the indicated alarms, the first action performed by the P603 Operator is to evaluate “Power, Pressure and Level” in accordance with ODE–10, Operations Department Expectations, which states:

> *“Following an event (equipment trip or electrical bus loss which causes multiple annunciators simultaneously, or something that has caused RPV parameters to change) the P603 Operator evaluates (this needs to include an evaluation of Feedwater/Condensate to determine if the current trend is expected to continue) and reports to the CRS the status of reactor Power, Pressure and Level.”*

As the event caused multiple alarms, the initial response for the P603 operator is to evaluate power, pressure and level. With 3D119 in alarm, the APRM Operator Display Assembly automatically reverts to OPRM Bar Graph Display, allowing an operator the immediate ability to monitor for thermal hydraulic instabilities. This evaluation will be completed and reported to the CRS before further action, such as bypass of APRM #1 per 3D111 is commenced.

Therefore, the applicant believes there are two correct answers to this question, and that C is additionally correct.

Additionally, another applicant asked during the exam “Is the initial response (Power/Pressure/ Level) evaluation complete for the transient, and we’re now in the Alarm Response mode?” The facility responded “Do your best with the information provided.”

Facility Position on Applicant Comment:

The facility licensee disagrees with the candidates’ position because no additional technical justification can be provided to support a change to the question.

NRC Evaluation/Resolution:

Recommendation not accepted. The question tests the ability of the applicant to:
(1) evaluate Recirc Loop Flow indications and interpret the significance of alarming annunciators on Panel H11-P603, resulting from failure of the flow input to APRM #1 from Recirc Loop A, and (2) determine the highest priority annunciator actions.

The applicant cites the expectation from ODE – 10, “OPERATIONS DEPARTMENT EXPECTATIONS,” for evaluating and reporting the status of reactor power, pressure, and level following the receipt of multiple annunciators simultaneously. The applicant contends, based on this expectation, (a) that power, pressure, and level would be evaluated and reported to the

CRS before any further action, such as bypassing APRM #1 per ARP 3D111 (correct answer), is taken, and (b) that choice C, *"Verify the Reactor is not exhibiting Thermal Hydraulic Instability per 3D119, OPRM Enable,"* is therefore also a correct answer.
The applicant did not answer the question as written. The focus of the question was not to test applicant knowledge of Operations Department Expectations upon the receipt of multiple alarms that occur simultaneously. The stem included an embedded graphic of annunciators on Panel H11-P603, showing 5 of the 24 annunciators in alarm, along with pertinent information consisting of 4 individual Recirc Loop Flow readings (i.e., three that were consistent with 100 percentpower operation and one that read 0.0 GPM) obtained from 3 separate panel instruments (B31-R617, B31-R613, B31-R614); information that was essential to answering the question correctly. The question statement, "*Which of the following annunciator actions has the highest priority now?",* required the applicant to evaluate and assess all of the information contained in the stem, not just the embedded annunciator graphic. Operators are trained and expected to use multiple indications when verifying plant status. To that point, Recirc Loop Flows are a key parameter during the initial assessment of Power, Pressure, and Level (PPL), given that Recirc Flow directly impacts reactor power in a BWR. The applicant's contention that choice C is correct, based on application of the ODE-10 expectation cited above, would mean (1) that the only information necessary to correctly answer this question is mere recognition of the fact that OPRM Enable Annunciator 3D119 is lit on the embedded annunciator graphic, and (2) that the Recirc Loop Flow indications were superfluous information within the context of the question. Had the applicant properly evaluated the Recirc Loop Flow information provided in the stem, it would have been apparent that an actual low flow condition did not exist for the given power level (Reactor Power remained unaffected) and that conditions leading to Thermal Hydraulic Instabilities were not present in the core. The Facility disagrees with the applicant's position on the basis that insufficient justification exists to support the requested change.

Based on the above, the NRC concludes that choice A, as annotated on the answer key, is the only correct answer, and that the question is considered acceptable as administered.

RO Question 70

The following conditions exist:

- ALL RPV Head Closure • Bolts are FULLY TENSIONED.
- Reactor Coolant System Temperature is 185°F.
- The Reactor Mode Switch is in REFUEL.
- LCO 3.10.4, Single Control Rod Withdrawn - Cold Shutdown, does NOT apply.

Based on these conditions, which ONE of the following is the correct MODE of operation per Technical Specifications?

A. MODE 2, Startup.
B. MODE 3, Hot Shutdown.
C. MODE 4, Cold Shutdown.
D. MODE 5, Refuel.

Answer: A

Reference(s) provided to NRC:

No reference provided.

Applicant Comment:

This question is at the wrong license level and should not have been included on the RO Section of the exam.

Facility Position on Applicant Comment:

The facility licensee disagrees with the candidates' position because no additional technical justification can be provided to support a change to the question.

