October 29, 2003

Mr. Joseph Solymossy
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 05000282/2003301(DRS); 05000306/2003301(DRS)

Dear Mr. Solymossy:

On September 18, 2003, the NRC completed administration of initial operator licensing examinations at your Prairie Island Nuclear Generating Plant. The NRC finalized the results of the examination on October 22, 2003, following the review of the post exam comments submitted by your staff on September 25, 2003. The enclosed report presents the results of the examinations.

NRC examiners administered the operating test during the weeks of September 8 and 15, 2003. Members of the Prairie Island Nuclear Generating Plant training staff administered the written examination on September 18, 2003. Six reactor operator (RO) and six senior reactor operator (SRO) applicants were administered written examinations and operating tests for initial operator licensing. Two applicants passed all sections of their respective examinations. Six SRO and four RO applicants failed the written examination and will not be issued a license. Two RO applicants scored an 80.8 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, the RO applicants' licenses will be withheld until any appeal rights of the other proposed license applicant failures, which may impact the outcome of the examination, are exhausted. Ten of twelve applicants failing the examination was an abnormally high failure rate. Your staff would be expected to evaluate these failures to determine whether deficiencies exist in your initial licensed operator training program.

In accordance with 10 CFR Part 2.790 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/

Roger D. Lanksbury, Chief
Operations Branch
Division of Reactor Safety

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

Enclosures:
1. Operator Licensing Examination Report 05000282/2003301(DRS); 05000306/2003301(DRS)
2. Post Examination Comments and Resolution
3. Simulation Facility Report
4. Written Examinations and Answer Keys (RO & SRO)

cc w/encls 1, 2 & 3: Plant Manager, Prairie Island
R. Anderson, Executive Vice President
Mano K. Nazar, Senior Vice President
John Paul Cowan, Chief Nuclear Officer
Manager, Regulatory Affairs
Jonathan Rogoff, Esquire General Counsel
Nuclear Asset Manager
Commissioner, Minnesota Department of Health
State Liaison Officer, State of Wisconsin
Tribal Council, Prairie Island Indian Community
Administrator, Goodhue County Courthouse
Commissioner, Minnesota Department of Commerce

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

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

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U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket Nos:	50-282; 50-306
License Nos:	DPR-42; DPR-60
Report No:	05000282/2003301(DRS); 05000306/2003301(DRS)
Licensee:	Nuclear Management Company, LLC
Facility:	Prairie Island Nuclear Generating Plant
Location:	1717 Wakonade Drive East Welch, Minnesota 55089
Dates:	September 8 through 18, 2003
Examiners:	C. Phillips, RIII NRC Chief Examiner R. Morris, RIII NRC Examiner N. Valos, RIII NRC Examiner R. Lanksbury, RIII NRC Observer
Approved by:	Roger Lanksbury, Chief Operations Branch Division of Reactor Safety

SUMMARY OF FINDINGS

ER 05000282/2003301(DRS), 05000306/2003301(DRS); 09/08/2003-09/18/2003; Prairie Island Nuclear Generating Plant, Units 1 and 2.

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.

Examination Summary:

- Twelve examinations (six Reactor Operator (RO) and six Senior Reactor Operator (SRO)) were administered.
- Six SRO applicants and four RO applicants failed the written examination and will not be issued an operator license. All twelve applicants passed the operating test. (Section 4OA5.1)
- Two RO applicants passed all sections of their respective examinations. These two applicants each scored an 80.8 percent on the written examination and will not receive an RO license until appeal rights of the other proposed license applicant failures, which may impact the outcome of the examination, are exhausted. (Section 4OA5.1)

REPORT DETAILS

4. OTHER ACTIVITIES (OA)

4OA5 Other

.1 Initial Licensing Examinations

a. Examination Scope

The NRC examiners conducted an announced operator licensing initial examination during the weeks of September 8 and September 15, 2003. The facility'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 during the weeks of September 8 and September 15, 2003. Members of the Prairie Island training department administered the written examination on September 18, 2003. Six Reactor Operator (RO) and six Senior Reactor Operator (SRO) applicants were examined.

b. Findings

Written Examination

The licensee developed the written examination. The NRC examiners determined that the written examination, as originally submitted by the licensee, was outside the acceptable quality range expected by the NRC. This determination was based on the fact that 27 out of 100 written questions required replacement or significant modification. Seven of the above questions were identified by the licensee during post-exam review. The problems identified with the written examination included, but were not limited to, questions submitted with an incorrect answer or multiple correct answers, 52 percent of the questions for the reactor operator exam were written at the memory level (NUREG-1021 allows for no more than 50 percent), and questions submitted containing non-plausible distractors. Examination changes, agreed upon during the examination validation week of August 11, 2003, between the NRC and the licensee, were made according to the guidance contained in NUREG-1021. The licensee indicated that they would be performing a root cause analysis to address the submitted examination quality. The licensee would be expected to incorporate any lessons learned from this effort into future examination submittals.

