UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION III
2056 WESTINGS AVENUE, SUITE 400
NAPERVILLE, IL 60563-2657

July 29, 2025

Mike Mlynarek
Site Vice President
Holtec Decommissioning
International, LLC
Palisades Nuclear Plant
27780 Blue Star Memorial Highway
Covert, MI 49043-9530

SUBJECT: PALISADES NUCLEAR PLANT - NRC INITIAL LICENSE EXAMINATION REPORT 05000255/2025301

Dear Mike Mlynarek:

On July 8, 2025, 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 13, 2025, with Mike Bailey, Training Manager, and other members of your staff. An exit meeting was conducted via Microsoft Teams on July 22, 2025, between Jeff Borah, General Manager Plant Operations of your staff, and Bryan Bergeon, Senior Operator Licensing Examiner, to review the proposed final grading of the written examination for the license applicants. During the Microsoft Teams conversation, the NRC resolutions of any post-examination comments submitted by the facility, initially received by the NRC on June 26, 2025 and July 8, 2025, were discussed.

The NRC examiners administered an initial license examination operating test during the week of June 9, 2025. The written examination was administered by Palisades Nuclear Plant training department personnel on June 18, 2025. Three Senior Reactor Operator and nine Reactor Operator applicants were administered license examinations. The results of the examinations were finalized on July 10, 2025. Ten applicants passed all sections of their respective examinations. Two applicants were issued Preliminary Results letters. One was issued an operator license. Issuance of licenses for two SRO applicants and seven RO applicants has been delayed pending the receipt of additional information.

The as-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 December 26, 2027. The enclosure contains details of this report.

However, since two applicants received a Preliminary Results letter because of a written examination grade that is less than 80 percent, the applicants were provided copies of the written examination. For examination security purposes, your staff should consider that written examination uncontrolled and exposed to the public.

Sincerely,

Paul J. Zurawski Signed by Zurawski, Paul on 07/29/25

Paul J. Zurawski, Chief
Operations Branch
Division of Operating Reactor Safety

Docket No. 50-255
License No. DPR-20

Enclosure:
1. Examination Report 05000255/2025301
2. Post-Examination Comments, Evaluation, and Resolutions
3. Simulator Fidelity Report

cc: Distribution via LISTSERV®
M. Bailey, Training Manager

Letter to Mike Mlynarek from Paul J. Zurawski dated July 29, 2025.

SUBJECT: PALISADES NUCLEAR PLANT - NRC INITIAL LICENSE EXAMINATION REPORT 05000255/2025301

DISTRIBUTION:
Tim Steadham
RidsNrrPMPalisades Resource
RidsNrrDorlLpl3
RidsNrrDroIrib Resource
Jack Giessner
Mohammed Shuaibi
Diana Betancourt-Roldan
John Pelchat
Jared Heck
Jennifer Dalzell
R3-DORS
April Nguyen
Julie Winslow
Joe Mancuso
Ikeda Betts
Maryia Bahdanovich-Kniazkova

ADAMS Accession Number: ML25210A240

OFFICE	RIII/DORS/OB:CE		RIII/DORS/OB:BC					
NAME	BBergeon:gmp		PZurawski					
DATE	07/29/25		07/29/25					

OFFICIAL RECORD COPY

U.S. NUCLEAR REGULATORY COMMISSION

REGION 3

Docket No:	50-255
License No:	DPR-20
Report No:	05000255/2025301
Enterprise Identifier	L-2025-OLL-0061
Licensee:	Holtec Decommissioning International, LLC.
Facility:	Palisades Nuclear Plant
Location:	Covert, MI
Dates:	June 9, 2025 to July 8, 2025
Examiners:	B. Bergeon, Senior Operations Engineer, Chief Examiner T. Henning, Senior Operations Engineer, Examiner G. Roach, Senior Operations Engineer, Examiner J. DeMarshall, Senior Operations Engineer, Examiner J. Nance, Operations Engineer, Examiner
Approved by:	Paul J. Zurawski, Chief Operations Branch Division of Operating Reactor Safety

SUMMARY OF FINDINGS

ER 05000255/2025301; 06/09/2025-07/08/2025; Holtec Decommissioning International, LLC, Palisades Nuclear Plant. Initial License Examination Report.

The announced initial operator licensing examination was conducted by regional Nuclear Regulatory Commission (NRC) examiners in accordance with the guidance of NUREG-1021, "Operator Licensing Examination Standards for Power Reactors," Revision 12.

<u>Examination Summary</u>:

Ten of twelve applicants passed all sections of their respective examinations. One applicant was issued an operator license. One SRO applicant and one RO applicant were issued Preliminary Results letters for failure of one section of the administered examination. Issuance of licenses for two SRO applicants and seven RO applicants has been delayed pending the receipt of additional information. (Section 4OA5.1).

