September 30, 2005

Mr. T. Palmisano
Site Vice-President
Prairie Island Nuclear Generating Plant
Nuclear Management Company, LLC
1717 Wakonade Drive East
Welch, MN 55089

SUBJECT: PRAIRIE ISLAND NUCLEAR GENERATING PLANT
NRC INITIAL LICENSE EXAMINATION REPORT NO. 05000282/2005301(DRS); 05000306/2005301(DRS)

Dear Mr. Palmisano:

On August 19, 2005, Nuclear Regulatory Commission (NRC) examiners completed initial operator licensing examinations at your Prairie Island Nuclear Generating Plant. The enclosed report documents the results of the examination which were discussed on August 19, 2005, with L. Clewett and other members of your staff. A subsequent telephone conversation was conducted on August 30, 2005, with Mr. T. Bacon and W. Markham of your staff to discuss the NRC's resolution of the written examination post-examination comments.

The NRC examiners administered an initial license examination operating test during the weeks of August 8, and August 15, 2005. A written examination was administered by Prairie Island Nuclear Generating Plant training department personnel on August 19, 2005. Six Reactor Operator and five Senior Reactor Operator applicants were administered license examinations. Two of the Senior Reactor Operator applicants were previously licensed Reactor Operators at the Prairie Island Nuclear Generating Plant. The results of the examinations were finalized on September 22, 2005. All applicants passed all sections of their respective examinations and were issued applicable operator licenses.

In accordance with 10 CFR Part 2.790 of the NRC's "Rules of Practice," a copy of this letter and its enclosures will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records 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 by M. Bielby acting for/

Hironori Peterson, Chief
Operations Branch

Docket Nos. 50-282; 50-306
License Nos. DPR-42; DPR-60

Enclosures:
1. Operator Licensing Examination Report 50-282/05-301(DRS); 50-306/05-301(DRS)
2. Simulation Facility Report
3. Post Written Examination Comments and Resolutions
4. Written Examinations and Answer Keys (RO & SRO)

cc w/encl 1:
C. Anderson, Senior Vice President, Group Operations
J. Cowan, Executive Vice President and Chief Nuclear Officer
Regulatory Affairs Manager
J. Rogoff, Vice President, Counsel & Secretary
Nuclear Asset Manager
Tribal Council, Prairie Island Indian Community
Administrator, Goodhue County Courthouse
Commissioner, Minnesota Department of Commerce
Manager, Environmental Protection Division Office of the Attorney General of Minnesota

cc w/encl 1, 2, 3 & 4: J. Lash, Training Manager, PINGP

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U. S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket Nos. License Nos.	50-282; 50-306 DPR-42; DPR-60
Report No:	50-282/05-301(DRS)/50-306/05-301(DRS)
Licensee:	Nuclear Management Company, LLC
Facility:	Prairie Island Nuclear Generating Plant, Units 1 and 2
Location:	1717 Wakonade Drive East Welch, MN 55089
Dates:	August 09 through August 19, 2005
Examiners:	D. McNeil, Chief Examiner D. Reeser, Examiner C. Zoia, Examiner
Approved by:	Hironori Peterson, Chief Operations Branch Division of Reactor Safety

SUMMARY OF FINDINGS

ER 50-282/05-301(DRS)/50-306/05-301(DRS), on 08/09-19/05, Nuclear Management Company, LLC, Prairie Island Nuclear Generating Plant. Initial License Examination Report.

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

Examination Summary:

- Eleven examinations were administered (five Senior Reactor Operator (SRO) and six Reactor Operators (RO)).

- Eleven applicants passed all sections of their respective examinations and were issued applicable operator and senior operator licenses. (Section 4OA5.1)

Report Details

4. OTHER ACTIVITIES (OA)

4OA5 Other

.1 Initial Licensing Examinations

a. Examination Scope

The Nuclear Regulatory Commission (NRC) examiners conducted an announced operator licensing initial examination during the weeks of August 08, and August 15, 2005. The plant's training staff 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 between August 09, 2005, and August 18, 2005. Prairie Island Plant training department staff members administered the written examination on August 19, 2005. Six Reactor Operator and five Senior Reactor Operator applicants were examined. Two of the Senior Reactor Operators were previously licensed Reactor Operators at the Prairie Island Nuclear Generating Plant.

b. Findings

Written Examination

The licensee developed the written examination. During the initial review of the examination, the examiners determined that the examination, as submitted by the licensee, was within the range of acceptability expected for a proposed examination. During examination validation the week of July 11, 2005, examination changes agreed upon between the NRC and the licensee were incorporated according to the guidance contained in NUREG-1021.