NRC Evaluation/Resolution:

Recommendation not accepted. The question tests applicant knowledge of the Mode Table (Table 1.1-1) in Section 1.1, "Definitions, of the Technical Specifications (TS). The applicant contends that the question is not RO level knowledge. The question requires the applicant to recall from memory that with the Mode Switch in the "Refuel" position and all RPV Head Bolts fully tensioned, that the reactor is considered to be in Mode 2 (Startup), regardless of Reactor Coolant System (RCS) Temperature.

NUREG-1021, "Operator Licensing Examination Standards for Power Reactors" (Rev. 11), Section ES-401, "Preparing Initial Site-Specific Written Examinations," Attachment 2, Clarification Guidance for SRO-Only Questions," Page 20 of 52, states:

> *"SRO-only knowledge generally cannot be claimed for questions that can be answered solely based on expected RO TS knowledge. ROs are typically expected to know the LCO statements and associated applicability information (i.e., information above the double line separating the ACTIONS from the LCO and associated applicability statements."*

ROs are expected to know "applicability" information. The *"above the double line"* Applicability Statements reference the applicable Mode(s) specified and defined in Mode Table 1.1-1 of the TSs. ROs are therefore, by extension, expected to know the information contained in TS Table 1.1-1.

Separately, NUREG-1021, Sections ES-403, "Grading Initial Site-Specific Written Examinations," and ES-501, "Initial Post-Examination Activities," provide the following guidance pertaining to the submittal of reference material for contested questions. No references were provided for RO Question 70.

> *ES-403, Paragraph D.1.a, Page 2 of 6, states: "Do not recommend deleting any question or changing any answer unless a valid reference supports the change."*

> *ES-501, Paragraph D.2.b, Page 4 of 34, states: "The chief examiner will not accept a change to the examination unless the facility licensee or license applicant submits a valid reference to support its recommendation."*

Based on the above, the NRC concludes that choice A, as annotated on the answer key, is the only correct answer, and that the question is considered acceptable as administered.

SRO Question 81

While operating the reactor in MODE 1, a fission product release into the reactor coolant occurs. Current plant conditions are:

- Reactor is shutdown following the SCRAM.
- Reactor Pressure is 940 psig.
- RPV level in +100 inches.
- Main Steam Line C Inboard & Outboard MSIVs failed to close.
- The Main Turbine is tripped.
- Dose assessment indicates site boundary doses are 980 mrem (TEDE) and 500 mrem (Adult Thyroid) and rising.

(1) What action does the Emergency Operating Procedures require?
(2) What is the event classification?

A. (1) Emergency Depressurize the RPV.
(2) Alert.

B. (1) Emergency Depressurize the RPV.
(2) Site Area Emergency.

C. (1) Use the SRV's to commence a reactor cool down at less than 90°F/hr rate.
(2) Site Area Emergency.

D. (1) Use the SRV's to commence a reactor cool down at less than 90°F/hr rate.
(2) General Emergency.

Answer: B

<u>Reference(s) provided to NRC</u>:

ODE-10, OPERATIONS DEPARTMENT EXPECTATIONS (Rev. 26)

BWROG EPGs/SAGs, Appendix B (Rev. 3)

***<u>Applicant Comment</u>*:**

More information is required to get to the keyed answer. Site Boundary dose is 980 mrem (TEDE) and 500 mrem (adult thyroid) and rising. Sheet 5 Radioactive Release states "BEFORE release rate reaches the GE release rate" then go to RC-1 then ED." The stem states the reactor is shut down. However, the rate of rise is not specific and may not justify ED prior to corrective action being completed. With a very slow rate of rise, alternative actions may be pursued while the plant continues to commence a reactor cooldown of < 90°F/hr as required administratively. Therefore, ED is not REQUIRED until 999 mrem.

The definition of BEFORE in ODE-10 and the BWROG EPGs/SAGs, Appendix B states:

> *"BEFORE" means the step should be performed, if possible, in advance of the specified condition, but the timing of the action is event-dependent. No particular margin is specified or intended. If the condition has already occurred when the "BEFORE" step is reached, the action should still be performed, unless expressly prohibited.*

Based on this information, the applicant contends that B and C are both correct answers.

<u>Facility Position on Applicant Comment</u>:

The facility licensee disagrees with the candidates' position because no additional technical justification can be provided to support a change to the question.

<u>NRC Evaluation/Resolution</u>:

Recommendation not accepted. The question tests the ability of the applicant to:
(1) conservatively assess degraded facility conditions concurrent with an in-progress fission product release, resulting in site boundary radiation levels approaching General Emergency (GE) Emergency Action Level (EAL) thresholds as confirmed by dose assessment, and
(2) identify the appropriate course of action within the Emergency Operating Procedures (EOPs) to mitigate the severity of the event (i.e., Emergency Depressurization).

The applicant contends that the stem of the question contained insufficient information to support the determination to Emergency Depressurization (ED) the reactor prior to attempting corrective action. The applicant states: (1) the rate at which site boundary doses were rising was not specific, (2) with a very slow rate of rise, alternative actions may be pursued while a normal cooldown is commenced (less than 90 degrees Fahrenheit/hour), and (3) ED is not required until 999 mrem (GE dose threshold at the site-boundary is >1000 mrem). The applicant also cites the definition of "BEFORE" from ODE-10 and the BWROG EPGs/SAGs, relative to its use and application in EOP 29.100.01 (SH 5), "SECONDARY CONTAINMENT AND RAD RELEASE," Step RR-4, regarding the determination of when ED is required.

