July 14, 2005

Mr. Paul A. Harden
Site Vice President
Nuclear Management Company, LLC
Palisades Nuclear Plant
27780 Blue Star Memorial Highway
Covert, MI 49043-9530

SUBJECT: PALISADES NUCLEAR PLANT
NRC INITIAL LICENSE EXAMINATION REPORT 050000255/2005301(DRS)

Dear Mr. Harden:

On May 31, 2005, the NRC completed initial operator licensing examinations at your Palisades Nuclear Plant. The enclosed report documents the results of the examination which were discussed on June 1 and June 16, 2005, with Mr. P. Harden and Mr. G. Smith, respectively, and with other members of your staff.

NRC examiners administered the operating test during the week of May 23, 2005, and on Monday May 30, 2005. NRC examiners and members of the Palisades Nuclear Power Plant Training Department staff administered the written examination on May 31, 2005. Five Reactor Operator (RO) and two Senior Reactor Operator (SRO) applicants were administered license examinations. The results of the examinations were finalized on July 7, 2005. Six applicants passed all sections of their examinations, three of these applicants were issued respective operator or senior operator licenses. One RO applicant failed the written examination and will not be issued a license. Three applicants scored less than 82 percent on the written examination; and, in accordance with the guidelines of NUREG 1021, "Operator Licensing Examination Standards for Power Reactors," ES-501.D.3.c, their licenses will be withheld until any appeal rights of the failed applicant are exhausted.

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

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

Sincerely,

/RA/

Hironori Peterson, Chief
Operations Branch
Division of Reactor Safety

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

Enclosures:
1. Operator Licensing Examination Report 050000255/2005301(DRS)
2. Simulation Facility Report
3. Post Examination Comments and Resolutions
4. Written Examinations and Answer Keys (RO & SRO)

cc w/encls 1 & 2:
J. Cowan, Executive Vice President and Chief Nuclear Officer
R. Fenech, Senior Vice President, Nuclear Fossil and Hydro Operations
D. Cooper, Senior Vice President - Group Operations
L. Lahti, Manager, Regulatory Affairs
J. Rogoff, Vice President, Counsel and Secretary
A. Udrys, Esquire, Consumers Energy Company
S. Wawro, Director of Nuclear Assets, Consumers Energy Company
Supervisor, Covert Township
Office of the Governor
Michigan Department of Environmental Quality - Waste and Hazardous Materials Division
Michigan Department of Attorney General

cc w/encls 1, 2, 3 & 4: G. Baustian, Training Manager

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U. S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket No:	50-255
License No:	DPR-20
Report No:	050000255/2005301(DRS)
Licensee:	Nuclear Management Company, LLC
Facility:	Palisades Nuclear Plant
Location:	27780 Blue Star Memorial Highway Covert, MI 49043-9530
Dates:	May 23 through June 1, 2005
Examiners:	B. Palagi, Chief Examiner N. Valos, Examiner R. Walton, Examiner C. Moore, Examiner (In Training)
Approved by:	H. Peterson, Chief Operations Branch Division of Reactor Safety

Enclosure 1

SUMMARY OF FINDINGS

ER 05000255/2005301(DRS); 05/23/2005-06/01/2005; Palisades Nuclear Plant; Initial License Examination Report.

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

<u>Examination Summary</u>:

- Seven examinations were administered (five Reactor Operator and two Senior Reactor Operator).

- Six applicants passed all sections of their examinations, three of these applicants were issued respective operator or senior operator licenses. One RO applicant failed the written examination and was not issued a license. Three applicants scored less than 82 percent on the written examination; and, in accordance with the guidelines of NUREG 1021, "Operator Licensing Examination Standards for Power Reactors," ES-501.D.3.c, their licenses will be withheld until any appeal rights of the failed applicant are exhausted.