The licensee graded the written examination on August 19, 2005, and on August 22, 2005, conducted a review of each question to determine the accuracy and validity of the examination questions. The licensee submitted seven post examination comments for the written examination. Resolution for the comments can be found in Enclosure 3, "Post Written Examination Comments and Resolutions."

Operating Test

The NRC examiners determined that the operating test, as originally submitted by the licensee, was within the range of acceptability expected for a proposed examination. Examination changes made during the validation of the operating test were agreed upon between the NRC and the licensee, and were made in accordance with the guidelines provided in NUREG-1021.

Examination Results

All applicants passed all sections of their respective examinations and were issued applicable operator licenses.

.2 Examination Security

a. Scope

The NRC examiners reviewed and observed the licensee's implementation of examination security requirements during the examination preparation and administration.

b. Findings

The licensee’s implementation of examination security requirements during examination, preparation, and administration were acceptable and met the guidelines provided in NUREG-1021, "Operator Licensing Examination Standards for Power Reactors," Revision 9. However, during the facility licensee’s development of questions to be used on the NRC initial written examination, a security incident occurred which had the potential to affect the integrity of the written examination.

During development of the written examination it was determined that several questions were left in an uncontrolled location at a copying machine. The questions were retrieved and the training staff immediately informed. This was a violation of the licensee’s examination security procedure for controlling examination material.

The licensee documented this incident in their corrective action program as CAP 042060. The NRC examiners were appropriately notified of the incident and the affected questions replaced. The examiners reviewed the licensee’s investigation and assessed the overall incident for possible violation of 10 CFR 55.49, “Integrity of Examinations and Tests.” The examiners determined there was no actual examination compromise, and; therefore, no violation of NRC requirements had occurred. The licensee’s failure to follow their examination security procedure was considered minor in nature and not subject to enforcement action in accordance with NRC enforcement policy.

4OA6 Meetings

Exit Meeting

The chief examiner presented the examination team's preliminary observations and findings on August 19, 2005, to Mr. L. Clewett and other members of the Prairie Island Nuclear Generating Plant Operations and Training Department staff. A subsequent exit meeting via teleconference was held on August 30, 2005, with Mr. T. Bacon and Mr. W. Markham to discuss NRC resolution of post-written examination comments. No proprietary or sensitive information was identified during the examination.

ATTACHMENT: SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee

T. Palmisano, Site Director
T. Bacon, General Superintendent, Operations Training
R. Graham, Director Site Operations
P. Huffman, Operations Manager
J. Kempkes, Shift Supervisor
J. Lash, Training Manager
T. Ouret, Examination Writer
J. Sorensen, Vice President NMC Training

NRC

J. Adams, Senior Resident Inspector, Prairie Island Plant
D. Karjala, Resident Inspector, Prairie Island Plant

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened, Closed, and Discussed

None

LIST OF DOCUMENTS REVIEWED

None

LIST OF ACRONYMS USED

ADAMS	Agency-Wide Document Access and Management System
DRS	Division of Reactor Safety
NRC	Nuclear Regulatory Commission

Enclosure 2

SIMULATION FACILITY REPORT

Facility Licensee: Monticello Plant

Facility Docket No.: 50-282; 50-306

Operating Tests Administered: August 09 - 18, 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
ERCS	During two simulator scenarios, the ERCS event timer would not reset. The failure did not impact the examinations. CAP 044022 has been written to document the failure and track the progress of software fixes.
121 CL Pump	During a simulator scenario the 121 CL pump would not start when it appeared that all interlocks were met, and the pump should have started. Applicants took compensatory measures and started other cooling systems during the scenario. They were unaware of the simulator failure. CAP 044021 was written to track the investigation of the 121 CL pump start failure.
Diesel Generators	During a simulator scenario the D1 Emergency Diesel Generator ran for approximately 30 minutes with an approximate 700-800 kW load. The diesel generator did not seize or trip as expected from a long run without cooling water. CAP 044020 was written to investigate the need for some type of response to an extended engine run without cooling water.

Post Written Examination Comments and Resolutions

Question # 04: Answer c.

During a Large Break LOCA, all ECCS flow is assumed to bypass the core until the completion of the Blowdown Phase. During the Refill Phase immediately following blowdown, the ECCS flow is directed to the _________.

a. cold legs AND reactor vessel simultaneously to refill the core from the top and bottom at the same time.

b. reactor vessel ONLY as complete core uncovery occurs during blowdown and core injection is the most effective cooling method.

c. cold legs ONLY to refill the core barrel and start the recovery of the core from the bottom up.

d. cold legs AND hot legs simultaneously to ensure either SI or Accumulator injection will pass through the core on the way to the break.