The Discussion section associated with Radioactivity Release Control Step RR-2 in the BWROG EPGs/SAGs, Appendix B states:

> *"An offsite radioactivity release rate above the General Emergency action level represents a substantial increase in the severity of the offsite radioactivity release, relative to the entry condition, and accordingly presents a more immediate threat to the continued health and safety of the public. Before the release rate reaches the General Emergency level, emergency RPV depressurization is performed to reduce the radioactivity release rate."*

The stem contains information that TEDE dose assessment at the site boundary was 980 mrem and "rising" (emphasis added). This value is almost 10 times greater than the Site Area Emergency (SAE) EAL threshold value of 100 mrem and is only 20 mrem shy of exceeding the GE EAL threshold value of 1000 mrem. The dose assessment provides positive indication that GROSS fuel failure exists and that the health and safety of the General Public is threatened. The stem also includes information that Main Steam Line (MSL) C Inboard and Outboard MSIVs failed to close, resulting in the inability to isolate containment with the reactor at pressure (940 psig) concurrent with GROSS fuel failure. The stem, as written, provides sufficient information to reasonably conclude that GE declaration is imminent unless conservative/timely action is taken to Emergency Depressurize the reactor before the GE threshold limit is exceeded. NUREG-1021, ES-403, "GRADING INITIAL SITE-SPECIFIC WRITTEN EXAMINATIONS," Section D.1.a, states *"An unreasonable assumption on the part of an applicant does not justify the acceptance of an alternative answer."* Accordingly, the use of SRV's to commence a "normal cooldown" (emphasis added) at some value less than 90 degrees Fahrenheit/hour (choice C) under the stated conditions is both non-conservative and incorrect.

The definition of "BEFORE" previously cited by the applicant, states: *"BEFORE means the step should be performed, if possible, in advance of the specified condition, but the timing of the action is event-dependent."* For reasons identified above, the event depicted by conditions provided in the stem (i.e., in progress fission product release, MSL failure to isolate, and dose readings at 980 mrem and rising) meets the intent of the "BEFORE" definition with respect to *"event-dependent,"* therefore, Emergency Depressurizing the reactor prior to exceeding the GE threshold, is completely justified, without the need for "rate of rise" information. NUREG-1021, Appendix E, "POLICIES AND GUIDELINES FOR TAKING NRC EXAMINATIONS," Section B.7, states *"When answering a question, do not make assumptions regarding conditions that are not specified in the question unless they occur as a consequence of other conditions that are stated in the question."*

Based on the above, the NRC concludes that choice B, as annotated on the answer key, is the only correct answer, and that the question is considered acceptable as administered.

SRO Question 83

You are the Control Room Supervisor (CRS).

A GROSS fuel failure occurred and the Mode Switch has been taken to shutdown. The following plant conditions currently exist:

- All Immediate operator actions have been taken.
- Post Scram Feedwater Logic actuated on the Scram.
- Reactor Pressure has been lowered to 900 psig.
- 3D148, FW/MTG RPV H2O Level 8 Trip is in alarm.
- 20.000.23, High RPV Water Level has been entered.
- RPV water level is 225" and rising at 0.8"/min.

One minute later, B21-R605, RPV Floodup Level Indicator, becomes unavailable.

In accordance with AOP 20.000.23, High RPV Water Level, which of the following level control actions will you direct and why?

A. Close the Inboard MSIVs to prevent flooding the Main Steam Lines.

B. Establish RWCU Blowdown to restore RPV Water Level to a band directed by the SM.

C. Reset Reactor Scram IAW 23.610, RPS System SOP, to minimize input from the CRD system.

D. Close C1100-F034, CRD Charging Wtr Header Iso Vlv, to minimize input from the CRD system.

Answer: D

***Reference(s) provided to NRC*:**

20.000.23, HIGH RPV WATER LEVEL (Rev. 5)

***Applicant (Multiple) Comment*:**

The question specifically asks what actions would be directed per 20.000.23 High RPV Level. Condition E specifies Reset SCRAM if possible or Close C1100F034. The CRS would order the SCRAM to be reset, based on this step. There are no notes or cautions in 20.000.23 that prohibit the CRS from directing the SCRAM to be reset. Once the NO (non-licensed operator) entered 23.610 (procedure to reset the SCRAM) the P&L would be addressed then the direction would change to close the C1100-F034. The Pre-req says not to reset SCRAM during a fuel failure, however, the order would be based on the AOP and the P&L's would be addressed, after directed, by the RO. Therefore, there is nothing explicitly prohibiting the CRS from directing the scram to be RESET and Answers C and D should be correct.

<u>Facility Position on Applicants' Comment</u>:

The facility licensee disagrees with the candidates' position because no additional technical justification can be provided to support a change to the question.

