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

August 13, 2020

Mr. Darrell Corbin
Vice President, Operations
Entergy Nuclear Operations, Inc.
Palisades Nuclear Plant
27780 Blue Star Memorial Highway
Covert, MI 49043–9530

SUBJECT: PALISADES NUCLEAR PLANT—NRC INITIAL LICENSE EXAMINATION REPORT 05000255/2020301

Dear Mr. Corbin:

On July 6, 2020, the U.S. Nuclear Regulatory Commission (NRC) completed the initial operator licensing examination process for license applicants employed at your Palisades Nuclear Plant. The enclosed report documents the results of those examinations. Preliminary observations noted during the examination process were discussed on June 30, 2020, with yourself and other members of your staff. An exit meeting was conducted by telephone on July 30, 2020, with Mr. Walter Nelson, Training Director, other members of your staff, and Mr. Bryan Bergeon, 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 plant's post-examination comments, received by the NRC on July 6, 2020, were discussed.

The NRC examiners administered an initial license examination operating test during the weeks of June 22, 2020, and June 29, 2020. The written examination was administered by Palisades Nuclear Plant's training department personnel on April 16, 2020. Eight Senior Reactor Operator and one Reactor Operator applicant were administered license examinations. The results of the examinations were finalized on July 29, 2020. Seven applicants passed all sections of their respective examinations. Six applicants were issued senior operator licenses and one applicant was issued an operator license. Two applicants failed one or more sections of the administered examination and were issued Preliminary Results Letters.

The administered written examination and 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 29, 2022. However, because two 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,

Patricia J. Pelke Digitally signed by Patricia J. Pelke Date: 2020.08.13 11:51:53 -05'00'

Patricia J. Pelke, Chief
Operations Branch
Division of Reactor Safety

Docket No. 50–255
License No. DPR–20

Enclosures:
1. OL Examination
   Report 05000255/2020301
2. Post-Examination Comments,
   Evaluation, and Resolutions
3. Simulator Fidelity Report

cc: Distribution via LISTSERV®
    W. Nelson, Training Director

Letter to Darrell Corbin from Patricia J. Pelke dated August 13, 2020.

SUBJECT: PALISADES NUCLEAR PLANT—NRC INITIAL LICENSE EXAMINATION REPORT 05000255/2020301

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REGION III

Docket No:	50–255
License No:	DPR-20
Report No:	05000255/2020301
Enterprise Identifier:	L-2020-OLL-0034
Licensee:	Entergy Nuclear Operations, Inc.
Facility:	Palisades Nuclear Plant
Location:	Covert, MI
Dates:	April 16, 2020, through July 6, 2020
Examiners:	B. Bergeon, Operations Engineer, Chief Examiner C. Zoia, Senior Operations Engineer, Examiner D. Reeser, Operations Engineer, Examiner
Approved by:	P. Pelke, Chief Operations Branch Division of Reactor Safety

SUMMARY

Examination Report 05000255/2020301; 04/16/2020–07/06/2020; Entergy Nuclear Operations, Inc., Palisades Nuclear Plant; 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

Seven of nine applicants passed all sections of their respective examinations. Six applicants were issued senior operator licenses and one applicant was issued an operator license. Two 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 March 9, 2020, with the assistance of members of the facility licensee's staff. During the onsite validation week, the examiners audited all nine license applications for accuracy. The facility licensee administered the written examination on April 16, 2020. 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 22, 2020, through June 29, 2020.

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 or equal to 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 30, 2022, (ADAMS Accession Numbers ML19213A180, ML19213A181, ML19213A184, and ML19213A183, respectively).

On July 6, 2020, the licensee submitted documentation noting that there were five post-examination comments for consideration by the NRC examiners when grading the written examination. The post-examination comments are documented in Enclosure 2 of this report.

The NRC examiners completed grading of the written examination on July 28, 2020, 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, extensive modifications were made to two job performance measures, minor modifications were made to several other job performance measures, and 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 30, 2022, (ADAMS Accession Numbers ML19213A180, ML19213A181, ML19213A184, and ML19213A183, respectively).

The NRC examiners completed grading of the operating test on July 29, 2020.