The licensee submitted 10 post-examination question changes. This number of post-examination changes was higher than would normally be expected. The examiners reviewed these changes with facility personnel. The examiners accepted 8 of the 10 requested changes. One of the other two changes requested resulted in a deletion of the question entirely instead of the answer change the licensee had requested. The other requested change was not accepted and the answer to the question remained as originally presented in the examination. The licensee has subsequently entered this issue into their Corrective Actions Program for further evaluation and action. The changes resulting from the examiner's post-examination review are documented in Enclosure 2, Post Examination Comments and Resolution.

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, agreed upon during examination validation the week of August 11, 2003, between the NRC and the licensee, were made according to the guidance contained in NUREG-1021 with one exception. The examiners requested a minor change to an event in one of the dynamic scenarios during the validation week. The documentation was changed but the actual input into the simulator was not. The first scenario was run with the event as originally written instead of as requested by the examiners. The examiners again requested that the event within the scenario be changed to what was agreed upon. The second scenario was run but the licensee had inserted a different change to the event than what was agreed upon. The examiners determined that the following scenarios would be run without the event at all. In addition, the Chief Examiner verified the simulator inputs before the running of all subsequent scenarios.

Examination Results

Six RO applicants and six SRO applicants were administered written examinations and operating tests for initial operator licensing. Two applicants passed all sections of their respective examinations. Six SRO and four RO applicants failed the written examination and will not be issued a license. Two RO applicants scored an 80.8 percent on the written examination and will not receive a license until all appeal rights of the other proposed license applicant failures, which may impact the outcome of the examination, are exhausted. Should the reactor operator candidates who failed the written examination appeal, a subsequent review of the written exam may result in question deletions or changes which may affect the licensing decision of the RO applicants with a score of 80.8 percent.

.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) and integrity (e.g., predictability and bias). The examiners observed the implementation of examination security and integrity measures (e.g., security agreements, sampling criteria, bank use, and test item repetition) throughout the examination process.

b. Findings

The following findings document four violations of NRC examination security requirements:

- On April 24, 2003, an instructor not on the 2003 initial license examination security agreement got access to a copy of the written exam outline. A new written exam outline was prepared.

- On August 12, 2003, an instructor that had prepared portions of the initial license examination and was on the 2003 initial license examination security agreement administered five plant Job Performance Measures (JPMs) to applicants during the licensee's audit exam. When this was identified the licensee replaced the individual as an audit exam administrator. The licensee determined that the individual provided no instruction or feedback to the license candidates. No exam material was replaced.
- During performance of two scenarios during the examination, the examiners identified the failure to erase place-keeping marks in some procedures between dynamic simulator scenarios. This could give an applicant an unfair advantage while completing the dynamic simulator scenario. In this particular case, there was no affect on the outcome of the operating test.

- During the performance of a JPM, during the examination, the licensee left a discarded sheet of material that had the correct answer to the JPM indicated on it in the examination material used by one of the applicants. The examiner was observing the applicant closely and determined that the applicant had successfully completed the JPM prior to the applicant becoming aware that the discarded sheet of material was present. This situation was discussed with the NRC Branch Chief and it was determined that no examination material had to be replaced.

These violations were considered minor in nature because no examination material was actually compromised; therefore, they would not be subject to enforcement action. These findings have been entered into the licensee's corrective action program.

4OA6 <u>Meetings</u>

<u>Exit Meeting</u>

The chief examiner presented the examination team's preliminary observations and findings on September 18, 2003, to Mr. Solymossy and other members of the Operations and Training Department staff. The licensee acknowledged the observations and findings presented.

ATTACHMENT: SUPPLEMENTAL INFORMATION

PARTIAL LIST OF PERSONS CONTACTED

<u>Licensee</u>
J. Solymossy, Site Vice President
T. Bacon, Operations Training Supervisor
S. Cook, Nuclear Oversight Manager
G. Eckholt, Regulatory Affairs Manager
M. Gardzinski, Simulator Support
B. Gillespie, Operations Manager
J. Lash, Training Manager
M. Werner, Plant Manager

<u>NRC</u>
P. Hiland, Deputy Director Division of Reactor Projects
C. Phillips, Chief Examiner
J. Adams, Senior Resident Inspector
R. Morris, Resident Inspector Point Beach