REPORT DETAILS

4OA5 Other Activities

.1 Initial Licensing Examinations

a. Examination Scope

The NRC examiners and members of the facility licensee's staff used the guidance prescribed in NUREG-1021, "Operator Licensing Examination Standards for Power Reactors," Revision 12, 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 developed the operating test outlines and developed the written examination and operating test. The NRC examiners validated the proposed examination during the week of April 28, 2025, with the assistance of members of the facility licensee's staff. During the on-site validation week, the examiners audited twelve 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 9, 2025 through June 13, 2025. The facility licensee administered the written examination on June 18, 2025.

On June 26, 2025 and July 8, 2025, the licensee submitted documentation noting that there were three post-examination comments for consideration by the NRC examiners when grading the written examination. The post-examination comments and the NRC resolution for the post-examination comments are provided in Enclosure 2 of this report.

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 2 percent of the proposed examination questions were determined to be unsatisfactory and required modification or replacement.

During validation of the written examination, several questions were modified or replaced. All changes made to the proposed written examination, were made in accordance with NUREG-1021, "Operator Licensing Examination Standards for Power Reactors," and documented on Form 2.3-5, "Written Examination Review Worksheet." The Form 2.3-5, the written examination outlines, 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 December 26, 2027, (ADAMS Accession Numbers for the following: administrative files ML24319A261, examination outlines ML24319A265, proposed exam ML24319A266, and as-administered exam ML24319A263, respectively).

The NRC examiners graded the written examination on July 9, 2025, and conducted a review of each missed question to determine the accuracy and validity of the examination questions. Post-examination analysis revealed generic weaknesses in applicant performance in the areas of Generic Knowledge and Abilities, specifically SRO knowledge of plant administrative procedures and with General Fundamentals Knowledge, specifically with two questions being incorrectly answered by more than 50 percent of applicants. These weaknesses have been captured in Palisades' corrective action program under CR 25010237.

(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. Less than 20 percent of the proposed operating test portion of the examination was determined to be unsatisfactory and required modification or replacement.

During the validation of the operating test, several Job Performance Measures (JPMs) were modified or replaced, and some modifications were made to the dynamic simulator scenarios. Changes made to the operating test portion of the examination, were made in accordance with NUREG-1021, "Operator Licensing Examination Standards for Power Reactors," and documented on Form 2.3-3, "Operating Test Review Worksheet." The Form 2.3-3, the operating test outlines, and both the proposed and final as administered dynamic simulator scenarios and JPMs, will be available electronically in the NRC Public Document Room or from the Publicly Available Records component of NRC's ADAMS on December 26, 2027, (ADAMS Accession Numbers for the following: administrative files ML24319A261, examination outlines ML24319A265, proposed exam ML24319A266, and as-administered exam ML24319A263, respectively).

The NRC examiners completed operating test grading on July 10, 2025.

Post-examination analysis revealed generic weaknesses in applicant performance in the areas of SRO knowledge of Technical Specifications at the functional unit level and reactor operator over-reliance on senior reactor operator abnormal and emergency operating procedure knowledge/actions and directions. These weaknesses have been captured in Palisades' corrective action program under CR 25010238.

(3) Examination Results

Three applicants at the Senior Reactor Operator (SRO) level and nine applicants at the Reactor Operator (RO) level were administered written examinations and operating tests. Ten of twelve applicants passed all sections of their respective examinations. One applicant was issued an operator license on July 10, 2025. One SRO applicant and one RO applicant were issued Preliminary Results letters for failure of the written examination section of the administered examination. Issuance of licenses for two SRO applicants and seven RO applicants has been delayed pending the receipt of additional information.

.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, Section 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 the administration of a scenario, a contractor was able to forcibly open an exterior door to the simulator area, outside of the simulator, but within the simulator envelope under exam security. The contract employee was promptly removed from the simulator area by the facility licensee training staff. The contractor was exposed to no examination material, resulting in no loss of examination material. The door was subsequently secured and the exam security boundary revalidated. The issue was determined to be minor.

(2) During the administration of a scenario, an applicant discovered a procedure that had been marked up by the previous crew and had not been replaced. The applicant who discovered the marked up procedure immediately pointed out the procedure to the lead examiner. As the marked up procedure did not provide an advantage to the applicant who discovered the issue, the issue was determined to be minor.

4OA6 Meetings

.1 Debrief

The chief examiner presented the examination team's preliminary observations and findings on June 13, 2025, to Mike Bailey, Training Manager and other members of the Palisades Nuclear Plant's Operations and Training Department staff.