<u>NRC Evaluation/Resolution</u>:

Recommendation not accepted. The question tests the ability of the applicant to determine the appropriate course of action within 20.000.23, HIGH RPV WATER LEVEL, to control rising RPV water level coincident with GROSS fuel failure.

The applicants contend that *"The question specifically asks what actions would be directed per 20.000.23 High RPV Level,"* and that the Control Room Supervisor (CRS) would order the scram to be reset in accordance with 20.000.23, Condition E, to control RPV water level, given that there are no notes or cautions in 20.000.23 that explicitly prohibit the CRS from directing this action. The applicants state that the Prerequisites section in the procedure to reset the scram (23.610, "REACTOR PROTECTION SYSTEM") would address the reset under fuel failure conditions, at which point the direction would be to close C1100-F034, "CRD Charging Water Header Isolation Valve," to minimize input from the CRD system, instead of resetting the scram.

The applicants' verbatim statement of contention (i.e., *"The question specifically asks what actions would be directed per 20.000.23 High RPV Level"),* omits the fact that *"GROSS fuel failure"* is specified up front, in the stem of the question. AOP 20.000.23, Condition E, Subsequent Action Step E.2 states:

> *"Reset scram if possible (23.610)*
>
> ***OR***
>
> *Close C1100-F034."*

All CRSs are expected to have knowledge of the prerequisites listed in 23.610, Section 6.2, for resetting a scram, one of which states: *"No fuel damage is suspected."* As noted above, the first part of Subsequent Action Step E.2 specifically states *"Reset scram if possible (23.610)."* Resetting the scram is a prohibited action in this question because of the GROSS fuel failure. It is imperative that the CRS understands plant conditions and the consequences associated with event driven mitigative strategies. CRS directed actions in all events must be carefully considered and deliberately applied to plant conditions. MOP01, "CONDUCT OF OPERAIONS," Section 3.1, "Roles and Responsibilities," Paragraph 3.1.4 and Subparagraph 3.1.4.2 state:

> *3.1.4 – "The CRS shall have responsibility and delegated authority for directing all plant operating activities. Specific responsibility and authority of the CRS shall include, but is not limited to, the following:*
>
> *2. Responsibility to always maintain the broadest perspective of operational conditions affecting the safety of the plant as a matter of highest priority."*

For this question, the applicant is informed that a GROSS fuel failure has occurred and that RPV level is 225" and rising at 0.8 inch/minute. A capable CRS, knowing that a GROSS fuel failure exists, is expected, at Step E.2 of AOP 20.000.23, to direct the closure of C1100-F034 in lieu of resetting the scram due to the undesirable radiological consequences of draining highly radioactive Scram Discharge Volume (SDV) inventory to the Torus Sump (i.e., elevated radiation levels in the Reactor Building and the spread of contamination). Contrary to the applicants' contention above, *as it pertains to the performance of AOP 20.000.23, Step E.2*, the CRS is expected to know that scram reset is prohibited under GROSS fuel failure conditions, and therefore, should not be blindly directed in this situation. The restriction associated with resetting a scram when fuel damage is suspected, is specifically addressed in Operations Training Document ST-OP-315-0027-001, "Reactor Protection System," under the System Operational Description for "Abnormal and Infrequent Operation."

Based on the above, the NRC concludes that choice D, as annotated on the answer key, is the only correct answer, and that the question is considered acceptable as administered.

SRO Question 84

During a reactor startup with power at 23% a rod drop accident causes a power spike and has resulted in the following:

- Reactor Pressure ……………………...…………….….…………… 960 psig
- Reactor Level ……………………......……………...………………… 197 inches
- Reactor power ……..……………….……………..…………………..…… 28%
- MCPR ……...………………………..……………..…...…………..…1.03
- MFLCPR ……..……….……….…..……..…..…………………….…..…1.38
- MAPRAT…...……………………….……..……………….……..……… 0.68
- MFLPD...…...…………………….....…..………………..…….…..……… 1.00

Which of the following action(s) is(are) required to be performed, within 2 hours, to satisfy Technical Specifications?

A. Restore all Thermal Limits to within limits.

B. Restore compliance with all Safety Limits ONLY.

C. Restore compliance with all Safety Limits AND Insert all insertable control rods.

D. Restore compliance with all Safety Limits AND Reduce THERMAL POWER to <25%.

Answer: C

Reference(s) provided to NRC:

TECHNICAL SPECIFICATIONS, 2.0 SAFETY LIMITS (Amendment No. 164)

TECHNICAL SPECIFICATIONS, 3.2 POWER DISTRIBUTION LIMITS, 3.2.2 MINIMUM CRITICAL POWER RATIO (MCPR) (Amendment No. 134)

***Applicant 1 Comment*:**

A & C are both true statements because both Tech Specs apply and since the question does not ask for the most limiting spec.

***Applicant 2 Comment*:**

There are no complete actions since there are 3 required actions per Tech Specs: 1. Get within Thermal Limits, comply with Safety Limits, Insert all insertable control Rods. None of the 4 answers have ALL required actions.