REPORT DETAILS

4. OTHER ACTIVITIES (OA)

4OA5 Other

.1 Initial Licensing Examinations

a. Examination Scope

The NRC examiners conducted an announced initial operator licensing examination during the weeks of May 23, 2005, and May 30, 2005. The NRC examiners used the guidance established in NUREG-1021, "Operator Licensing Examination Standards for Power Reactors," Revision 9, to prepare the examination outline and to develop the written examination and operating test. The NRC examiners administered the operating test during the week of May 23, 2005, and on Monday May 30, 2005. The NRC examiners and members of the Palisades Nuclear Power Plant (PNPP) Training Department administered the written examination on May 31, 2005. Five Reactor Operator (RO) and two Senior Reactor Operator (SRO) applicants were examined.

b. Findings

Written Examination

The licensee reviewed the written examination developed by NRC examiners. Written examination comments developed during review by the Palisades staff and as a result of examination validation were incorporated into the written examination in accordance with the guidance contained in NUREG-1021.

A total of seven post-examination comments (4 RO; 3 SRO exam comments) were submitted by the station's training department personnel on June 7, 2005. The results of the NRC's review of the station's comments are documented in Attachment 3, Post Examination Comments and Resolutions.

Operating Test

The NRC examiners validated the operating test during the validation week and replaced or modified several items in the proposed operating test. Test changes, agreed upon between the NRC and the licensee, were made in accordance with NUREG-1021 guidelines.

Test/Examination Results

Six applicants passed all sections of their examinations, three of these applicants were issued respective operator or senior operator licenses. One RO applicant failed the written examination and was not be issued a license. Three applicants scored less than 82 percent on the written examination; and, in accordance with the guidelines of NUREG 1021, "Operator Licensing Examination Standards for Power Reactors," ES-501.D.3.c, their licenses will be withheld until any appeal rights of the failed applicant are exhausted.

.2 Examination Security

a. Inspection Scope

The NRC examiners briefed the facility contact on the NRC's requirements and guidelines related to examination physical security (e.g., access restrictions and simulator considerations). The examiners observed the implementation of examination security and integrity measures (e.g., security agreements) throughout the examination process.

b. Findings

No findings were noted in this area. The licensee staff was observed to be enforcing correct examination security procedures.

4OA6 Meetings

.1 Exit Meeting

The chief examiner presented the examination team's preliminary observations and findings on June 1, 2005, to Mr. P. Harden and other members of the Operations and Training Department staff. A subsequent exit via teleconference was held on June 16 with Mr. G. Smith following review of the site post examination comments. The licensee acknowledged the observations and findings presented. No proprietary information was identified by the station's staff during the exit meeting.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee
P. Harden, Site Director
G. Hettel, Plant General Manager
K. Smith, Operations Manager
G. Baustian, Training Manager
T. Davis, Operations Training Supervisor
G. Smith, Initial Operations Training Supervisor
D. Hensley, Initial Operations Training
K. Yeager, Operations Supervisor
R. Snuggerud, Operations Training

NRC
J. Ellegood, Senior Resident Inspector

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened, Closed, and Discussed

None

LIST OF ACRONYMS USED

ADAMS	Agency-Wide Document Access and Management System
DRS	Division of Reactor Safety
NRC	Nuclear Regulatory Commission
PARS	Publicly Available Records
RO	Reactor Operator
SRO	Senior Reactor Operator

SIMULATION FACILITY REPORT

Facility Licensee: Palisades Nuclear Power Plant

Facility Docket No.: 50-255

Operating Tests Administered: May 23 - May 30, 2005

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	

Question No. 13:

During a Station Blackout what indication(s) are available to determine when Battery No. 1 (D01) is approaching a full discharged condition?

A. **ONLY** Voltage indication for Battery No. 1 can be used.

B. **EITHER** Voltage or Amperage indications for Battery No. 1 can be used.

C. **ONLY** Amperage indication for Battery No. 1 can be used.

D. **EITHER** Voltage, Amperage, CR annunciator, or Frequency indications for Battery No. 1 can be used.

Original correct answer: B.

Answer: c.
Facility Reference:
ONI-R22-1, attachment 1.

Facility Comment:

Distractor B: " **EITHER** Voltage or Amperage..." could be interpreted to imply that either voltage ALONE, or amperage ALONE could be used, but NOT both. While amperage does respond and may be helpful in diagnosing a battery near fully discharged condition it cannot be used alone. High or low amperage can be indicative of battery loading. Without a relative voltage reading, amperage indication alone is not adequate for diagnosing a battery approaching a fully discharged condition.