(3) Examination Results

Eight applicants at the Senior Reactor Operator level and one applicant at the Reactor Operator level were administered written examinations and operating tests.

Seven applicants passed all portions of their examinations. Seven applicants were issued their respective operating licenses on July 29, 2020. Two Senior Reactor Operator applicants 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 Title10 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

(1) During administration of the operating test, a facility training instructor signed onto the exam security agreement communicated the combination to the simulator door lock to another facility training instructor outside of the NRC exam security envelope without first validating that the facility training instructor was signed onto the exam security agreement. Another facility training instructor overheard the conversation and changed the combination to the simulator door lock before the other training instructor could enter the NRC exam security envelope. The training instructor who received the combination for the simulator door lock was subsequently determined to be on the exam security agreement, but this was not validated prior to providing the combination. A follow-up evaluation determined that no exam compromise occurred and therefore, no

replacement of examination material was warranted. This issue, which was of minor significance, was documented in Condition Report (CR) CR-PLP-2020-01975.

(2) During the administration of the operating test, a procedure was identified by an applicant as not being adequately erased on two separate occasions. In one instance, an applicant opened an Abnormal Operating Procedure and identified the first step was place-kept. The step had no Response Not Obtained and the place-keeping provided no pertinent information to the applicant, other than that the procedure had been performed. The applicant stopped, notified the Chief Examiner, and a clean procedure was provided to the applicant. Similarly, in the other instance, an applicant opened the Operating Requirements Manual (ORM) to validate equipment inoperability, and the conditions to be entered were place-kept. In this instance, the applicant had written on their rough log the inoperable equipment and the conditions to be entered prior to consulting the ORM. A follow-up evaluation determined that no exam compromise occurred and therefore, that no replacement of examination material was warranted. These issues, which were of minor significance, were documented in Condition Report CR-PLP-2020-01990.

4OA6 Management Meetings

.1 Debrief

The chief examiner presented the examination team's preliminary observations and findings on June 29, 2020, to Mr. Walter Nelson, Training Manager, and other members of the Palisades Nuclear Plant staff.

.2 Exit Meeting

The chief examiner conducted an exit meeting on July 30, 2020, with Mr. Walter Nelson, Training Manager, and other members of the Palisades Nuclear Plant 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. Security related information, as used in the written examination, was determined to be considered proprietary information and is to be withheld from public disclosure in accordance with Title10 of the *Code of Federal Regulations*, Part 2.390, "Public inspections, exemptions, requests for withholding."

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee

D. Corbin, Site Vice President
D. Lucy, General Manager Plant Operations
B. Baker, Senior Operations Manager
S. Moore, Manager Operations Support
J. Byrd, Operations Manager
P. Adams, Shift Manager
W. Nelson, Training Manager
F. Korfias, Training Superintendent
K. Robinson, Training Superintendent
D. Karnes, Operations Training Instructor
R. Rendler, Training Contractor
J. Hardy, Regulatory Assurance Manager
B. Dodson, Licensing Specialist

U.S. Nuclear Regulatory Commission

P. LaFlamme, Senior Resident Inspector
C. St. Peters, Resident Inspector
B. Bergeon, Operations Engineer, Chief Examiner
C. Zoia, Senior 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

NRC Resolution to the Palisades Nuclear Station Post-Examination Comments

RO Question 7

The plant has just been shut down from a 100 day run at full power.

- Component Cooling Water Temperature is 95ºF.
- E-60A, Shutdown Cooling (SDC) Heat Exchanger is unavailable.

Given these conditions, complete the following statements:

The SDC system _(1)_ meet its design basis.

With SDC in service and a loss of air to CV-3212, CV-3213, CV-3223, and CV-3224, SDC HX Isolation valves occurs, PCS flows through the SDC HXs _(2)_.

	(1)	(2)
A.	can	stops
B.	can	continues
C.	can NOT	stops
D.	can NOT	continues

Answer: D

Reference(s) provided to NRC:

- PL-SDC Shutdown Cooling Lesson Plan, Rev. 7
- Final Safety Analysis Report, Chapter 6, Engineered Safeguards Systems, Rev. 52

Applicant Comment:

Challenge to Question 7. There should be two correct answers. Both (B) and (D) are correct based on an ambiguous time reference and/or no reference to Primary Coolant System (PCS) temperature.