LIST OF ACRONYMS USED

AC	Alternating Current
ADAMS	Agency-Wide Document Access and Management System
AFW	Auxiliary Feedwater
AMSAC/DSS	ATWS Mitigation System Activity Circuit/Diverse Scram System
ATWS	Anticipated Transient Without Scram
CRDM	Control Rod Drive Mechanism
CST	Condensate Storage Tank
DRS	Division of Reactor Safety
EAL	Emergency Action Level
EOP	Emergency Operating Procedure
HX	Heat Exchanger
JPM	Job Performance Measure
LCO	Limiting Condition for Operation
NC	Natural Circulation
NIS	Nuclear Instrumentation System
NPSH	Net Positive Suction Head
NRC	Nuclear Regulatory Commission
NUE	Notice of Unusual Event
PARS	Publicly Available Records
PI	Prairie Island
PORV	Power Operated Relief Valve
RHR	Residual Heat Removal
RNO	Response Not Obtained
RO	Reactor Operator
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RWST	Refueling Water Storage Tank
SAMG	Severe Accident Management Guideline
SBO	Station Blackout
SI	Safety Injection
SRO	Senior Reactor Operator
TS	Technical Specification

Post Examination Comments and Resolution

<u>Written Examination Question #12 on the Reactor Operator (RO) Examination:</u>

The applicant is asked, with all alternating current (AC) power lost, how long of a cooldown a single affected Unit can sustain given the Technical Specification (TS) minimum allowed volume of water in the condensate storage tanks (CSTs).

<u>Facility Comment:</u>

The facility licensee recommended that RO Question #12 be deleted from the examination.

The Station Blackout (SBO) Rule Compliance Report credits the TS (3.7.6) minimum volume of 100,000 gallons with meeting the requirement of the SBO rule.

The Basis for TS LCO [limiting condition for operation] 3.7.6 discusses the minimum quantity of water in a CST [condensate storage tank] on a single unit basis with the unit drawing water from its associated CST.

The exam question refers to CSTs (plural) and affected unit (singular). The normal plant configuration for Prairie Island (PI) has the CSTs crosstied via valve C-41-2. We [the facility licensee] ensure this valve is maintained open by placing a Safeguards Hold card on the valve in accordance with SWI O-3, "Safeguards Hold Cards & Component Blocking or Locking."

This normal alignment provides a minimum of 200,000 gallons to a single affected unit. This is double the water discussed in the LCO basis. Since the answer was based on the LCO basis and the question stem establishes conditions that are different, there is no correct answer for RO Question #12.

<u>NRC Resolution:</u>

The question stem is worded such that it implies that only one unit is affected by the station black out. A single unit impacted by a station black out is a possible scenario. The question attempts to determine if the candidate knows the basis for the minimum volume of water in the condensate storage tanks for an operating unit but does not specifically ask that question. Rather the question asks, if a station black out occurs and given the minimum available water in CSTs then how long can decay heat be removed and/or a cooldown be sustained. The minimum required volume of water in the CSTs is 100,000 gallons of water per Unit per TS 3.7.6. If the candidate assumes that only the 100,000 gallons of water for one unit is available for use then only answer D is correct. However, if the candidate assumes that only one unit is affected by the station black out (possible), and that there is the minimum volume of water available for both units in the CSTs (probable), and knows that the CSTs are cross-connected then all of the answers are possibly correct but it is indeterminate because no analysis was presented based on the availability of 200,000 gallons of water. The licensee's recommendation was accepted and this question was deleted.

Written Examination Question #23 on the Reactor Operator Examination:

The question asks what will be the status of safety injection (SI) after a SI has been reset (by momentarily depressing the SI reset pushbuttons) following a spurious SI signal.

Facility Comment:

Recommend answer key be modified to accept either B or C for RO Question 23.

Determining the status of Train B SI signal status at the completion of the momentary depression of the SI reset pushbuttons requires the candidate to make an assumption regarding the status of the spurious SI signal. A spurious SI is an SI actuation that is caused by any event other than plant conditions exceeding SI setpoints. No information is provided to the candidate to lead to any specific conclusion regarding the spurious SI signal status.

For Train A, the SI signal is reset and auto SI actuation is blocked when the SI reset pushbutton is depressed. When the pushbutton is released, as performed in the step by momentarily depressing the pushbutton, then the SI signal will remain reset and blocked.

For Train B, the response is dependent upon the assumed status of the initiating spurious signal. If the spurious SI signal has cleared, as perhaps from transient electrical perturbation or relay bumping, then the SI signal will reset when the pushbutton is depressed. Since the Train B reactor trip breaker is closed, the auto block function will not occur. In this case however, since there is no current SI initiation signal present, the SI signal will remain reset and un-blocked or capable of re-actuation for initiation signal. Making the assumption that the spurious signal is no longer present leads to (b) as the correct answer.