Facility Position on Applicants' Comments:

The station staff supports Applicant 1's position.

The stem of the question includes BOTH an unmet Safety Limit (MCPR <1.08) and an unmet MCPR Power Distribution Limit (MFLCPR > 1.00). For the Safety Limit violation, Actions 2.2.1 Restore compliance with all SLs and 2.2.2 Insert all insertable control rods, shall be completed within 2 hours. LCO 3.2.2 also requires that any MCPRs not within limits be restored to within limits within 2 hours. Since the stem of the question did not specify actions to be taken for a violated Safety Limit, nor did it ask for the “most limiting” set of actions, Action A.1 of LCO 3.2.2 would also be required. During the exam review, several candidates expressed that, for Distractor A to be incorrect, it should have included the word ONLY at the end. Since it did not, it was impossible to discriminate between A and C.

During exam question writing, the primary author of this question failed to consider LCO 3.0.2, which states in part “Upon discovery of a failure to meet an LCO, the Required Actions of the associated Conditions shall be met, except as provided in LCO 3.0.5 and LCO 3.0.6.” This means that upon discovery of LCO 3.2.2 not being met, the Required Action of restoring MCPR(s) to within limits, within 2 hours, would also have been entered.

Therefore, the facility staff agrees with Applicant 1 that the question, as well as distractor A, do not provide all the necessary information to allow a competent operator to discriminate between A and C. Therefore, 2 correct answers should be accepted as per the first bullet of ES-403, Paragraph D.1.b., which allows for post-exam changes for questions with an unclear stem that confused the applicants.

The station staff recommends accepting both A and C as correct.

The station staff disagrees with Applicant 2’s position.

The facility licensee disagrees that there are no correct answers to this question. Use of the word ONLY at the end of Distractor B differentiates it from Distractor C adequately to allow a competent operator to differentiate between those 2 responses, unlike Distractor A, which could not.

NRC Evaluation/Resolution:

Recommendation not accepted. The question tests the ability of the applicant to evaluate the status of Reactor Core Safety Limits (SLs) and Power Distribution Limits following an inadvertent reactivity addition and determine the required actions to be performed within 2 hours to satisfy the Technical Specifications (TSs).

The first applicant contends that MCPR Power Distribution Limit TS 3.2.2 also has applicability, which makes choice “A” a second correct answer, given that the question does not ask for the most limiting TS actions. The second applicant contends that there are no correct answers because none of the choices include all of the required actions (i.e., restore the MCPR Thermal Limit, restore compliance with the MCPR Safety Limit, and insert all insertable control rods).

For this question, the applicant is required to determine that both the MCPR Safety Limit and the MCPR Power Distribution Limit are NOT met.

- TS 2.1, “Safety Limits,” Reactor Core SL 2.1.1.2, states: *“MCPR shall be ≥ 1.08 for two recirculation loop operation or ≥ 1.09 for single loop recirculation operation.”* The MCPR Safety Limit value provided in the stem is 1.03. Therefore, the MCPR Safety Limit is NOT met.

- TS 3.2, “Power Distribution Limits,” 3.2.2, “MCPR,” LCO 3.2.2, states: *“All MCPRs shall be greater than or equal to the MCPR operating limits specified in the COLR.”* LCO 3.2.2 is met when the MCPR Power Distribution Limit (MFLCPR) value calculated by the plant’s core monitoring/process computer is ≤ 1.00. The MFLCPR value provided in the stem is 1.38. Therefore, the MCPR Power Distribution Limit is NOT met.

Per TS 2.2, “SL Violations,” the following actions shall be completed within 2 hours when any SL is violated:

2.2.1 Restore compliance with all SLs: and
2.2.2 Insert all insertable control rods

Per TS 3.2.2, “MCPR,” the Required Action is to restore MCPR to within limits (i.e., MFLCPR ≤1.00) within 2 hours.

The NRC disagrees with both applicant contentions; (a) that there are two correct answers (A and C), and (b) that there are no correct answers. The question states *“Which of the following action(s) is(are) required to be performed, within 2 hours, to satisfy Technical Specifications?”* Choice A is incorrect because the actions to *“Restore all Thermal Limits to within limits,”* does NOT satisfy TSs. While actions taken to restore Thermal Limits would be expected to restore compliance with the MCPR SL, the required action to “*Insert all insertable control rods”* for violation of the SL would not be completed. Therefore, the applicant’s assertion that omission of the word “ONLY” in choice A made it impossible to discriminate between A and C, is without basis. Choice C is the only correct answer because the action to *“Restore compliance with all Safety Limits AND Insert all Insertable control rods”* within 2 hours, restores MCPR to within Power Distribution Limits, restores compliance with the MCPR SL, and completes the required action to *“Insert all insertable control rods,”* <u>all</u> of which are required to satisfy TSs.

Based on the above, the NRC concludes that choice C, as annotated on the answer key, is the only correct answer, and that the question is considered acceptable as administered.