.2 Exit Meeting

The chief examiner conducted an exit meeting on July 22, 2025, with Jeff Borah, General Manager Plant Operations, via Microsoft Teams. The NRC's final disposition of the station's grading of the written examination and post-examination comments were disclosed and discussed during the Microsoft Teams meeting. The chief examiner asked the licensee whether any of the retained submitted 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

<u>Licensee</u>

Jeff Borah, General Manager Plant Operations
Steve Vonk, Assistant General Manager Plant Operations
Mike Bailey, Training Manager
Jim Byrd, Assistant Operations Manager, Training
Coty Dover, Superintendent Operations Training
Tom Giebelhausen, Superintendent Operations Training
Amy Filbrandt, Supervisor Regulatory Affairs
Bert Kosbar, Lead Examination Author

<u>U.S. Nuclear Regulatory Commission</u>

J. Mancuso, Senior Resident Inspector
T. Okamoto, Resident Inspector
B. Bergeon, Chief Examiner
T. Henning, Examiner
G. Roach, Examiner
J. DeMarshall, Examiner
J. Nance, Examiner
A. Nguyen, Team Leader
J. Winslow, Senior Project Engineer

ITEMS OPENED, CLOSED, AND DISCUSSED

<u>Opened, Closed, and Discussed</u>

CR 25010238	Initial License Exam Operating Test General Weaknesses
CR 25010237	Initial License Exam Written Exam General Weaknesses
PAL-07695	Potential Exam Security Event

LIST OF ACRONYMS USED

ADAMS	Agencywide Document Access and Management System
CR	Condition Report
PAL	Palisades Condition Report
NRC	U.S. Nuclear Regulatory Commission
RO	Reactor Operator
SRO	Senior Reactor Operator

Facility/Applicant Comments and NRC Resolutions

Question #44

Plant conditions:

- An Excess Steam Demand Event occurred at full power, resulting in a reactor trip
- The turbine had to be manually tripped
- Containment pressure = 3.8 psig and rising slowly
- 'A' S/G pressure = 495 psia and lowering
- 'B' S/G pressure = 595 psia and rising

Assuming NO operator actions, what is the status of each S/G's Feed Reg Valve?

	__'A' S/G__	__'B' S/G__
A.	open	open
B.	open	closed
C.	closed	open
D.	closed	closed

Answer: C

A. Incorrect: 'A' S/G FW will isolate on low S/G pressure. 'B' S/G FW will remain unaffected.

B. Incorrect: 'A' S/G FW will isolate on low S/G pressure. 'B' S/G FW will remain unaffected; plausible that it will also isolate on a FW isolation signal

C. Correct: Feedwater is isolated from either a CHP signal or low S/G pressure. A CHP signal will close FRV and FRV bypass valves for both S/Gs. The setpoint for containment isolation on containment high pressure is 4 psig, so CHP will not cause the FW isolation in this case. Feedwater will isolate to the 'A' S/G as pressure is less than 512 psia. Feedwater will be unaffected to the 'B' S/G as pressure is > 512 psia.

D. Incorrect: 'B' S/G FW will remain unaffected; plausible that it will also isolate on a FW isolation signal.

Facility Comment:

The question postulates a steam line break and provides SG pressures at some point during the event and asks if there was a FDW Isolation. The isolation actuates at 512 psia. See below.

The suggested correct answer assumes that when the isolation occurred, one SG is over the setpoint and increasing and one is below the setpoint and lowering.

Palisades contends that there is a psychometric flaw with the question in that the question does not reference the time in the sequence where the isolation actually occurred (how low B SG pressure was when the isolation occurred).

The key is based on the assumption that the isolation just occurred with one SG below and one SG above the setpoint but there is nothing in the question to base that assumption on. It would have been just as easy to state that 30 seconds ago, both SGs were below the setpoint and the isolation occurred with one recovering and one continuing to depressurize.

Without knowing what each SG pressure was at the time of the FW Isolation, it is impossible to determine if one or both SGs feedwater isolation signals were actuated. Recommend accepting answer D as well as answer C as correct.

PALISADES NUCLEAR PLANT
ALARM AND RESPONSE PROCEDURE

Proc No ARP-21
Revision 55
Page 10 of 33

TITLE: REACTOR PROTECTIVE SYSTEM SCHEME EK-06 (C-06)

RACK "B"			
1	2	3	4
5	6	7	8

LO PRESSURE SG1 CHANNEL TRIP	
Sensor:	Bistable Trip Unit (Any 1 of 4)
Trip Setpoints:	512 psia
Alternate Indication:	Steam Generator E-50A Pressure Indicators

AUTOMATIC FUNCTION:

- Reactor Trip on 2 of 4 coincidence logic.
- Both MSIVs (CV-0501 and CV-0510) close at 512 psia on 2 of 4 coincidence logic in the MSIV closing circuitry.
- 'A' S/G Main Feedwater Regulation and Bypass Valves (CV-0701 and CV-0735) close at 512 psia on 2 of 4 coincidence logic in the MSIV closing circuitry.