If the candidate assumes that the spurious signal is still present, as perhaps from a relay or switch failure, then SI will re-actuate when the Train B SI reset pushbutton is released. For this case Train B will be actuated, i.e., not reset, and (c) is the correct answer.

Since no information is presented for the candidate to make one assumption over the other, both choices should be acceptable.

NRC Resolution:

The examiners reviewed drawing X HIAW 1-992 and determined that B (the answer key correct answer) would be the correct answer if the spurious SI signal is the type that came in and then immediately cleared (as, for example, from a transient caused by an electrical perturbation or relay bumping). For this case, the SI signal on Train B will reset when the SI reset pushbutton is depressed. However, since the Train B reactor trip breaker is closed (it did NOT trip following the Safety Injection), the auto SI block function will NOT occur for Train B.

The examiners determined that C would be the correct answer if the spurious SI signal is the type that came in and stayed in (as, for example, from a relay or switch failure). For this case, SI will re-actuate on Train B when the SI reset pushbutton is released. For this case, Train B of SI will be actuated, that is, NOT reset. This would make C the correct answer. In summary, both B and C are considered correct answers for this question.

Written Examination Question #34 on the Reactor Operator Examination:

The question asks the candidate to determine what will happen to boric acid concentration in the reactor coolant system (RCS) and pressurizer given a boric acid addition prior to two different forms of cooldown, a normal cooldown and a natural circulation cooldown.

Facility Comment:

Recommend RO Question #34 be deleted from the examination.

The answer identified in the key assumed all charging flow is directed to the pressurizer and remains that way for the entire 200 gallon boration. If this occurs, then the entire 200 gallons of boric acid is added via the pressurizer and answer (d) is correct.

If all charging flow is not through the pressurizer during the entire 200 gallon boration, then the answer is a function of actual flow to the RCS and pressurizer compared to the normal cooldown condition.

Initiation of auxiliary spray flow will result in RCS pressure reduction. Since the RCS cooldown and depressurization have not been initiated, RCS pressure should be maintained by the operator within the normal control band. Thus auxiliary spray flow may not be supplied continuously, but rather to the extent possible while maintaining RCS pressure within the normal band.

The amount of flow through the auxiliary spray line is affected by the position of CV-31328, Regen HX [Heat Exchanger] Charging Line Outlet valve. If the valve is left open, then charging flow is supplied through both paths (auxiliary spray line and normal charging return). The normal cooldown procedure recognizes this aspect and directs the closure of CV-31328, Regen HX Charging Line Outlet valve, during RCS depressurization. E-3, “Steam Generator Tube Rupture,” Step 18 also recognized that this valve may need to be closed and provides this information in a note. Since RCS depressurization is not the intent in Step 2 of ES-0.3A, “Natural Circulation Cooldown with CRDM [control rod drive mechanism] Fans,” CV-31328, Regen HX Charging Line Outlet valve, may be either open or closed depending upon RCS pressure response and the position of this valve will affect RCS and pressurizer boration. In fact, during an actual NC [natural circulation] cooldown performed in 2001, RCS boron and pressurizer boron concentrations were logged after the boration with actual results that match choice (c) instead of the keyed answer.

If all charging (boration) flow is directed to the pressurizer, then (d) is the correct answer. If most of the charging (boration) flow is directed to the loops, then (c) is the correct answer. However, depending on pressurizer heater status and pressurizer ambient heat losses, the operator may cycle CV-31328 (for RCS pressure control) differently in the each Natural Circulation cooldown case. This factor makes all the intermediate cases between answers (c) and (d) possible. In fact, it is possible that pressurizer and RCS boron concentrations could both be the same after adding 200 gallons of boric acid making answers (a) and (b) correct as well. We recommend deleting question 34 as all answers could be correct.

NRC Resolution:

The examiners reviewed the following procedures: 1ES-0.3A, “Natural Circulation Cooldown With CRDM Fans,” Revision 12; and 1C1.3, “Unit 1 Shutdown,” Revision 51. The addition of the boric acid would take place prior to the actual cooldown in both circumstances. Step 2 of 1ES-03A requires the alignment of auxiliary spray prior to the addition of boron. However, the position of CV-31328 is normally open and changing the position of CV-31328 is not addressed in Step 2 of 1ES-03A. There is no normal procedure for aligning auxiliary spray that would discuss changing the position of CV-31328. The facility licensee asserts that the operator may change the position of CV-31328 as necessary to facilitate pressurizer pressure control without additional procedural guidance. According to the basis for 1ES-03A the purpose of aligning auxiliary spray is to ensure that enough boric acid would be added to the pressurizer such that there would not be a significant dilution caused by an out surge of pressurizer water into the RCS during cooldown. Based on the uncertainty of the possible combinations of boric acid addition flow the NRC agrees with the facility licensee’s recommendation to delete this question from the exam.