When reviewing the question, the first sentence states, “The plant has just been shutdown from a 100 day run at full power.” The initial conditions, specifically the word “just”, is ambiguous and allows interpretation of the frame of reference. For instance, if you have a 100-day frame of reference for a refueling outage, is it reasonable to discuss two days (48 hours) into the outage that the plant was just shut down.

Additionally, the PCS temperature is not provided in question 7. At Palisades, shutdown cooling (SDC) is only placed in service during modes 4, 5, and 6. These three modes are less than 300 degrees Fahrenheit PCS Tcold.

Based on the above, it is required to make assumptions on current plant parameters (i.e., Tcold and time after shutdown). Either parameter is vital to answer the first part of the question correctly. Without knowing the time after shutdown (specifically greater or less than 27.5 hours) or Tcold (greater or less than 130 degrees F) either answer (B) or (D) could be correct based on the assumption of the student.

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

The station does not agree with the candidate's assertion for the reason below:

Part 1 of this question asks if the system meets its design basis, which is described in FSAR 6.1 and PL-SDC as:

> *"The system is designed to cool the primary water from 300°F to refueling temperature with the low-pressure injection pumps and 90°F component cooling water. The maximum pressure of the primary coolant during this cooldown is 270 psia."*

The stem specifies that CCW inlet temperature is 95°F, which is above the design basis temperature of the SDC system. This means distractor B cannot be correct. The secondary part of this question is not being challenged.

The facility recommends that the grading of this question remains as approved.

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

The distractor/answer choices under discussion consist of the following:

<u>Distractor B</u>:

(1) can
(2) continues

<u>Answer D</u>:

(1) can NOT
(2) continues

The stem of the question explicitly provided the following key information to the applicant:

- *The plant has just been shutdown from a 100 day run at full power.*
- *Component Cooling Water Temperature is 95°F.*
- *E-60A, Shutdown Cooling (SDC) Heat Exchanger is unavailable.*

Part 1 of this question asks, given the initial conditions, if the SDC system meets its design basis. The design basis for the SDC system is described in the facility licensee's Final Safety Analysis Report (FSAR) Section 6.1 and the Shutdown Cooling system Lesson Plan PL-SDC as:

The system is designed to cool the primary system from 300°F to refueling temperature with the low-pressure injection pumps and 90°F component cooling water. The maximum pressure of the primary coolant during this cooldown is 270 psia.

The shutdown cooling heat exchangers are used to remove decay heat and sensible heat during Plant cooldowns and cold shutdowns. The units, operating together, are sized to hold the refueling temperature with the design component cooling water temperature of 90ºF.

Under the conditions provided in the question stem, the SDC system can NOT meet its design basis and Distractor (B) can NOT be correct, for two distinct reasons:

1) The CCW inlet temperature to the SDC Heat Exchanger (95ºF) exceeds that of the designed value in FSAR Section 6.1 and FSAR Table 6.4 (90ºF), and
2) Both SDC Heat Exchangers are required to be operating in order to meet the design basis, as discussed in FSAR Section 6.1.

The challenge to the question pertains to part 1 of the question, and whether assumptions of information not provided in the question stem are necessary to determine whether the SDC system can meet its design basis. With the information provided in the question stem, both the time from shutdown from full power and the PCS temperature are unnecessary to determine whether the SDC system can meet its design basis.

NUREG-1021, “Operator Licensing Examination Standards for Power Reactor” (Rev. 11), Appendix E, “Policies and Guidelines for Taking NRC Examination,” Subsection B, “Written Examination Guidelines,” Paragraph B.7, states, in part, *“If you have any questions concerning the intent or the initial conditions of a question, do not hesitate to ask them before answering the question.”* The applicant was briefed on the contents of APPENDIX E prior to exam administration, and all paragraph items contained in Subsection B were read “verbatim.” No questions associated with the adequacy of conditions, or any other aspect of Question 7, were raised by the applicant or any of the other applicants during administration of the exam.

Therefore, the NRC concluded that no change to the key for this exam question was required.