EOP-3.0 Station Blackout, requires that if bus voltage drops to 105 volts that the shunt trip push buttons be pressed for that bus. This ensures the battery can perform its safety function prior to being overdutied. The requirement does not mention bus amperage. Therefore, Distractor A is also acceptable.

Facility Recommendation: accept both A and B as correct.

NRC Resolution:

Upon review of the question and the facility comment it was decided to accept both A and B as correct answers. The intent of the question was that the candidate recognized that both Voltage (in EOP-3.0) and Amperage (in EOP Supplement 7) indications are available to diagnose a battery problem that could result in loss of the battery. However, at least one of the candidates argued that since procedure EOP-3.0 Station Blackout uses only voltage to indicate that action must be taken to prevent a battery from becoming dangerously discharged answer "A.," **"ONLY** Voltage indication for Battery No. 1 can be used.," should also be considered correct. The argument for answer "A" also being a correct answer was reasonable, and both answers A and B were accepted as correct.

Question No. 23:

The plant is operating at 100% Rx power when a failure of Cooling Tower Pump P-39A has caused condenser vacuum to degrade. Loss of Condenser Vacuum procedure ONP-14 has been entered. A rapid power reduction (per ONP-26) was ordered by the SRO. Following the power reduction, and reactor trip, condenser pressure stabilized at 15" Hg. During the rapid downpower, what was the fastest allowable rate of power reduction, and assuming condenser pressure remains constant what would PCS temperature be after the reactor trip?

A. 60%/Hr and 532 degrees F

B. 300%/Hr and 532 degrees F

C. 60%/Hr and 535 degrees F

D. 300%/Hr and 535 degrees F

Original correct answer: B

<u>Facility Comment</u>:

The question stem asks, “what would PCS temperature be.” The briefing provided to the candidates just prior to the exam, in accordance with Appendix E of NUREG 1021, Rev. 9, instructed them to answer all questions based on actual plant operation, procedures, and references, and that if they believed the answer would be different based on simulator operation or training references, they should answer based on the *actual plant.*

By design, the turbine bypass valve (TBV) does control main steam header pressure at 900 psia (531.95 degrees F at saturation). However, pressure losses between the main steam header and the steam generators, along with efficiency losses in the steam generators, resulted in a stable Tave of slightly less than 535 degrees F.

This question and answer B reflect system design, but not actual plant response. Please see attached copies of both actual plant data and simulator response that show that actual PCS temperature (Tave) stabilizes at approximately 535 degrees F with turbine bypass valve available.

Facility Recommendation: Change correct answer to D.

<u>NRC Resolution</u>:

Data from actual 1998, 2004, and 2005 reactor trips were used to verify that for the conditions given in the stem of the question actual PCS temperature (Tave) stabilizes at approximately 535 degrees F. The correct answer was changed to “D” to reflect actual plant response.

Question No. 66:

A plant shutdown is required for refueling. When can the Operating Crew declare that they have reached Mode 6?

A. When the Reactor Head is removed with SDM > 1%

B. When the Reactor Head is removed with SDM N/A

C. When the first Reactor Vessel Closure Bolt less than fully tensioned with SDM > 1%

D. When the first Reactor Vessel Closure Bolt less than fully tensioned with SDM N/A

Original correct answer: D

Facility Comment:

The question does not ask for the definition of Mode 6. The stem presents a decision point and asks, “When can the Operating Crew declare that they have ***reached*** Mode 6?” As soon as the first reactor vessel closure bolt is less than fully tensioned, the conditions of the stem are met. Since both answers C and D contain this Condition (less than fully tensioned), and since SDM is N/A for Mode 6, answer C and D are both correct.

Answer A and B are not correct, since the crew would have to declare Mode 6 entry long before the conditions of A and B are true.

Facility Recommendation: Accept both C and D as correct.

NRC Resolution:

Upon review of the question and the facility comment it was decided to accept both C and D as correct answers. The intent of the question was to test the candidates ability to recognize entry into Mode 6 based on the definition of Mode 6, answer “D.” However, the stem of the question set up a situation in which the plant was leaving Mode 5, which requires a SDM > 1%, and entering Mode 6. Under these conditions although a SDM > 1% would not be required by the definition of Mode 6 it would be present as a requirement of Mode 5. Therefore answer “C” and “D” are both correct.