OPERATOR ACTION:

- **CHECK** Steam Generator E-50A pressure indicators:
  - PIC-0751A
  - PIC-0751B
  - PIC-0751C
  - PIC-0751D

FOLLOW UP ACTION:

- IF faulty instrument, THEN:
  - **BYPASS** Low Pressure SG1 Trip unit for affected channel per SOP-36.
  - **REFER TO** Technical Specifications LCO 3.3.1, LCO 3.3.3, LCO 3.3.4, LCO 3.3.7.

REFERENCES:

- Technical Specifications LCO 3.3.1, LCO 3.3.3, LCO 3.3.4, LCO 3.3.7
- SOP-36, "Reactor Protective System and Anticipated Transient Without Scram (ATWS) System"

Supplemental information provided by the facility on 07/08/2025:
During exam review session, one candidate verbally asserted that both options C and D should be considered correct. The candidate identified the range at which a containment high pressure signal can actuate at is 3.7 to 4.3 psig per LCO 3.3.3 table 3.3.3-1 (provided below). With the given containment pressure given as 3.8 and rising it is not possible to determine the state of the CHPs signal. Additionally, the way it was taught supports this conclusion (provided below).

Table 3.3.3-1 (page 2 of 2)
Engineered Safety Features Instrumentation

FUNCTION	APPLICABLE MODES	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
5. Containment High Pressure (CHP)			
a. Containment High Pressure — Left Train	1,2,3,4	SR 3.3.3.2 SR 3.3.3.3	≥ 3.7 psig and ≤ 4.3 psig
b. Containment High Pressure — Right Train	1,2,3,4	SR 3.3.3.2 SR 3.3.3.3	≥ 3.7 psig and ≤ 4.3 psig
6. Containment High Radiation			

Lesson Plan: PL-SIS r08-lp, Safety Injection System

Lesson Content	Instructor Notes
b. Indicated by red lights on panel C-13	
c. The setpoint was chosen so as to be low enough to avoid actuation during plant operating transients, but to be high enough to be quickly actuated by a LOCA or MSLB.	
VA-64, 65	
5. Containment High Pressure (CHP) @ 3.7 to 4.3 psig.	
a. Indicated by EK-1361 "CONTAINMENT HI PRESSURE	> 3.7 < 4.3 psig LCO Table 3.3.3-1
b. The setpoint was chosen so as to be high enough to avoid actuation by containment temperature or atmospheric pressure changes, but low enough to be quickly actuated by a LOCA or a MSLB in containment.	

After subsequent review, it is Palisades' position that this produced a set of overlapping correct answers described in Nureg 1021 ES-4.2 section D.12. and that both answers C and D should be accepted as correct.

NRC Resolution: Recommendation accepted.

The question tests applicant knowledge of the main steamline pressure setpoint to cause automatic feedwater isolation.

The facility contends that choice D, "A S/G [feed reg valve] closed, B S/G [feed reg valve] closed" is also a correct answer to the question, given that the question stem does not state the lowest steam generator (S/G) pressures observed in both S/G's main steamlines or a time at which those pressures occurred. The facility contends that it is reasonable to assume that the intact steam generator ('B') could have lowered below the low pressure S/G channel trip (main steamline isolation and feedwater isolation) setpoint of 512 psia.

The facility states that "it would have been just as easy to state that 30 seconds ago, both SGs were below the setpoint and the isolation occurred with one recovering and one continuing to depressurize." While valid, the intent of the question was to test the applicant's knowledge of knowing, from memory, the main steamline and feedwater isolation setpoint (low pressure S/G channel trip) and the system response to the reaching that setpoint. Providing information that the low pressure S/G channel trip setpoint had been exceeded or not exceeded would have reduced the question to a simple recollection of the resulting effects from exceeding that setpoint (and eliminated any need for understanding the difference between a high containment pressure (CHP) isolation and the low pressure S/G channel trip) and reduced the difficulty and cognitive level of the question.

With both MSIVs open and the intact S/G ('B') coupled to the faulted S/G ('A'), both S/G pressures would lower until the S/Gs were decoupled, which occurs by isolating the intact S/G from the faulted S/G. The main steam isolation and feedwater isolation signals (low pressure S/G channel trip setpoint) occur at 512 psia (as referenced in annunciator response procedure EK-0602). As referenced in EK-0602, low steamline pressure (below 512 psia) will close both S/G MSIVs but will only cause a train specific feedwater isolation (closing the associated feedwater regulating valve (FRV) and bypass valve).