RO Question 63

The plant is in MODE 1 at 13% power, preparing for synchronization,

- Main Turbine is at 1800 RPM
- The Primary, Backup, and Coastdown Relays (386P, B, and C) have NOT been reset

Then,

- Actual main turbine speed reaches 1900 RPM
- Reactor Power rises to 16% by Nuclear Instrumentation

At the turbine controls in the Main Control Room, the operator observes the closure of ...

A. ONLY the Main Stop valves

B. ONLY the Governor and Intercept valves

C. ONLY the Governor and Reheat Stop valves

D. ALL the Main Stop, Governor, Intercept, and Reheat Stop valves

Answer: D

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

- ARP-1, Turbine Condenser and Feedwater Scheme, Rev 85
- Drawing E-17, Sheet 9, Logic Diagram Turbine-Generator Trips and Fast Transfer, Rev 25
- Drawing E-121, Sheet 1, Schematic Diagram, Turbine Control, Rev 43
- SOP-8, Main Turbine and Generating System, Rev 111
- PL-EHC, Main Turbine Control and Supervisory, Rev 8

***Applicant Comment*:**

No applicants challenged Question 63.

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

The facility licensee discovered new technical information proposing the question is technically incorrect and has no correct answers.

The drawing [E-17 sheet 9] depicts the contacts in the shelf state (de-energized) position, however this circuit is an energized to actuate and trip the turbine vice a safety circuit that would normally de-energize to cause the trip. SOP-8, Main Turbine and Generating System, section 7.1, K-1 Turbine Generator, step 7.1.1.z directs that relays 386P, Gen Direct Trip Lockout Relay (primary), 386B, Gen Direct Trip Lockout Relay (backup), and 386C, Generator Indirect Trip Lockout Relay, be reset. Without resetting these relays, DEH will continue to dump and will not be able to actuate the Main Stop, Governor, Intercept, and Reheat Stop valves.

Since the stem indicates that the Main Turbine is operating at 1800 RPM, these relays MUST be reset, therefore the conditions of the stem are technically incorrect. The facility recommends throwing out the question as it is technically incorrect and therefore has no correct answers.

***NRC Evaluation/Resolution*:**

The initial conditions provided in the question stem indicate that the Main Turbine is operating at 1800 RPM with the Primary, Backup, and Coastdown Relays (386P, B, and C) NOT reset.

Newly discovered technical information showed that the 386P, B, and C relays must, in fact, be reset in order to even latch the turbine. SOP-8, Main Turbine and Generating System, section 7.1, K-1 Turbine Generator, step 7.1.1.z directs that relays 386P, Gen Direct Trip Lockout Relay (primary), 386B, Gen Direct Trip Lockout Relay (backup), and 386C, Generator Indirect Trip Lockout Relay, be reset prior to latching the turbine.

This was further shown in the facility licensee's drawings, E-17 sheet 9 and E-121 Sheet 1, depict the 386P, B, and C contacts in the shelf state (de-energized) position, however this circuit is an energized to actuate (and trip the turbine) vice a safety circuit that would normally de-energize to cause the trip. Without resetting these 386P, B, and C relays, the Solenoid Trip 20/AST relay remains energized and Electrohydraulic Control (EHC) fluid will continually dump to the EHC reservoir, and will not be able to actuate the Main Stop, Governor, Intercept, and Reheat Stop valves. The Main Stop, Governor, Intercept, and Reheat Stop valves are all hydraulic to open valves. With EHC continually dumping to the reservoir, these valves would not be initially opened. Therefore, the initial conditions could not be met [the valves in question could not be opened, the turbine could not be latched, and the turbine speed could not be accelerated to 1800 rpm] with the 386P, B, and C relays not reset. Therefore, the question is not operationally valid and there is no correct answer, since the turbine cannot be latched (much less reach 1900 RPM, as provided in the question stem) without first resetting the 386P, B, and C relays.

Therefore, the NRC concluded that no correct answer existed, and the question should be deleted from the administered examination.

RO Question 74

Question 74 withheld from public disclosure due to security related content.

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

- EN-TQ-113, Initial License Operator Training Program, Rev 17

***Applicant Comment*:**

Challenge to Question 74.