SRO Question 92

The plant is in MODE 1 with the following conditions:

- Movement of irradiated fuel is in progress in the Spent Fuel Pool (SFP).
- An irradiated fuel bundle is currently suspended from the Refueling Bridge.
- Fuel Pool Cooling and Cleanup (FPCCU) system is in service on the East FPCCU Pump and the A Filter Demineralizer.
- The B (West) Pump and B Filter Demineralizer are both in standby.

A leak develops in the FPCCU system causing the following:
- Alarm 2D1, Fuel Pool Water Level Low, is received.
- Refuel Floor Coordinator reports that SFP level is 21' 6" and lowering.

(1) How will the FPCCU system be impacted?

(2) What action shall the CRS direct?

A. (1) West FPCCU Pump automatically starts.
(2) Immediately suspend ALL fuel movement.

B. (1) West FPCCU Pump automatically starts.
(2) Return the fuel bundle to its storage location and THEN suspend fuel movement.

C. (1) East FPCCU Pump will trip.
(2) Immediately suspend ALL fuel movement.

D. (1) East FPCCU Pump will trip.
(2) Return the fuel bundle to its storage location and THEN suspend fuel movement.

Answer: D

<u>Reference(s) provided to NRC</u>:

TECHNICAL SPECIFICATIONS, 3.7 PLANT SYSTEMS, 3.7.7 SPENT FUEL STORAGE POOL WATER LEVEL (Amendment No. 201)

TECHNICAL SPECIFICATIONS BASES, B 3.7 PLANT SYSTEMS, B 3.7.7 SPENT FUEL STORAGE POOL WATER LEVEL (Rev. 29)

***<u>Applicant (Multiple) Comment</u>*:**

Per Tech Spec 3.7.7, Action A.1, Suspend movement of irradiated fuel assemblies in the SFP, the stem has the Plant in Mode 1, so the CRS would not have immediate knowledge of exact fuel assembly status due to not having direct communications established as in MODE 5. The CRS would enter the Tech Spec A.1 to immediately suspend movement and then further evaluate additional actions related to final state of an irradiated fuel bundle.

Once determined, the fuel assembly would be inserted back to a safe location. Based on this, the candidates contend that both answers C and D are correct.

***Facility Position on Applicants' Comment*:**

The facility licensee disagrees with the candidates' position because no additional technical justification can be provided to support a change to the question.

***NRC Evaluation/Resolution*:**

Recommendation not accepted. The question tests the ability of the applicant to determine the impacts of Low Spent Pool Level on the Fuel Pool Cooling and Clean-Up System, and the required actions to be performed in accordance with plant Technical specifications (TSs) based on knowledge of the associated TS Bases.

The applicants contend that because the movement of irradiated fuel assemblies in the SFP is being performed in Mode 1, when the CRS would not have immediate knowledge of exact fuel assembly status (direct communications not established as in Mode 5), that the required action is the immediate suspension of irradiated fuel movement in accordance with TS 3.7.7, Condition A, Required Action A.1, with an allowance for further evaluation of additional actions to determine the final state of the irradiated fuel assembly prior to the completion of movement to place it in a safe position.

No basis or reference has been provided to support the applicants' contention that an allowance exists for further evaluation of irradiated fuel assembly status while the assembly remains suspended with Spent Fuel Pool Level continuing to lower. This position is not supported by the Facility. The fact that movement of irradiated fuel assemblies is being performed in Mode 1, when direct communications have not been established between the Control Room and the Refuel Floor, has absolutely no bearing on the required actions. The Applicability Statement for LCO 3.7.7 states: *"During movement of irradiated fuel assemblies in the spent fuel storage pool."* There are no Mode restrictions associated with LCO 3.7.7 Applicability. The movement of irradiated fuel in Modes 1, 2, or 3 is independent of reactor operations. The applicants' contention is directly refuted by the Bases description for TS 3.7.7, Required Action A1, which is stated below. Knowledge of the TS Bases are required to correctly answer this question.

> *"When the initial conditions for an accident cannot be met, action must be taken to preclude the accident from occurring. If the spent fuel storage pool level is less than required, the movement of irradiated fuel assemblies in the spent fuel storage pool is suspended immediately. Suspension of this activity shall not preclude completion of movement of an irradiated fuel assembly to a safe position. This effectively precludes a spent fuel handling accident from occurring."*

Based on the above, the NRC concludes that choice D, as annotated on the answer key, is the only correct answer, and that the question is considered acceptable as administered.

SRO Question 95

~~The plant is operating at 100%.~~ (Removed during exam administration with concurrence of chief examiner)

You are on shift as the Control Room Supervisor (CRS).

With control room manning in compliance with MOP19, Reactivity Management, which of the following evolutions would require you to position yourself in close proximity to the P603 Operator to maintain proper Reactivity Management oversight?