Written Examination Question #47 on the Reactor Operator Examination:

The question gives the candidate a set of plant parameters and expects the candidate to determine how this will impact the auxiliary feedwater (AFW) system.

Licensee Comment:

Recommend changing the keyed answer to C vice A for RO Question 47. The question considers the effect of loss of control power on auto-start signals in the AFW system. However, the answer did not look at the effect of the conditions stated in the stem on the AMSAC/DSS [ATWS [anticipated transient without Scram] mitigation system activity circuit/diverse scram system] which can also start the AFW pumps. The loss of power to Buses 11 and 12 results in the opening of the reactor coolant pump (RCP) breakers due to undervoltage. The opening of the RCP breakers results in AMSAC/DSS actuation. The AMSAC/DSS actuation provides a start signal to both 11 and 12 AFW pumps. As a result, both AFW pumps should be running based on a full evaluation of the conditions stated in the question stem.

NRC Resolution:

The examiners reviewed drawings NF 40795 and NF 40781-1. The loss of Busses 11 and 12 results in the opening of the RCP breakers. The opening of the RCP breakers results in an AMSAC/DSS actuation. The AMSAC/DSS actuation will start the AFW pumps. The NRC agrees with the recommendation to change the answer to C vice A.

Written Examination Question #66 on the Reactor Operator Examination:

This question gives the candidate a choice of common plant evolutions and asks which is required to be logged in the control room log.

Licensee Comment:

Recommend answer key be modified to accept either B or C for RO Question 66.

In accordance with Section 7.3.3 of SWI-O-25, “PERIODIC DATA ACQUISITIONS & LOG KEEPING,” the distracters for this question are events that should be logged in the Turbine Building Log. This would make the question correct as written.

However, Section 7.3.2, “Unit 1 [Unit 2] Reactor Logs,” Item b.1 states a requirement to log (in the Reactor Log): “All operations affecting the operation of the reactor or major unit equipment. Realignments done by or for a Work Order should include the W.O. number in the entry.”

The addition of oil to the main turbine oil reservoir is an operation which is performed under a Work Order. Since the oil addition is work on major equipment done under a Work Order, the oil addition could be interpreted as needing to be logged in the Reactor Log. A search of the Reactor Log found the log entry for a previous oil addition to the main turbine oil reservoir. Furthermore, the Control Room Log (Unit 1: PINGP 97) records Turbine Oil Reservoir level hourly. Therefore, either choice B or C is correct.

NRC Resolution:

Licensee procedure SWI O-25, “Periodic Data Acquisitions & Log Keeping,” Revision 34, Step 7.3.2.b.10, specifically states that placing a Boric Acid Tank on Recirculation will be logged in the Control Room Log making B a correct answer.

The licensee’s comment is that the addition of lube oil to the main turbine lube oil reservoir is controlled under a work order and that any major unit equipment realignments performed under a work order must be recorded in the control room log. In addition, the main turbine lube oil levels are checked hourly as part of the control room log. Any sudden increase in lube oil reservoir level would have to be explained in the control room log. The management expectation is that any addition of lube oil to the main turbine lube oil reservoir shall be documented in the control room log. The NRC agrees with the recommendation to make C a correct answer also.

Written Examination Question #3 on the Senior Reactor Operator Examination:

The question asks the candidate to pick from an assortment of plant conditions that would require a 1 hour emergency notification to the NRC.

Licensee Comment:

Recommend SRO Question #3 be deleted from the examination.

The notification to the NRC was predicated on a Notification of Unusual Event (NUE) declaration per the E-Plan Emergency Action Level (EAL) 1B for a “Failure of a safety or relief valve in a safety-related system to close following reduction of applicable pressure.”

The information in the keyed answer stated that the pressurizer power operated relief valve (PORV) has seat leakage and isolation was unsuccessful. However, there was no information provided that indicates whether the PORV opened to reduce RCS pressure. On the far left of Page 3 of F3-2, “Classification of Emergencies,” Attachment 1 is the initial logic gate which reads: “PZR safety or relief valve opens and then fails to reseat.” Since this condition is not met in the keyed answer choice, the described condition is NOT a NUE.

If this amount of PORV leakage developed at power, it would be addressed by T.S. LCO 3.4.14, “RCS Operational Leakage” and C4 AOP1, “Reactor Coolant Leak.” This would require us to notify the NRC Resident Inspector per SWI O-28, “Notification of the GSPO and Resident Inspector,” with no specified time limit.

The other answer choices have been reviewed and are still incorrect for the reasons stated on the key. Since there is not adequate information to support a NUE declaration, this question has no correct answer and this question should be deleted from the exam.