This topic was not trained on prior to the exam and the procedures were not made available to candidates until the training was conducted (security sensitive procedures not for public disclosure).

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

The station supports the applicant's assertion for the reason below:

While the K/A is applicable to Palisades, Fleet Procedure EN-TQ-113, Initial License Operator Training Program, places topics such as Extreme Damage Mitigation Guidelines and B.5.b in the Post-NRC Exam transition training portion of the training program. Candidate performance on this question was 100% failure as the topic and AOPs have yet to be presented to the class.

The facility recommends throwing out this question as the subject matter is directed for training in the post-NRC exam portion of the training program.

***NRC Evaluation/Resolution*:**

The applicant, with support of the facility licensee, contends that due to the facility licensee's procedure, EN-TQ-113 "Initial License Operator Training Program," not requiring that the material be taught until the post-ILE (Initial License Exam) portion of the training program, the question should be deleted from the administered examination.

While the facility licensee's procedure EN-TQ-113 Attachment 9.1 Section IX may allow for presentation of the various topics (B.5.b, EDGMs, FLEX, etc.) the "Post-NRC Transition Training" phase of the training program, failing to cover/train on a specific topic or objective within the training program is not bounds for dismissing the topic or K/A. Per NUREG-1021 ES-401 D.1.

> *The fact that a K/A does have a learning corresponding facility training objective, was not covered in training, or is subject to selection in multiple tiers are not sufficient bases for eliminating the K/A from any tier of the outline.*

The given K/A was accepted by both the facility licensee and the NRC, an acceptable question was developed, validated, and approved by both the facility licensee and the NRC, failing to cover the topic within the bounds of the training program is not subject for dismissal of the question post-exam administration.

Additionally, the facility licensee has the following training program learning objective that was associated with the question:

> *Given a Security threat, implement the required actions in accordance with AOP-44, "Response to Attack on Palisades."*

NUREG-1021, "Operator Licensing Examination Standards for Power Reactor" (Rev. 11), Appendix E, "Policies and Guidelines for Taking NRC Examination," Subsection B, "Written Examination Guidelines," Paragraph B.7, states, in part, *"If you have any questions concerning the intent or the initial conditions of a question, do not hesitate to ask them before answering the question."* The applicant was briefed on the contents of APPENDIX E prior to exam administration, and all paragraph items contained in Subsection B were read "verbatim." No questions associated with the adequacy of conditions, or any other aspect of Question 74, were raised by the applicant or any of the other applicants during administration of the exam.

The question is a valid question as-administered, as discussed in NUREG 1021 ES-401 D.1, and the existence of a facility licensee training program learning objective directly supporting the K/A. Therefore, the NRC concluded that no change to the key for this exam question was required.

SRO Question 84

SRO ONLY:

The plant is in MODE 5 for a maintenance outage.

An UNPLANNED ENTRY into a Higher Risk Plant Operating States (HRPOS) is required.

Given these conditions, complete the following statements:

Any Hot work in progress __(1)__.

LCO 3.0.9, which addresses situations where required barriers are unable to perform their related support functions and provides instructions and conditions for meeting the supported system LCOs __(2)__ applicable to Fire Barriers.

	(1)	(2)
A.	must be stopped	is
B.	must be stopped	is NOT
C.	can continue	is
D.	can continue	is NOT

Answer: C

Reference(s) provided to NRC:

- LCO 3.0.9, Limiting Condition For Operation (LCO) Applicability, Amendment 252
- BLCO 3.0.9 Bases, Limiting Condition For Operation (LCO) Applicability Bases, Revised 07/26/2017
- Admin 4.49, Non Power Operation Fire Risk Management, Rev. 0

Applicant Comment:

The question does not specify a barrier in question. Several barriers in the plant are both fire and flood or HELB doors. These barrier's fire functions are not covered by LCO 3.0.9, but 3.0.9 does still apply to them for other functions. Question seems too non-specific to blatantly state 3.0.9 does not apply. Recommend accepting answers a and b.