Question #04 Applicant Comment:

USAR Section 14.6 Table 14.6-3 assumption 2C states “assuming offsite power available.” The note states “studies show that continued operation of the reactor coolant pumps results in worse peak cladding temperature.” Thus offsite power must be maintained as described in the question. Table 14.6-4 timeline and the description of page 14.6-3 show SI injection to the cold legs at 12 sec. and upper plenum injection (RHR) at 22 sec. Since the blowdown phase did not end until 21.5 sec., and the bottom of core uncovery starts at 33.5 sec., RHR is injecting during the refill stage.

Facility response:

There are conflicting references regarding this question. B18B, Emergency Core Cooling System, Section 4.3.C states that refill is accomplished by the accumulators. The above referenced USAR section gives a timeline showing that RHR is injecting during the refill stage. A Procedure Change Request has been submitted to change B18 B to refer to the current LBLOCA analysis.

Based on these conflicting references, the site recommends accepting both answers “a.” and “c”.

NRC resolution:

After reviewing the comment, the NRC agreed that distractor “a.” was a correct answer. However, if “a.” is a correct answer, “c.” cannot be accepted as a correct answer. Distractor “c.” allows ECCS injection ONLY to the cold legs. Since the reference material indicated there would be RHR injection occurring into the upper plenum, then distractor “c” must be an incorrect answer. The answer key was modified to accept distractor “a” as the correct answer and distractor “c” as an incorrect answer.

Question #11: Answer d.

All offsite power has been lost and safeguards buses are being supplied by their respective diesel generators.

How can the status of each Pressurizer Backup Heater Bank be determined from the Control Room?

a. Check for RED light indication on the associated Heater Bank Control Switch; if LIT, the bank is ENERGIZED.

b. Check for RED light indication on the Bank A and B Heater Bank Control Switches; if LIT, the bank is ENERGIZED. Banks D and E are NOT energized regardless of control switch indication.

c. Use the ERCS M1.97 display to view the power supplied to Bank A heaters. Banks B, D, and E are NOT energized regardless of control switch indication.

d. Use the ERCS M1.97 display to view the power supplied to Bank A or Bank B heaters. Banks D and E are NOT energized regardless of control switch indication.

Question #11 Applicant Comment:

Reference NUREG 1021, Appendix E Part B.7 states not to assume conditions that are not specified in the stem of the question. Due to the fact we are trained to assume a normal lineup makes “c” and “d” correct.

Facility response:

A normal Pressurizer Heater Lineup would have Bank A powered from a safety-related source and Banks B, D, and E from a non-safeguard source that would be deenergized based on stem conditions. ERCS display “M1.97" displays the power (KW) supplied to A and B heater banks. If a normal lineup is assumed, the operator could verify power to Bank A while knowing power is lost to Banks B, D, and E. This would make answer “c.” correct. Since the ERCS display will show the status of banks A and B regardless of electrical lineup, and states correctly that D and E Banks are deenergized regardless of breaker indication in the second part, “d.” is also correct.

NRC resolution:

After reviewing the comment, the NRC agreed that the grading for the question should be changed to accept answers “c.” and “d.” as correct answers. The original reasoning for distractor “c.” being incorrect was that the status of heater Bank B may not be determined since the heater bank may be aligned to its alternate source. Since no conditions were provided in the stem of the question that would cause the heater bank to be aligned to its alternate source, an applicant can assume heater Bank B is aligned to its normal source. With heater Bank B aligned to its normal source, distractor “c.” is an additional correct answer. Distractor “d.” was correct for any plant conditions. The answer key was amended to accept distractors “c.” and “d.” as correct answers for question # 11.

Question #25: Answer c.

Given the following:

- 1ECA-3.2 SGTR WITH LOSS OF REACTOR COOLANT: SATURATED RECOVERY is in progress.
- RCS Tave is 552°F and lowering.
- 12 SG is isolated with level 65% NR, rising at 2%/minute.
- Cooldown of the RCS is in progress using the Condenser Steam Dumps from 11 SG.
- A 95°F/hr cooldown rate has been established using steam dump MANUAL control.
- 11 SG steam flow is 0.53x106 lbm/hr.
- Pressurizer level is 30% and rising at 3%/minute.
- NO further operator action is taken.

Which ONE of the following conditions will occur FIRST assuming current trends continue?

a. 11 MSIV automatically closes.

b. 12 SG level goes offscale high.

c. Steam Dump flow is lost.

d. Pressurizer fills water solid.