Given the initial conditions in the stem of the question, only the 'A' S/G was below the isolation setpoint of 512 pisa (at 495 psia), with 'B' S/G above the setpoint at 595 psia. Given this information, both MSIVs were closed due to the 2 of 4 coincidence logic in the MSIV closing circuitry, but only the 'A' S/G FRVs were closed as only the 'A' S/G was below the 512 psia setpoint on 2 of 4 coincidence logic in the MSIV closing circuitry. With the 'B' S/G at 595 psia, the 'B' MSIV would be closed as a result of the 'A' S/G below the low pressure S/G channel trip setpoint, but the 'B' S/G FRVs would remain open.

While no lowest observed S/G pressures were listed in the question initial conditions, it is not a reasonable assumption to make that the intact 'B' S/G pressure given at 595 psia and rising would have lowered below the low pressure S/G channel trip setpoint of 512 psia, also given that the faulted 'A' S/G pressure was given at 495 psia and lowering. Assuming both S/G pressures were equalized and lowering while coupled (MSIVs open), when the isolation setpoint of 512 psia was reached and the S/Gs became decoupled, the faulted 'A' S/G would have lowered an additional 17 psia and the intact 'B' S/G would have recovered 83 psia. This not an operationally valid assumption, especially considering the ongoing cooldown that would impact the pressure recovery of the intact 'B' S/G.

NUREG-1021, "Operator Licensing Examination Standards for Power Reactor" (Rev. 12), Section ES-1.2, "Guidelines for Taking NRC Examinations," 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."* All applicants were briefed on the contents of section ES-1.2 prior to exam administration, and all paragraph items contained in Subsection B were read "verbatim." No questions concerning any aspect of Question 44 were raised by any of the applicants during administration of the exam.

Evaluation of supplemental information provided on 07/08/2025

An applicant and the facility provided additional supplemental information on 07/08/2025 contending that "the range at which a containment high pressure signal can actuate at is 3.7 to 4.3 psig per LCO 3.3.3 table 3.3.3-1 (provided below). With the given containment pressure given as 3.8 and rising it is not possible to determine the state of the CHPs signal." The contention that the provided containment pressure was within the allowable tolerance of the

CHP actuation setpoint, which would result in a closure of both MSIVs and both train FRVs. With both trains' MSIVs and FRVs closed, this would result in answer choice D being correct.

NUREG 1021 (Rev. 12), Section ES-1.2, Subsection C, "General Operating Test Guidelines," Paragraph C.3, states, in part, "However, you should know from memory certain automatic actions, setpoints, interlocks, operating characteristics, and the immediate actions of emergency and other procedures, as appropriate to the facility." Applicants are expected to know setpoints from memory, as described. Applicants are not expected to know from memory, setpoint tolerances as referenced in Technical Specifications, for example. However, as the applicant and facility contend, the nominal CHP actuation setpoint of 4 psig has a tolerance of 3.7 psig to 4.3 psig that is specifically taught to applicants in lesson plan PL-SIS r08-lp, Safety Injection System and is also documented in Technical Specification Section 3.3.3, Table 3.3.3-1. Additionally, UFSAR Chapter 7, Instrumentation and Controls, Section 7.2.3.9, Containment High Pressure, states: "Four independent pressure switches actuate trip units which are connected in a two-out-of-four coincidence logic to initiate the reactor protective action when the containment pressure reaches 3.7 psig."

UFSAR Chapter 7, Instrumentation and Controls, Section 7.3.3.2 states: "Containment high-pressure (CHP) signal will initiate closure of the main steam isolation valves to reduce the inventory blowdown from the intact steam generator in the case of a main steam line break, reducing the peak containment pressure and temperature as required in the accident analysis. CHP also closes the main and bypass feedwater regulating valves (Modification FC-906-1990). See Subsection 7.5.1.3."

The question stem provides a containment pressure of 3.8 psig and rising. With a designed allowed tolerance for a CHP actuation of 3.7 to 4.3 psig, it is reasonable for an applicant to assume that CHP actuation has been exceeded, and CHP has actuated. Given the reasonability in the assumption that CHP has actuated based on exceeding the low actuation tolerance of 3.7 psig, both trains' MSIVs and FRVs would close.

Based on the above, the NRC concludes that choice D is also correct. Two answer choices, C and D, as annotated on the answer key, are considered correct. The final written examination and answer key is revised to reflect the acceptance of D as an additional correct answer.

Question #80

Initial plant conditions

- Time = 1000
- The plant experienced a complete loss of offsite power and onsite emergency power
- Offsite power is expected to be restored at 1300

Current plant conditions

- Time = 1110
- EOP-3.0, “Station Blackout Procedure” is being performed.
- Both Station Batteries No. 1 and No. 2 voltages are reading 104 volts.