Facility Position on Applicant Comment:

The station does not support the applicant's assertion for the reason below:

This question requires knowledge of the applicability of LCO 3.0.9 to Fire Doors. The site team reviewed LCO 3.0.9 bases, which states:

This provision does not apply to barriers which support ventilation systems or to fire barriers. The Technical Specifications for ventilation systems provide specific Conditions for inoperable barriers. Fire barriers are addressed by other regulatory requirements and associated plant programs.

The question stem was determined to be correct as written and requires no further information to answer the question posed.

The facility recommends no change to the grading of this question.

***NRC Evaluation/Resolution*:**

The distractor/answer choices under discussion consist of the following:

Distractor A:
(1) must be stopped
(2) is

Answer B:
(1) must be stopped
(2) is NOT

Part 2 of the stem of the question explicitly asks the applicant:
LCO 3.0.9, which addresses situations where required barriers are unable to perform their related support functions and provides instructions and conditions for meeting the supported system LCOs ___(2)___ applicable to Fire Barriers.

The applicant contended that the question did not specify a specific barrier in question, and while several barriers ("fire and flood or HELB doors") are not covered by Technical Specification LCO 3.0.9, but the LCO still applies to those barriers for other functions.

NUREG-1021, "Operator Licensing Examination Standards for Power Reactor" (Rev. 11), Appendix E, "Policies and Guidelines for Taking NRC Examination," Subsection B, "Written Examination Guidelines," Paragraph B.7, states, in part, *"If you have any questions concerning the intent or the initial conditions of a question, do not hesitate to ask them before answering the question."* The applicant was briefed on the contents of APPENDIX E prior to exam administration, and all paragraph items contained in Subsection B were read "verbatim."
No questions associated with the adequacy of conditions, or any other aspect of Question 84, were raised by the applicant or any of the other applicants during administration of the exam.

The LCO 3.0.9 Bases specifically states that the provisions of the LCO do not apply to Fire Barriers. Per LCO 3.0.9 Bases:

Barriers are doors, walls, floor plugs, curbs, hatches, installed structures or components, or other devices, not explicitly described in Technical Specifications, that support the performance of the safety function of systems described in Technical Specifications.

This provision does not apply to barriers which support ventilation systems or to fire barriers. The Technical Specifications for ventilation systems provide specific Conditions for inoperable barriers. Fire barriers are addressed by other regulatory requirements and associated plant programs.

While a specific barrier is not provided, as the applicant contended, providing a specific barrier is not necessary to answer part 2 of the question. Part 2 of the question only asked whether LCO 3.0.9 was applicable to Fire Barriers. According to the LCO 3.0.9 Bases, as referenced, LCO 3.0.9 does not apply to fire barriers and fire barriers are addressed by other regulatory requirements and associated plant programs. Therefore, Distractor A cannot be considered a correct answer, because part 2 of the distractor was not correct. Therefore, the NRC concluded that no change to the key for this exam question was required.

SRO Question 89

SRO ONLY:

The plant is in MODE 1, with the instrumentation issues listed below:

Instrument	Issue
LI-0702, ‘A’ SG Narrow Range Level	failed low
LI-0757B, ‘A’ SG Wide Range Level	failed high
FI-0727A, Aux Feedwater Flow to ‘B’ SG	failed high
LI-0752D, ‘B’ SG Safety Channel	failed low

Given these conditions:

The number of INOPERABLE Post Accident Monitoring (PAM) Instrument(s) is __(1)__.

A. one

B. two

C. three

D. four

Answer: A

Reference(s) provided to NRC:

- LCO 3.3.7, Post Accident Monitoring (PAM) Instrumentation, Amendment 221
- Regulatory Guide 1.97, Instrumentation For Light-Water Cooled Nuclear Power Plants To Assess Plant And Environs Conditions During And Following An Accident, Rev. 3
- Operating Requirement Manual (ORM) Table 3.17.6, Rev. 15
- Final Safety Analysis Report, Appendix 7C, Rev. 34
- EN-OP-129, Operations Equipment Labeling

Applicant Comment:

Challenge to Question 89. The correct answer should be (B), two inoperable instruments based on the stem of the question.