A. Reactor startup per the GOP until criticality is achieved.

B. Planned downpower to perform a rod pattern adjustment.

C. Reactor shutdown per the GOP until the plant reaches 25% power.

D. Reactor Recirculation flow adjustments required to maintain thermal power.

Answer: D

<u>Reference(s) provided to NRC</u>:

OPERATIONS CONDUCT MANUAL MOP19 – Reactivity Management (Rev. 26A)

***<u>Applicant 1 Comment</u>*:**

That MOP19 page 17, Step 5.8.1, CRS shall supervise the approach to criticality. Additionally, step 5.8.2 states Operators shall anticipate criticality anytime reactivity is added to the core. Therefore, during startup until criticality the CRS should maintain proper oversight per MOP19, making distractor A correct.

***<u>Applicant 2 Comment</u>*:**

That the downpower for the rod pattern adjustment (in distractor B) was not specified; If < 30% per MOP 19 page 5, step 4.3, a RMSRO will not be required. In the plant, reductions in power for Rod Pattern Adjustments have occurred anywhere from 10 to 25% power, not requiring the use of a RMSRO, therefore the CRS would provide direct oversight and therefore distractor B is correct.

<u>Facility Position on Applicants' Comments</u>:

The facility licensee disagrees with the candidates' position because no additional technical justification can be provided to support a change to the question.

NRC Evaluation/Resolution:

Recommendation not accepted. The question tests applicant knowledge of station Conduct of Operations procedures associated with Reactivity Management; specifically, the degree of on-shift reactivity oversight to be provided by the CRS position. The question requires the applicant to determine which evolution requires the CRS to be positioned in close proximity to the P603 Operator to maintain proper Reactivity Management oversight. MOP19-100, "REACTIVITY MANAGEMENT IMPLEMENTATION," Step 5.3.2, states:

5.3.2 The CRS shall maintain oversight of all reactivity manipulations in accordance with MOP19, "Reactivity Management":

- *For the small Reactor Recirculation flow adjustments required to maintain thermal power, the CRS shall be positioned in close proximity to the P603 Operator and Peer Checker so as to maintain this oversight role.*

- *For larger power maneuvers, a Reactivity Management SRO will be assigned to provide direct oversight of the manipulation of reactivity controls.*

Applicant 1 contends that choice A, *"Reactor startup per the GOP until criticality is achieved,"* is a correct answer because MOP19, "REACTIVITY MANAGEMENT," contains guidance that states (1) the *"CRS shall supervise the approach to criticality"* (Step 5.8.1), and (2) *"Operators shall anticipate criticality anytime reactivity is added to the core"* (Step 5.8.2).

A Reactivity Management Senior Reactor Operator (RMSRO), as noted above in MOP19-100, Step 5.3.2, will be assigned to provide direct oversight of the manipulation of reactivity controls for large power maneuvers. Large power maneuvers include "Reactor Startup" as delineated in MOP19, "REACTIVITY MANAGEMENT," Step 4.3 which states:

4.3 "Large reactivity changes requiring a Reactivity Management Senior Reactor Operator (RMSRO) are planned changes to Control Rod position or Recirculation pump speed. This role is normally fulfilled by the on-shift STA/IA with an active SRO license with the following exceptions:

- *Large drops of greater than 30% and restoration with major BOP equipment manipulations where the CRS cannot provide adequate Reactivity Management oversight.*

- *Reactor startup.*

- *Reactor shutdown per the GOP until the unit has reached Mode 3.*

- *As determined by the Operations Engineer."*

Accordingly, an SRO with an active license, *other than the on-shift STA/IA,* will fulfill the RMSRO role to provide direct oversight of reactivity manipulations during a Reactor Startup. Further, MOP19, Section 3.0, "Responsibilities," Steps 3.6 and 3.7 state the following:

3.6 "The CRS shall concur with, have direct authority over, and provide immediate oversight for all evolutions that could affect reactivity unless a Reactivity Management SRO (RMSRO) is assigned."

3.7 "The RMSRO shall concur with, have direct authority over, and provide immediate oversight for all evolutions that could affect reactivity when assigned."

Lastly, MOP19-100, Section 5.3, "On-Shift Reactivity Oversight," Step 5.3.4, provides guidance on how CRS and RMSRO responsibilities should be divided. Step 5.3.4 states:

5.3.4 "The CRS and RMSRO should divide responsibilities including alarm response as follows:

- *After directed by the CRS, the RMSRO should provide direct oversight of Reactivity manipulations contained in the GOPs and approved maneuvering plans.*
- *The CRS directs all other plant operations."*