What action will the CRS choose next?

A. Go to EOP-9.0, “Functional Recovery Procedure” and implement FLEX Support Guides (FSGs).
B. Isolate BOTH station batteries from their respective DC buses and continue in EOP-3.0.
C. Go to EOP-9.0, “Functional Recovery Procedure" and refer to AOP-41, “Alternate Safe Shutdown Procedure.”
D. Stay in EOP-3.0 and implement FLEX Support Guides (FSGs).

Answer: C

A. Incorrect: Plausible as EOP-9.0 will be implemented based on the conditions described in the question, but the correct procedure for FLEX response is EOP-3.0. An ELAP is not declared if offsite power is expected to be restored within the license basis station blackout time period of 4 hours.

B. Incorrect: Plausible as the correct action is to isolate the station batteries from the DC bus at this voltage level, but EOP-3.0 does not contain appropriate steps to address the loss of both DC buses.

C. Correct: Power is expected to be restored within the license basis station blackout; however, this is not an uncomplicated station blackout. Some additional failures have resulted in a loss of safety function requiring the additional actions of the functional recovery procedure and potentially requiring the evacuation of the control room.

D. Incorrect: Plausible as FLEX Support Guidelines provide steps to avoid DC Bus voltages lowering to 105 VDC, however, this event is not an ELAP by definition and as such, the appropriate declarations and actions have not been taken to avoid the conditions described in the question.

Facility Comment:

The question is asking for actions during a station blackout. The station would be initially in EOP-3 and battery voltage would be monitored per EOP Supplement 7/8. EOP-3 Steps 19 and 20 address the actions required for battery voltage less that 105 vdc. EOP-3 directs isolating both station batteries (RNO Steps 19.1 and 20.1).

19. **VERIFY** the following on ED-10L/ED-10R DC Bus:	19.1 **PUSH** 72-01, Isolation Breaker to DC Battery No 1 ED-01 (Shunt Trip Breaker). LOCATION: ED-11A behind 1C Bus
• Any faults are isolated	19.2 IF ELAP declared, THEN **GO TO** Step 20.
• Voltage greater than 105 volts on EVI-27/D1 LOCATION: Cable Spreading D30L	19.3 **GO TO** EOP-9.0, "Functional Recovery Procedure." 19.4 **REFER TO** AOP-41, "Alternate Safe Shutdown Procedure."
20. **VERIFY** the following on ED-20L/ED-20R DC Bus:	20.1 **PUSH** 72-02, Isolation Breaker to DC Battery No 2 ED-02 (Shunt Trip Breaker). LOCATION: EP-1011 in 1-2 EDG room
• Any faults are isolated	20.2 IF ELAP declared, THEN **GO TO** Step 21.
• Voltage greater than 105 volts on EVI-27/D2 LOCATION: Cable Spreading D30R	20.3 **GO TO** EOP-9.0, "Functional Recovery Procedure." 20.4 **REFER TO** AOP-41, "Alternate Safe Shutdown Procedure."

Procedure No 4.06, Emergency Operating Procedure Development and Implementation, Attachment 15, EOP Performance Standards, states "*It is permissible to perform multiple steps simultaneously*." See next page.

The stem of the question asks, "*What action will the CRS choose next*?" The next action the CRS will take is "*Isolate BOTH station batteries from their respective DC buses and continue in EOP-3.0*." This answer is "B". After isolating both station batteries the CRS will remain in EOP-3.0 and perform steps 19.2 and 20.2 to determine that an ELAP is NOT declared. He will then GO TO EOP-9.0, "Functional Recovery Procedure."

The station recommends that the answer key be changed to indicate that "B" is the correct answer. The question will be modified to update the Answer Analysis to explain why "B" is correct and "C" is not. Corrected question is attached.

Proc No 4.06
Attachment 15
Revision 28
Page 4 of 16

EOP PERFORMANCE STANDARDS

NOTE:	Due to the concern of reducing the readability of the EOPs, User Alert boxes for allowance of steps out of sequence will not be used.

a. Non-sequential steps should be utilized whenever the condition specified in the step is met. If a non-sequential step is needed prior to its normal sequence, then the operator shall ensure that all sequential steps required to perform the function specified by the non-sequential step, are completed prior to performing the non-sequential step. Note that this does not require completion of all sequential steps, only steps applicable to the non-sequential step.

b. It is permissible to perform multiple steps simultaneously. The CRS is responsible for ensuring the completion of all steps and is also responsible for any steps marked N/A. It is imperative that the CRS coordinating the procedure keep track of steps completed and steps bypassed. The procedure is provided with a section for placekeeping (Section 5.0) to aid the CRS in keeping track of steps completed.