When answering question 89, both FI-0727A, Aux Feedwater Flow to Steam Gen E-50B, and LI-0757B, ‘A’ Steam Generator Wide Range Level, Post Accident Monitoring (PAM) Instruments were inoperable. The answer key states the correct answer is (A), One PAM instrument is inoperable. The question stem does not specify that only Technical Specification (TS) 3.3.7 PAM instruments should be considered to answer the question. Additionally, LI-0757B is a TS 3.3.7 indication, but the question analysis states LI-0757B is the only PAM instrument inoperable. The following points support that FI-0727A is considered a PAM instrument while answering this question:

1. When reviewing question 89, I recognized that LI-0757B was a TS 3.3.7 instrument and realized it was inoperable. I also recognized that FI-0727A is required post-accident to monitor auxiliary feedwater flow to the Bravo Steam Generator (B S/G). There was no reference to TS 3.3.7 in the question stem and interpreted the question to be asking about all Regulatory Guide 1.97, "Instrumentation For Light-Water Cooled Nuclear Power Plants To Assess Plant And Environs Conditions During And Following An Accident" instruments. I even wrote "Reg. Guide 1.97" on my exam. (Also, ORM section 3.17.6, See explanation below)

    a. Reg. Guide 1.97 states auxiliary feedwater flow indication is considered a Type D, Category 2 instrument for PWRs that is required to be available to the operator to monitor operation of the safety system and inform the operator of the necessity for unplanned actions to mitigate the consequences of an accident. (See reference 2, highlighted sections page 3 paragraph 2 and page 27)

    b. At Palisades, FI-0727A is listed as a Type D, Category 2 instrument in Appendix 7C, "Palisades Plant Regulatory Guide 1.97 Rev. 3 Parameter Summary Table," of the Palisades Final Safety Analysis Report (FSAR). (see reference 1, document page 15). FSAR Section 7.4.3.2.1 "Auxiliary Feedwater Flow Controls and Isolation Design Basis" states that "Reliable AFW flow and steam generator level instrumentation are necessary in order to adequately determine and control, from the control room or alternate shutdown stations, the performance of the safety-related portion of the Auxiliary Feedwater System since the operation of this system is considered as an anticipated operational occurrence by 10 CFR 50, Appendix A, GDC 13." (See Reference 5, first paragraph highlighted on page 7.4-12 of 7.4-21)

    c. Additionally, FSAR section 7.4.3.2.3 states, "The performance of the safety-related portion of the AFW system can be assessed by the AFW flow indicators, two for each steam generator located in the control room and alternate stations outside the control room and a wide-range water level indicator for each steam generator. (see Reference 5, page 7.4-15 of 7.4-21)

2. The question stem states 'INOPERABLE'. The Palisades Operating Requirements Manual (ORM) Table 3.17.6 items 6.1 and 6.2 address "inoperability" of the auxiliary feedwater flow indicators and specifically uses the word "inoperable" in the condition statements and in the Specification statement for section 3.17.6 of the ORM. (see reference 3) When I read the question, I knew the ORM specifically required operability of the auxiliary feedwater indications and further convinced me that the question was asking about all Reg. Guide 1.97 instruments in the stem.

3. Instrument labeling in the control room is controlled by EN-OP-129, "Operations Equipment Labeling," and specifically drawing E-49, "Nameplate Standards." Note 7 of E-49 requires that all Reg. Guide 1.97 instruments will be labeled in blue. Reference 4 depicts FI-0727A with a blue label and provides the note from drawing E-49.

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

The station does not support the candidate's assertion for the reason below:

Post-Accident Monitoring Instrumentation (PAM) is discussed in LCO 3.3.7 and its basis. From LCO 3.3.7 basis:

> *The availability of PAM instrumentation is important so that responses to corrective actions can be observed and the need for, and magnitude of, further actions can be determined. The required instruments are identified in FSAR Appendix 7C (Ref. 1) and address the recommendations of Regulatory Guide 1.97 (Ref. 2), as required by Supplement 1 to NUREG-0737, "TMI Action Items" (Ref. 3).*

It is the stations position that Regulatory Guide 1.97 is not the governing document for determination as to whether an instrument is a PAM instrument or not, but that LCO 3.3.7 is the governing document and that Regulatory Guide 1.97 recommendations have already been addressed. Based on this, only one PAM instrument is INOPERABLE as described in LCO 3.3.7.