For Reactor Startup, a RMSRO will be assigned and directed by the CRS to provide direct oversight of the manipulation of reactivity controls during the entire evolution, including the approach to, and determination of criticality. The question specifically asks what evolution would require the CRS to be positioned in close proximity to the P603 Operator to maintain proper Reactivity Management oversight. Procedurally, the responsibility to provide *"close proximity*" oversight of reactivity manipulations during the approach to, and determination of criticality, lies with the RMSRO, under the supervision of the CRS (emphasis added). There is no guidance in either MOP19 or MOP19-100 that provides insight as to whether MOP19, Step 5.8.1 *("The CRS shall supervise the approach to criticality"*), requires the CRS to be positioned in *"close proximity"* to the P603 Operator for maintenance of the Reactivity oversight role while the RMSRO is stationed, unlike steady-state power operations when the RMSRO is not stationed and "*small Reactor Recirculation flow adjustments are required to maintain thermal power"* (MOP19-100, Step 5.3.2). There is no transfer of responsibility for direct oversight of reactivity manipulations between the RMSRO and the CRS during the approach to, and determination of criticality. Certainly, there is nothing to preclude the CRS from assuming a position of *"close proximity"* to the P603 Operator during this time. The RMSRO, who will be in a position of "close proximity" to the P603 Operator, is under the direct supervision of the CRS. This would meet the intent of MOP19, Step 5.8.1. Even if specific guidance were to exist, requiring the CRS to be positioned in *"close proximity"* to the P603 Operator during the approach to, and determination of criticality, choice A would still be incorrect because the CRS would not physically assume said position for the length of time it takes from withdrawal of the first control rod until criticality is achieved. That responsibility/function is fulfilled by the RMSRO. No additional bases or reference, other than MOP19, has been provided by the applicant that would substantiate prior observance of the contended operational practice at the actual plant. The Facility disagrees with the applicant's position on the basis that insufficient justification exists to support the requested change.

Applicant 2 contends that choice B, *"Planned downpower to perform a rod pattern adjustment,"* is a correct answer because (1) the magnitude of the downpower was not specified, (2) MOP19, "REACTIVITY MANAGEMENT," Step 4.3, contains guidance that a RMSRO is not required when <30 percent power, and (3) power reductions in the plant ranging from 10 percent to 25 percent to support Rod Pattern Adjustments have occurred that did not require the use of a RMSRO; therefore, the CRS would provide direct oversight.

The applicant misinterpreted the guidance in MOP19, Step 4.3, regarding use of the on-shift STA/IA as the RMSRO. Step 4.3 provides exceptions to this practice, as stated below:

4.3 "Large reactivity changes requiring a Reactivity Management Senior Reactor Operator (RMSRO) are planned changes to Control Rod position or Recirculation pump speed. This role is normally fulfilled by the on-shift STA/IA with an active SRO license with the following exceptions:

- *Large drops of greater than 30% and restoration with major BOP equipment manipulations where the CRS cannot provide adequate Reactivity Management oversight.*
- *Reactor startup.*
- *Reactor shutdown per the GOP until the unit has reached Mode 3.*
- *As determined by the Operations Engineer."*

The applicant interpreted the first bulleted item to mean that a RMSRO is not required below 30% power, which is incorrect. The exceptions listed in Step 4.3 require an SRO with an active license, other than the on-shift STA/IA, to fulfill the role of RMSRO.

The applicant's contention that power reductions in the plant have occurred anywhere from 10 percent up to as high as 25 percent for Rod Pattern Adjustments, without requiring a RMSRO, conflicts with existing Facility guidance for Reactivity Management in MOP19, "REACTIVITY MANAGEMENT," and MOP19-100, "REACTIVITY MANAGEMENT IMPLEMENTATION. MOP19-100, Step 5.3.2, in part, states:

5.3.2 The CRS shall maintain oversight of all reactivity manipulations in accordance with MOP19, "Reactivity Management":

- *For the small Reactor Recirculation flow adjustments required to maintain thermal power, the CRS shall be positioned in close proximity to the P603 Operator and Peer Checker so as to maintain this oversight role.*
- *For larger power maneuvers, a Reactivity Management SRO will be assigned to provide direct oversight of the manipulation of reactivity controls.*

Power reductions ranging from 10 percent to 25 percent for Rod Pattern Adjustments are planned power maneuvers involving multiple coordinated activities, and as such, require Maneuver Plans in accordance with MOP19, Step 5.1.1. Planned downpowers within this range of magnitude are not minor reactivity adjustments based on the Reactivity Management guidance in MOP19 and MOP19-100, and therefore warrant additional reactivity oversight beyond what is specified for small Recirc Flow adjustments required to maintain thermal power (i.e., assignment of a RMSRO versus a CRS that is positioned in close proximity to the P603 Operator). MOP19-100 defines "significant" reactivity manipulations as >5 percent power change. No bases or reference documentation of any kind has been provided by the applicant that would substantiate prior observance of the contended operational practice at the actual plant. The Facility disagrees with the applicant's position on the basis that insufficient justification exists to support the requested change.

Based on the above, the NRC concludes that choice D, as annotated on the answer key, is the only correct answer, and that the question is considered acceptable as administered.

SIMULATOR FIDELITY REPORT

Facility Licensee: Fermi Power Plant, Unit 2

Facility Docket No: 050–341

Operating Tests Administered: June 17, 2019, through June 28, 2019

The following documents observations made by the U.S. Nuclear Regulatory Commission 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 Title 10 of the *Code of Federal Regulations*, Part 55.45(b). These observations do not affect U.S. Nuclear Regulatory Commission 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
None	N/A