Question No. 73:

The following plant conditions exist:

- All Waste Gas Decay Tanks are full except the tank currently in service
- A Containment Purge is in Progress
- D/G 1-2 is currently running for surveillance testing
- Minimum crew manning is onsite due to a Holiday

Waste Gas Decay Tank T-68B needs to be released but Radiation Monitor RE-1113 is NOT OPERABLE. What conditions must exist for the WGDT to be released?

A) Radiation Monitor RE-1113 must be returned to OPERABLE status
The Containment Purge must be secured

B) Two independent verifications of the release rate calculation are performed
Two qualified Aux. Operators independently verify the WGDT discharge line-up
Plant Stack Radiation Monitor is continuously monitored throughout the release

C) Two independent tank samples are collected
Two independent verifications of the release rate calculation are performed
Two qualified Aux. Operators independently verify the WGDT discharge line-up
The Containment Purge must be secured

D) Two independent tank samples are analyzed
Two independent verifications of the release rate calculation are performed
Two qualified Aux. Operators independently verify the WGDT discharge line-up
Plant Stack Radiation Monitor is continuously monitored throughout the release

Original correct answer: C

Facility Comment:

The stem of this question only lists some of the conditions needed to be in place to release a gas batch. It dose not list ALL required conditions (e.g., main exhaust fan must be in service).

Answer C is correct, since it is reasonable to assume that a “collected” sample would also be “analyzed.”

The last requirement in answer D, “Plant Stack Radiation Monitor is continuously monitored throughout the release,” was originally intended to be incorrect, with the other three items being correct. However, the attached references show that the plant stack radiation monitor is continuously used as a monitoring instrument. Therefore, D is also correct.

NRC Resolution:

While the Plant Stack Radiation Monitor is designed to continuously monitor exhaust gas it is not required to be operable during a Waste Gas Decay Tank release. This was verified by review of the PALISADES NUCLEAR PLANT OFFSITE DOSE CALCULATION MANUAL which allows releases to continue with the stack gas effluent system inoperable provided grab samples are taken at least once per 12 hours. Answer “C” was retained as the only correct answer.

Question 82:

Given the following:

- Power level is stable at 100%.
- Pressurizer level is being controlled by Pressurizer Level Controller LIC-0101A.
- The output of level controller LIC-0101A has just failed at 100% output signal.
- No other failures occur.

<u>Assuming no Operator actions</u>, what will charging flow be after the level controller output fails and what is the expected plant response?

a. 0 gpm; and the Reactor trips on Thermal Margin/Low Pressure.

b. 33 gpm; and Pressurizer level cycles in an approximately 11% band.

c. 44 gpm; and Pressurizer level stabilizes at approximately 57%.

d. 133 gpm; and the Reactor trips on High Pressurizer Pressure.

Original correct answer: B

<u>Facility Comment</u>:

This question has no correct answer. The correct answer was selected originally based on an understanding of the backup pressurizer level control system design, specifically, that it controls in an approximately 11 percent band. However, with the presurizer level control malfunction standing, the pressurizer level will actually oscillate over a 2 percent range, the range between where the backup program takes control (~-6%) and where it gets a signal to reset (~-4%).

Facility Recommendation: Delete question from exam since no correct answer is provided.

<u>NRC Resolution</u>:

Review of the controller design, verified that no correct answer was provided and the question was deleted. The conditions given in the stem would have resulted in control transferring back and forth between the failed and operable controller resulting an oscillation between - 4% and - 6%.

Question 95:

All plant equipment functioned as designed following a Large Break LOCA. When and why are the Charging Pump suctions aligned to the SIRWT in EOP-4.0, Loss of Coolant Accident Recovery?

A) Approximately 30 to 45 minutes; to reduce the effects of boric acid precipitation in the core

B) Approximately 30 to 45 minutes; to prevent Charging Pump cavitation due to a loss of suction

C) Within 1 hour; to ensure adequate SIRWT inventory is injected into the PCS / Containment

D) Within 1 hour; to ensure adequate shutdown margin is established

Original correct answer: A

Facility Comment:

The concern for boric acid precipitation in the core is addressed by securing emergency boration. Refer to EOP-4.0 Basis, Step 19, and EOP Supplement 40 Basis.