The facility recommends that the question should be graded as approved.

***NRC Evaluation/Resolution*:**

The distractor/answer choices under discussion consist of the following:

Answer A:
A. one

Distractor B:
B. two

The stem of the question explicitly asks the applicant:

The number of INOPERABLE Post Accident Monitoring (PAM) Instrument(s) is __(1)__.

The applicant contended that the Auxiliary Feedwater flow indication, FI-0727A "Aux Feedwater Flow to 'B' SG" was also considered a PAM instrument required to be OPERABLE, and that the stem lacked sufficient information to differentiate whether "Post Accident Monitoring (PAM) Instrument(s)" should only include Technical Specification LCO 3.3.7, "Post Accident Monitoring (PAM) Instrumentation" instruments or, all instruments as specified in Regulatory Guide (RG) 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Address Plant and Environs Conditions During and Following an Accident."

While FI-0727A was INOPERABLE per Operating Requirements Manual (ORM) 3.17.6 "Other Instrumentation" Action 6.1 and 6.2 (for Auxiliary Feedwater flow indication as listed in ORM Table 3.17.6), as the applicant contended, FI-0727A is not a PAM Instrument. PAM Instrumentation is controlled by LCO 3.3.7 and PAM Instruments are specified in LCO 3.3.7 Table 3.3.7-1 "Post Accident Monitoring Instrumentation." LCO 3.3.7 is the facility licensee's governing document that defines the instrumentation that is included as "Post Accident Monitoring (PAM) Instruments." The facility licensee's Final Safety Analysis Report (FSAR) Appendix 7C, "Palisades Plant Regulatory Guide 1.97, Rev. 3 Parameter Summary Table" provides for how the facility incorporates the recommendations of RG 1.97 Rev. 3, however, not all instruments listed in RG 1.97 Rev. 3 or FSAR Appendix 7C are considered PAM Instruments.

LCO 3.3.7 specifically details which of those FSAR Appendix 7C instruments are PAM instruments.

FI-0727A is a Type D, Category 2 instrument per FSAR Appendix 7C (and RG 1.97 Rev. 3) that provides for "monitoring systems operation." As such, FI-0727A is covered under ORM 3.17.6 "Other Instrumentation," which states:

> *The Safety Functions required by Specification 3.17.6 provide alarm and indication functions to assist the operator in monitoring plant conditions. None of the required functions provide automatic actions assumed to be available in the safety analysis, therefore, operation may continue even though the function is degraded or lost provided that the specified action is met.*

The applicant assumed that all RG 1.97 Rev. 3 instruments should be considered when answering the question, however, the stem specifically asked for "Post Accident Monitoring Instruments." There was no implication that any other instrumentation should be considered. It was incorrect to make such assumptions. NUREG-1021 (Rev. 11), Appendix E, Paragraph B.7 also states, in part, "*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.*" The applicant incorrectly assumed that all RG 1.97 Rev. 3 instruments were to be included.

Furthermore, NUREG-1021, "Operator Licensing Examination Standards for Power Reactor" (Rev. 11), Appendix E, "Policies and Guidelines for Taking NRC Examination," Subsection B, "Written Examination Guidelines," Paragraph B.7, states, in part, *"If you have any questions concerning the intent or the initial conditions of a question, do not hesitate to ask them before answering the question."* The applicant was briefed on the contents of APPENDIX E prior to exam administration, and all paragraph items contained in Subsection B were read "verbatim." No questions associated with the adequacy of conditions, or any other aspect of Question 89, were raised by the applicant or any of the other applicants during administration of the exam.

Based on the information provided in the question stem, there is only <u>ONE</u> INOPERABLE Post Accident Monitoring (PAM) Instrumentation, LI-0757B, 'A' SG Wide Range Level. FI-0727A, while INOPERABLE, is not Post Accident Monitoring Instrumentation and is considered "Other Instrumentation" for monitoring systems operation. Therefore, the NRC concluded that no change was required to the key for this exam question.

SIMULATOR FIDELITY REPORT

Facility Licensee: Palisades Nuclear Plant

Facility Docket No: 050–255

Operating Tests Administered: June 22, 2020, through June 29, 2020

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