NRC Resolution: Recommendation accepted.

The question tests applicant knowledge of procedural requirements in response to station battery discharge during a station blackout.

The facility contends that answer choice C, "Go to EOP-9.0 and refer to AOP-41" is incorrect and the correct answer, is choice B, "Isolate BOTH station batteries from their respective DC buses and continue in EOP-3.0."

The question stem states that a station blackout (SBO) has occurred (implied by a complete loss of offsite power and onsite emergency power) and offsite power is expected to be restored three hours after the onsite of the SBO. Additionally, the question states that 70 minutes after the onset of the SBO, both station batteries 1 and 2 voltages are reading 104V. Given the station blackout, expected power restoration time, and the battery discharge state, the question asked what action with the CRS choose next.

Given that EOP-3.0, "Station Blackout Procedure" is being performed, step 1 states:

Within 1 hour, VERIFY any of the following AC power sources will be restored within 4 hours:

- *offsite power*
- *EDG 1-1*
- *EDG 1-2*

If any of these sources cannot be confirmed to be restored in the four hours, an extended loss of all AC power (ELAP) must be declared. In the bounds of the question stem, power is expected to be restored in three hours, so an ELAP would not be declared. Per the EOP-3.0 Basis procedure, *"As discussed in previous sections of this basis document during an ELAP the strategy is to utilize FLEX alternate equipment via the FSGs as necessary to maintain key* *safety functions. Step 30 of EOP-3.0 initiates FSGs based on actual plant conditions."*After determining an ELAP is not declared in step 1, the next procedural action to be taken, given the initial conditions in the question stem, would be to isolate the station batteries in step 19 and 20. Given that DC battery voltages on both station batteries are less than 105 volts, step 19.1 states *"PUSH 72-01, Isolation Breaker to DC Battery No 1 ED-01 (Shut Trip Breaker)".* Doing so will isolate the station battery from the bus. This is directed prior to steps 19.2, 19.3, and 19.4, which read as follows:

Step 19.2: *"IF ELAP declared, THEN GO TO Step 20"*
Step 19.3: *"GO TO EOP-9.0, Functional Recovery Procedure"*
Step 19.4: *"REFER TO AOP-41, Alternate Safe Shutdown Procedure"*

Step 20 repeats the same process for the No.2 station battery. While step 20.1 (isolating the No. 2 station battery from its bus; which would result in both station batteries isolated from their respective buses) is not necessarily required to be completed prior to transitioning to EOP-9.0 and referring to AOP-41. However, Procedure No 4.06, Emergency Operating Procedure Development and Implementation, Attachment 15, EOP Performance Standards, states "*It is permissible to perform multiple steps simultaneously.*" This provides allowance for the CRS to direct performing step 20 prior to exiting to EOP-9.0. The EOP-3.0 Basis procedure for step 19 states: *"Stripping DC Bus ED10 will result in less than three of the four Preferred AC Buses being energized. This condition will warrant exiting this procedure and going to EOP-9.0. AOP-41 is referenced to assist the operator with equipment control since DC control power from the deenergized DC Bus is not available."* So, while transitioning to EOP-9.0 and referring to AOP-41 is a correct answer, it is only correct AFTER at least one station battery is isolated from its bus. Given the initial conditions in the question stem, both station batteries are degraded, at 104V and in accordance with EOP-3.0, both should be isolated from their respective buses in step 19.1 and 20.1. There are no other earlier procedural transitions from EOP-3.0 to EOP-9.0 prior to step 19, given the initial conditions of the question stem.

Based on the above, the NRC concludes that choice B is correct and not choice C, as annotated on the answer key. The final written examination and answer key is revised to reflect the change from choice C to B.

Question #95 *Supplemental information provided on 07/08/2025.*

In accordance with EN-FAP-OM-030, Operational Decision Making, complete the following statements:
1) **__(1)__** will be addressed by the Operational Decision Making (ODM) process.
2) The Shift Manager **__(2)__** be assigned as the ODM Team Leader for long term safety or operational risk decisions.

	(1)	**(2)**
A.	Operating at 25% power for 24 hours	will
B.	Operating at 25% power for 24 hours	will NOT
C.	Developing a 5 gpm S/G tube leak at 100% power	will
D.	Developing a 5 gpm S/G tube leak at 100% power	will NOT

Answer: B

A. Incorrect: First part is correct. Second part is plausible because the Shift Manager can be the ODM Team Leader for short term safety or operational risk decisions.

B. Correct: Prolonged operation at reduced power levels would be addressed using the ODM process. EN-FAP-OM-030 page 10 has a note that reads that "low power operation at extended periods of time (12 or more hours) is increased risk." The Shift Manager would NOT be the ODM Team Leader for long term safety or operational risk decisions.