Question #25 Applicant Comment:

RO candidates are not required to memorize sequence of steps after transitions and if SI is reset or not. If it was stated SI was reset in the stem then I would know “c.” was the only acceptable answer. Recommend accepting “a.” and “c.” as correct.

Facility response:

RO candidates should know the major actions involved in an EOP, including the steps taken to establish an operator-controlled cooldown to the condenser. SI is reset prior to establishing the cooldown as it is preferred to use the Condensate and Feedwater system for steam generator level control, and these pumps cannot be operated until SI is reset. SI will be reset even if these pumps are not used or available. Recommend no change to this question.

NRC resolution:

After reviewing the comment, the NRC agreed that the grading for the question should not change. The NRC agrees that Reactor Operator candidates are not required to memorize sequence of steps after transitions, but believes that Reactor Operators must have an overall understanding of what actions and plant responses that should have taken place during an evolution to correct/assist a Senior Reactor Operator if an incorrect order is issued to the Reactor Operator. For example, if the Senior Reactor Operator orders a cooldown without resetting SI, the Reactor Operator would be responsible to question/remind the Senior Reactor about resetting SI. No change was made to the answer key for question # 25.

Question #38: Answer d.

Given the following on Unit 1:

- The Unit was at 100% power.
- A steam line break occurred in Containment.
- The reactor and turbine tripped.

The following conditions are noted:

- Containment pressure is 28 psig and increasing.
- B Train Containment Spray failed to actuate automatically or manually.

What action (if any) is required to prevent exceeding Containment design pressure limits?

a. Locally start 12 Containment Spray Pump and manually open the discharge valve.

b. Reset Containment Spray and stop Train A Containment Spray.

c. Verify all four CFCUs are operating in Slow with full Cooling Water flow.

d. None, one train of Containment Spray is adequate.

Question #38 Applicant Comment:

The bases for TS 3.6.5 Containment Cooling Systems under the LCO section states one train of CS and one train of CFCUs are required as a minimum for a SLB. Also, above 23 psig containment pressure the crew would be in FR-Z1 which ensures all CFCU's are operating and CL pressure is adequate. Of the selections given, "c." is correct. Accept both "c." and "d." due to conflicting references.

Facility response:

TS 3.6.5 states the requirement for one train of Containment Spray (CS) and one train of Containment Fan Coil Units (CFCU) for a Steam Line Break. The crew would be in FR-Z.1 given the current conditions. Answer “d.” states one train of CS is adequate, and other than B train CS, no other failures are given. With this information, answer “d.” is correct. Additionally, answer “c.” states, that, all four CFCU’s are operating in Slow with full Cooling Water flow. Since it can be inferred from the stem that A train CS is working properly, “c.” is also correct. Recommend accepting answers “c.” and “d.” as correct.

NRC resolution:

After reviewing the comment, the NRC agreed that distractor “c” was a correct answer. The A train of CS was in service (given in the stem) and four CFCUs would be in service after completing the action to verify they were in service specified by distractor “c.” Since only one train of CS and one train of CFCUs is required to protect containment for a Steam Line Break (SLB), distractor “c.” is correct. Distractor “d.” states, that, no action is required because one train of CS is adequate. This distractor is correct because the stem does not prevent automatic start of the CFCUs, resulting in one train of CS and two trains of CFCUs in service. Distractor “d.” was accepted because no action would be required to protect the containment. The answer key was modified to accept distractors “c.” and “d.” as correct answers.

Question #54: Answer “a.”

You are an extra operator performing a post-LLRT lineup on a containment penetration.

You hear a variable tone siren (wailing, like a police siren) with about a 4-second cycle, but due to noise in the area are unable to hear the announcement that follows.

What has occurred, and what action is required?

a. Containment Evacuation Alarm. Immediately evacuate containment using the nearest airlock and card out of containment.

b. Fire Alarm. Complete the valve lineup to ensure the containment penetration is isolated, then exit containment and assist the Fire Brigade or Control Room as directed.

c. High Flux at Shutdown Alarm. Immediately evacuate containment using the nearest airlock and card out of containment.

d. Site Evacuation Alarm. Complete the valve lineup to ensure the containment penetration is isolated, then exit containment and report to the North Warehouse.

Question #54 Applicant Comment:

New police sirens sound more like our fire alarm - they are shorter in cycle. Answer “a.” and “b.” should be correct since it is too subjective based on the description of the siren.

Facility Response:

The cycle of the evacuation alarm is significantly longer than the fire alarm and should be recognizable by the information given.

Additionally, the action for distracter “b.” is not correct. Containment boundary control is maintained by tagging closed the redundant valves while performing the post-LLRT lineup. Continuing the lineup is not expected provided the job is in a safe condition. Recommend no change to this question.