NRC Resolution:

The examiners reviewed facility licensee procedure F3-2, “Classification of Emergencies,” Attachment 1, Revision 32. The classification logic within the procedure does require that either a safety or power operated relief valve lift and fail to reseat. The distractor originally designated as the correct answer does not contain sufficient information to make the determination that an Unusual Event had occurred. Specifically, the distractor does not state that the pressurizer PORV lifted prior to the leakage detected. None of the other three distractors provides information that would indicate that a 1 hour NRC notification would be required. Therefore, the Chief Examiner agrees that there is no correct answer to this question. The NRC agrees with the recommendation that the question be deleted.

Written Examination Question #6 on the Senior Reactor Operator Examination:

The question gives the candidate a series of electrical distribution equipment failures in a timeline and asks the candidate to determine the required LCO actions at a specific time.

Licensee Comment:

Recommend changing the keyed answer to A vice B for SRO Question 6.

The following is a step by step explanation of the event described in the question stem.

Date	Time	Condition	Completion Time	Explanation
6/10	1000	A	8 hours	When the 8 hour Completion Time expires Condition D must be entered. If LCO is not met in 16 hours, then Condition D must also be entered for this reason.
6/10	1800	D due to A not met	6 hours 36 hours	Enter Condition D 6 hours to enter Mode 3. 36 hours to enter Mode 5 will force a Mode 5 entry on 6/12 @ 0600
6/10	1925	A	8 hours	This is a 2nd entry into Condition A which is already not met. No new time line is established.
6/11	0100	-	-	Information only. Not related to the time line.
6/11	0200	D due to LCO-16 not met	6 hours 36 hours	6 hours to enter Mode 3. 36 hours to enter Mode 5 based on initial entry into Condition D will force a Mode 5 entry on 6/12 @ 0600.
6/11	0615	C	2 hours	This does not affect the 16 hour clock, which has timed out. Enter Condition D for this reason.
6/11	0730	Exit A Remain in C & D	-	This does NOT clear the requirement to be in Mode 5 by 0600 on 6/12 because Condition D is not exited.

The key concept here is that initial entry into Condition D establishes the time line, requiring entry into Mode 5 by 0600 on 6/12. Additional entries into Condition D do not reset the time line. Also, removing the initial reason (inoperable MCC 1X1) for entry into Condition D, does not reset the time line since Condition D was never exited because of the inoperability of Panel 113. Thus, the original time to achieve Mode 5 remains at 0600 on 6/12; the correct answer is A and B is an incorrect answer. This derives from T.S. 1.3 which states, “An ACTIONS Condition remains in effect and the Required Actions apply until the Condition no longer exists or the unit is not within the LCO Applicability.”

NRC Resolution:

The examiners reviewed T.S. 3.8.9, Bases B3.8.9, and T.S. Section 1.3 “Completion Times.” This was an administrative error on the licensee’s part. The answer key at the time the exam was given was incorrect. The question was originally submitted to the NRC with the correct answer as stated above to the Chief Examiner. However, the administrative portion of the question that explains the rationale behind the answer was incorrect. The Chief Examiner recommended that the facility licensee change the rationale to correspond with the correct answer as given. The licensee changed the answer key to reflect the incorrect rationale instead of vice versa. The NRC agrees with the recommendation to change the correct answer to A vice B.

Written Examination Question #7 on the Senior Reactor Operator Examination:

The question gives the candidate a set of conditions where a loss of coolant accident has occurred, recirculation from the containment has begun and the emergency core cooling system pump suction strainers get clogged. The candidate is asked if a transition should be made to procedure ECA-1.1, “Loss of Emergency Coolant Recirculation,” and why.

Licensee Comment:

Recommend key be modified to accept B or C vice A for SRO Question 7.

ECA-1.1, “Loss of Emergency Coolant Recirculation,” has several strategies which would be beneficial in the situation proposed in the question stem.

1. In Step 10, one RHR [residual heat removal] pump would be stopped. This would reduce the pressure loss due to the blockage and increase the NPSH [net positive suction head] for the running RHR pump. This could restore both long-term and short-term core cooling.

2. In Step 13-RNO [response not obtained], the total injection flow could be reduced to the minimum for decay heat removal which would reduce the pressure loss due to the blockage and increase the NPSH for the running RHR pump. This could restore both long-term and short-term core cooling.

3. In Step 13-RNO b.2), charging is used to establish minimum injection flow.

4. In Step 2, actions are performed to start filling the RWST [refueling water storage tank]. Combined with Step 10, which starts an SI pump, these two steps would restore short-term cooling.