C. Incorrect: First part is plausible and would be true if primary to secondary leakage remained within LCO 3.4.13 limits; but it exceeds those limits; and requires shutting down the unit. Second part is plausible because the Shift Manager can be the ODM Team Leader for short term safety or operational risk decisions.

D. Incorrect: First part is plausible because it does require shutting down the unit as stated in choice C. Second part is correct.

Facility Comment:

Question 95 asks which plant condition requires use of the ODM process as well as team lead assignment. The plant conditions are a 5 gpm tube leak which would be controlled by the Abnormal Procedure for a SGTL (This is the stated incorrect answer) and Operation at 25 percent power for 24 hours (This is the stated correct answer).

The justification for the correct answer was determined per notes in EN-FAP-OM-030, Operational Decision Making, Section 3.3 which establishes criteria for ODM:

- The risk thresholds are such that, unless potential consequences are negligible or improbable, the adverse condition is monitored and the formal decision-making process is recommended.
- Reactor operation at low power levels for extended periods of time (12 or more hours) is considered increased risk to nuclear safety.

The question makes the assertion that 25 percent power is considered "Low Power."

It is Palisades contention that 25 percent reactor power is not considered "low power" and hence not an increased risk to nuclear safety, so it does not meet the ODM criteria.

Low Power is not defined in GOP-5, Power Up In Mode 1 or GOP-8, Power Down To Mode 2 which are the procedures that would be in progress for the stated correct answer. Per NUREG 1021 Glossary: LOW POWER: In accordance with NUREG-1449, Shutdown and Low-Power Operation at Commercial Nuclear Power Plants in the United States, issued September 1993, the range of reactor power from criticality to 5 percent.

Therefore, it is Palisades position that there is no correct answer to Question 95 and it should be removed from the exam.

NRC Resolution: Recommendation denied.

The question tests the applicant's knowledge of the administrative Operational Decision Making (ODM) process.

The facility's contention surrounds part 1 of the question, which requires the applicant to determine which circumstance will be addressed by the ODM process. The facility contends that no correct answer exists as "25% reactor power is not considered "low power" and hence not an increased risk to nuclear safety so it does not meet the ODM criteria" and the question should be removed from the exam.

Part 1 of the question references operation at 25 percent power for a 24-hour duration. Palisades procedure EN-FAP-OM-330, ODM, step 1.4 states *"The following list is examples of issues that may require entry into this procedure and would be addressed using the Operational Decision Making process: Prolonged operation at reduced power levels."*

A 24-hour operation at 25 percent power would be considered prolonged operation at a reduced power level. This meets the requirements of EN-FAP-OM-330 for examples that would be addressed using the ODM process.

The only mention of "low power" in EN-FAP-OM-330 is a note that states *"Reactor operation at low power levels for extended periods of time (12 or more hours) is considered increased risk to nuclear safety."* This does not state that <u>only</u> operation at "low power" levels is within the bounds of the ODM process.

The other answer choice to part 1 of the question is "Developing a 5 gpm S/G tube leak at 100% power." This would be incorrect as this would be handled under the appropriate abnormal operating procedure.

Step 1.3 of EN-FAP-OM-330 states *"Three typical scenarios illustrate operational and technical decisions in response to plant conditions. First, Shift Managers lead the control room team to make immediate decisions in response to off-normal conditions. Second, station managers make decisions in response to plant conditions that often fall below action threshold defined in license documents or are not clearly defined by existing normal operating procedures. Third, utility executives make decisions that address longer term protection of the public, the workforce, one or both plants, and the overall site. The focus of this procedure is on those decisions in the second scenario above. In this scenario, plant conditions may involve reductions of safety margins and can occur over a period of days or weeks."*

Given that the steam generator tube leak would require entry and mitigation in accordance with the abnormal operating procedure, this would fall under the first decision discussed in step 1.3

above. The prolonged operation (24 hours) at a reduced power level (25% power) would fall into that second decision discussed in step 1.3 above.

NUREG-1021, “Operator Licensing Examination Standards for Power Reactor” (Rev. 12), Section ES-1.2, “Guidelines for Taking NRC Examinations,” 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.”* All applicants were briefed on the contents of section ES-1.2 prior to exam administration, and all paragraph items contained in Subsection B were read “verbatim.” No questions concerning any aspect of Question 95 were raised by any of the applicants during administration of the exam.

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.

SIMULATION FACILITY FIDELITY REPORT

Facility Licensee: Palisades Nuclear Plant

Facility Docket No: 50-255; 72-007

Operating Tests Administered: June 9, 2025 to June 13, 2025

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

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

ITEM	DESCRIPTION
None	N/A