Re-aligning charging pump suction from the concentrated boric acid storage tanks to either the volume control tank (VCT) or the safety injection refuel water tank (SIRWT) is done for the purpose of flushing the lines associated with boric acid injection. However, it does *assist* in reducing the effects of boric acid precipitation in the core. Therefore, answer A is correct.

During a LBLOCA, once shutdown margin (SDM) requirements are met, emergency boration would be secured. However, prior to this, charging pump suction would be re-aligned to a lower boron source (SIRWT) for the purpose of flushing the injection lines, as noted in the previous paragraph. Refer to EOP Supplement 40, Charging pump Suction Alignment. Once the flushing is complete, boration would be secured by shutting off charging pumps. This action is the one that addresses the concern for excess boron in the PCS (boron precipitation).

The stem is worded to ask when and why the suction source of the charging pumps would be realigned to the SIRWT. It is reasonable that answer D is also correct; i.e., the action given in the stem (swapping suction to SIRWT) is done only after emergency boration is secured, and emergency boration is only secured once adequate SDM is established. This would occur within one hour of the condition stipulated in the stem of the question. Refer tp EOP-4.0 Basis, Step 43.

NRC Resolution:

For the conditions provided in the stem, “a Large Break LOCA.” it would be expected that the Charging Pump suction would initially be taken from the Concentrated Boric Acid Tanks for 30 to 45 minutes to establish the required shutdown boron concentration, and then by 60 minutes after the LOCA Charging Pump suction would be switched to the Safety Injection/Refueling Water Tank to reduce the effects of boric acid precipitation in the core which may occur due to boil off during a Large Break LOCA. Answer "D) Within 1 hour; to ensure adequate shutdown margin is established," is incorrect because it would imply that there is no lower time limit on the suction switch over, and does not provide the correct answer to "...why are the Charging Pump suctions aligned to the SIRWT in EOP-4.0." "A" is the only answer with both a correct timing for the suction swap combined with a correct reason for the action, therefore "A" will be the only answer accepted as correct.

Question 96:

Following a refueling outage, during core reloading in what manner is the core reloaded and why?

A) The core reloading is started at the center of the core and loaded towards the periphery to ensure both source range detectors are monitoring the core

B) The core reloading is started near an operable source range detector and loaded to the center of the core so that core uncoupling does not occur

C) The core reloading is started at the center of the core and loaded towards the periphery to ensure a potential critical configuration is not shielded from the source range detectors

D) The core reloading is started near an operable source range detector and loaded to the center of the core so that the initial fuel assemblies are supported by the core barrel

Original correct answer: B

<u>Facility Comment</u>:

Answer D is also correct. When reloading the core, or any fuel bundle, procedures require that the bundle be supported on at least one side by either another fuel assembly, or by the core shroud. This is done by starting loading on the peripheral of the core, and working inward, as noted in the provided in Reference 1. Reference 2 describes that the core shroud is an integral part of the core barrel.

<u>NRC Resolution</u>:

While it is true that Procedure EM-04-29 Step 6.1.19 states that "Fuel assemblies in the core must be supported on at least on side by either another fuel assembly or the core shroud.", there are many core locations were this would be possible and does not explain why "core reloading is started near an operable source range detector." EM-04-29 also states in Step 6.3.2 "The core shall be loaded in a manner such that core uncoupling does not occur. This can be accomplished by working from operable excore detectors toward the center of the core. It is imperative that a potential critical configuration is not shielded from the excore detectors".

This is the bases for the correct answer "B." Additional, the reference supplied to argue that the core barrel and the core shroud should be considered one in the same does not validate that claim. The reference, actually states that the Core Support Assembly "consists of the core support barrel, the core support plate and support columns, the core shrouds, ...", and later that the core shroud is attached to the core support plate which is support by the core support barrel. While it can be argued that the core barrel provides vertical fuel support it does not supply the lateral support required by Procedure EM-04-29 step 6.1.19. Therefore "B" is the only correct answer.

Enclosure 4

WRITTEN EXAMINATIONS AND ANSWER KEYS (RO/SRO)

RO/SRO Initial Examination ADAMS Accession # ML051930220.