This makes either choice B or C correct. Choice A is NOT correct for the reasons stated above. As stated in the answer key, there are no explicit EOP [emergency operating procedures] transitions into ECA-1.1 after placing the unit in recirculation. However, implementing ECA-1.1 is warranted and the transition is supported by procedure ES-0.0, Rediagnosis”, Step 3 which may be entered using ES-0.0 entry conditions. In addition, E-1, “Loss of Reactor or Secondary Coolant,” Step 30 says, “Evaluate Long Term Plant Status” which would be a prudent transition to ECA-1.1 under the direction of the Technical Support Center staff. At this time of the event, all emergency response organizations would be operational.

This is further justified as Step 23 of ES-1.2, “Transfer to Recirculation,” would have been completed and the basis (in “Background Information for . . . Transfer to Recirculation”) describes that if such degradation were detected then appropriate actions . . . should be developed. These actions are developed in ECA-1.1 as described above.

Choice D is NOT correct, without making assumptions, for the reasons stated on the answer key.

NRC Resolution:

The examiners reviewed ECA 1.1, “Loss of Emergency Coolant Recirculation,” Revision 9; 1ES 0.0, “Rediagnosis,” Revision 8; and SAG 8, “Flood Containment,” Revision 0. A transition to ECA-1.1 could be a correct transition based on either using the ES-0.0, “Rediagnosis” procedure or from E-1, “Loss of Reactor or Secondary Coolant” (under the direction of the Technical Support Center) as detailed by the facility. Thus, distractor “A” is NOT a correct answer.

The second part of the question in the stem is associated with the reason for the decision to either transition or NOT to transition to ECA-1.1. The objectives of the ECA-1.1 procedure are threefold:

1. To continue attempts to restore emergency coolant recirculation capability
2. To delay depletion of the RWST by adding makeup flow and reducing outflow
3. To depressurize the RCS to minimize break flow and cause SI accumulator injection

Distractor “B” states that the reason for the transition to ECA-1.1 is because it would be effective in restoring long-term cooling. For the conditions given in the stem of the question (debris blocking the suction lines from Containment Sump B to both trains of RHR causing both RHR pumps to cavitiate), the actions taken in ECA-1.1 to reduce to one running RHR pump is NOT taken to increase the NPSH of an RHR pump, but to reduce the outflow from the RWST to delay RWST depletion. That such action would be effective in restoring long-term cooling is highly speculative and NOT supported by the bases for performing the step. For this reason, distractor “B” is NOT a correct answer.

Distractor “C” states that the reason for the transition to ECA-1.1 is because it would provide temporary core cooling until the containment is flooded. While the first part of the reason for transition to ECA-1.1 is valid (because it would provide temporary core cooling), the second part “until the containment is flooded” is NOT valid. The bases for the actions in ECA-1.1 do NOT include flooding the containment to provide core cooling. Flooding containment to provide core cooling is NOT a strategy used in the EOPs. The only place where flooding of the containment to provide core cooling is potentially used is in the Severe Accident Management Guidelines (SAMGs), where a guideline SAG-8, “Flood Containment” is one option potentially used for core cooling. However, the SAMGs are only used for core damage events (events for which the EOPs have NOT been effective at restoring core cooling). Also, once the SAMGs are entered, the EOPs are no longer used. In addition, use of SAG-8 to flood containment is evaluated before use by the licensee’s staff in the Technical Support Center because of the potentially very negative effects that flooding containment can have (e.g. loss of key instrumentation and equipment). For these reasons, distractor “C” is NOT a correct answer.

In summary, there is NO correct answer for this question and this question will be deleted.

Written Examination Question #13 on the Senior Reactor Operator Examination:

The candidate is given a situation where during a surveillance test an as found reactor trip setpoint is found to be high outside the acceptable calibration range but below the TS Allowable Value for that trip setpoint. The candidate is then told the trip setpoint cannot be lowered so that it is within the calibration range. The candidate must then make a determination of as found and as left operability.

Licensee Comment:

Recommend answer key be modified to accept A or C for SRO Question 13.

Prairie Island has discovered conflicting references as discussed in CAP032714 during the post-exam review of this question.

T.S. Bases Page 3.3.1-10 second paragraph states: "A channel is OPERABLE with an actual setpoint value outside its calibration tolerance provided the actual setpoint 'as-found' value does not exceed its associated Allowable Value and provided the setpoint 'as-left' value is adjusted to a value within the 'as-left' calibration tolerance band." This justifies the keyed answer of C.

The surveillance used to prove operability in this question would be SP 1198 [2198], "NIS Power Range Startup Test." On Page 14, there is a graphical guideline for resolving out-of-tolerance conditions that requires an inoperable declaration if the setpoint is greater than the T.S. Allowable Value of 40%. This guideline is referenced from Precautions and Limitations, Step 3.3.

Since the T.S. Basis was not provided during the exam, the candidates had to draw on memory to make the "as-left" operability determination. Depending on the plant reference remembered, the candidates could select either choice A or C. Either could be correct depending on the plant document which was referenced.