NRC resolution:

After reviewing the comments, the NRC agrees with the facility’s response. The description of the siren was adequate to determine it was a Containment Evacuation alarm. Additionally, the NRC agrees that continuing the job assignment without determining the cause of the alarm is not a correct response. Distractor “a.” was retained as the only correct answer to question # 54.

Question #67: Answer d.

The control room has been evacuated per F5 Appendix B CONTROL ROOM EVACUATION-FIRE. You are conducting Attachment D, Unit 2 Reactor Operator Actions.

You have been unsuccessful in contacting the Shift Supervisor at the Hot Shutdown Panel using the radio.

What is the preferred alternate method of communication with the Shift Supervisor at the Hot Shutdown Panel?

a. Have an extra person (runner) deliver a written message

b. Sound Powered Telephone on Circuit 1

c. Gaitronics Page

d. Telephone

Question #67 Applicant Comment:

F5 Appendix B does not list alternative methods in order of preference. It only does a list of alternative methods, and leaves it up to the operator.

Facility response:

F5 Appendix B addresses five methods of communications:

- Radio (directed to obtain prior to evacuation)
- Telephone (phone numbers for stations have been added to the attachments)
- Gaitronics paging

- Sound Powered Telephone
- Runners

F5 Appendix B Assumption 2.1.5 states, that, runners are the only credited means of communications. The other communication methods have connections in the control or relay room BUT they may be used only as long as they continue working. This guidance is repeated in Section 3.0 and the note at the top of Section 4.0.

The answer given was based on the procedural note for the phone number (listed at the top of Attachment A), but each method has unique advantages and disadvantages, and one or more may be used to establish and maintain communication. As an example, runners are specifically credited and not affected by the fire. Sound powered phones can be worn on station and would connect all the stations together. telephones would be the fastest direct form of communication if available. Paging systems may be used to get the attention of other personnel and direct them to establish communications.

Since the communications methods in the question are not listed in order of preference, any of the answers given could be considered “preferred.” Recommend deleting this question from the exam.

NRC resolution:

After reviewing the comments, the NRC agreed with the station’s recommendation to delete the question from the examination. Distractors “a.,” “b.,” and “d.” are “preferred” correct answers under a broad range of circumstances not provided in the stem of the question, but necessary to correctly answer the question. Using a runner may be “preferred” if the other circuits are out as a result of the fire, making distractor “a.” correct. If the sound powered phone system is cross-connected and being monitored because the radio and telephone systems are not working, it would be the “preferred” method of communicating. If available, the telephone would be the fastest method of establishing communications with the Shift Supervisor. Because there are three correct answers to this question, the NRC determined that the question should be deleted from the examination. The answer key was modified to delete question #67 from the examination.

Question #71: Answer “c.”

You are the RO during a shutdown for a refueling outage. The Containment HP requests that Containment In-Service Purge be placed in service "as soon as possible" to reduce dose to workers.

Which of the following is the EARLIEST plant mode reached that will allow for establishment of Containment In-service Purge?

a. Entry into MODE 3 Hot Standby.

b. Entry into MODE 4 Hot Shutdown.

c. Entry into MODE 5 Cold Shutdown.

d. Entry into MODE 6 Refueling.

Question #71 Applicant Comment:

Procedurally, the earliest in-service purge may be lined up is in Mode 5, but the earliest by Tech Specs is Mode 3. Nothing prevents changing our procedures to perform leak testing to initiate in-service purge earlier in Mode 3. Recommend accepting “a.” and “c.”

Facility response:

Facility operating procedures do not allow operation of Containment In-Service Purge until MODE 5 is reached. This action is required as the blank flanges must be removed and the inside containment dampers do not meet containment integrity requirements. Recommend no change to this question.

NRC resolution:

Upon review of the comments, the NRC concurs with the stations recommendation that no change be made to the answer key for this question. Technical Specifications are not operating procedures. The station’s operating procedures are designed to prevent operating the station in violation of the station’s Technical Specifications. NRC examination questions must be answered referring to the current plant conditions and procedures unless otherwise specified in the question. Following the station’s current plant procedures, the station is prohibited from operating Containment In-Service Purge until Mode 5 is reached. Because current plant procedures and plant physical limitations prohibit Containment In-Service Purge operation until Mode 5, the NRC accepts only distractor “c.” as a correct answer to question #71 and did not modify the answer key.

Enclosure 4

WRITTEN EXAMINATIONS AND ANSWER KEYS (RO/SRO)

RO/SRO Initial Examination ADAMS Accession # ML052720170