NRC Resolution:

The examiners reviewed TS Table 3.3.1-1, TS Basis 3.3.1, Surveillance Procedure SP 1198[2198], "NIS [nuclear instrumentation system] Power Range Startup Test," Revision 15, and Licensee Operator Training Module P8184L-004.

Technical Specification Basis 3.3.1 states, "The Allowable Value specified in Table 3.3.1-1 serves as the LSSS [limiting safety system setpoint] such that a channel is operable if the actual setting is found not to exceed the Allowable Value during the channel operating test. As such, the Allowable Value differs from the trip setpoint by an amount primarily equal to the expected instrument loop uncertainties, such as drift, nuclear instrument accuracies, and heat balance accuracies, during the surveillance interval. In this manner, the actual setting of the device has not drifted beyond that expected during the surveillance interval. Note that, although the channel is 'operable' under these circumstances, the trip setpoint should be left adjusted to a value within the established trip setpoint calibration tolerance band, in accordance with uncertainty assumptions stated in the referenced setpoint methodology (as-left criteria), and confirmed to be operating with the statistical allowances of the uncertainty terms assigned." This information supports the conclusion that based on the as-found surveillance test data the instrument was operable. Even though the as-found setpoint was not within the established trip setpoint calibration tolerance band it did not exceed the TS Allowable Value of 40 percent.

The above information also supports the conclusion that with the as-left trip setpoint outside of the established trip setpoint calibration tolerance band the instrument is inoperable. If the trip setpoint cannot be returned to and is left above the setpoint calibration tolerance band then it can be expected to drift to a point above the Allowable Value within the surveillance interval and would be inoperable. If the instrument cannot be relied upon to stay below the Allowable value within the surveillance interval it must be declared inoperable. This is supported by licensee procedure 5AWI 3.15.5, "Operability Determinations," Revision 8, Paragraph 6.5 which states, "If a system should fail while it is being tested in the safety mode of operation, the system is to be declared inoperable."

Title 10 of the Code of Federal Regulations Part 55.43 states in part, that the written examination for an SRO will contain a representative sample of questions on the knowledge, skills, and abilities needed for licensed senior operator duties, and those knowledge, skills, and abilities will be identified, in part, from learning objectives derived from an analysis of licensed senior operator duties. The examiners verified that the candidates had a learning objective which stated that given a set of plant conditions determine if a Limiting Condition for Operation exists. The examiners also verified that the candidates were trained on TS 3.3.1 and the Basis for TS 3.3.1. In addition, the candidates were trained that the trip setpoint for the power range low nuclear flux trip was 25 percent.

The wording of the question is problematic however, because it does not use the term "setpoint calibration tolerance band" in the stem. Rather the question stem refers to the setpoint calibration tolerance band as the "desired range." Surveillance Procedure SP 1198[2198], "NIS [nuclear instrumentation system] Power Range Startup Test," Revision 15, Step 7.3.2 clearly refers to the reactor trip setpoint as falling within an allowable range and reactor trip <u>re</u>set point as falling within a "desired range." Reactor trip reset points are not TS related and do not pose an operability question. However, the stem also clearly states that the "trip function setpoint" could be lowered no further than 25.2 percent. Even if the candidate mistakenly understood this as a reset point then the trip setpoint would still have been left above 25 percent and would have left the instrumentation inoperable.

The NRC disagrees with the licensee's recommendation. There cannot be a dual state of operability or an indeterminate state of operability. The use of the term "trip function setpoint" makes it clear that the question is being asked about the trip setpoint and not the trip reset point. With the as left value of the trip setpoint out of the calibration range then the as left value is clearly inoperable and therefore C is the only correct answer.

Written Examination Question #17 on the Senior Reactor Operator Examination:

The question gives the candidate a set of choices of plant operating modes, procedures entered, and pressurizer levels with trend information and asks in which circumstance should an initiation of safety injection and transition to a different procedure not take place.

Licensee Comment:

Recommend answer key be modified to accept B or D for SRO Question 17.

The question tests the candidates' ability to recognize entry conditions for emergency operating procedures. Specifically, the question asks, "In which circumstance should we **NOT** initiate safety injection flow and transition to a different procedure?"

Choice B is correct and choices A and C are incorrect for the reasons stated in the answer key. However, choice D is also correct because although the circumstances do require initiation of SI flow, there is NO requirement to transition to a different procedure as required by the question stem. Refer to the SI reinitiation criteria on the information page of ES-1.1.

NRC Resolution:

The licensee's response states that the question can be read more than one way. The intent of the question was to have it ask in which circumstance should both the safety injection and the procedure transition not be allowed. However, the question can be read to ask in which circumstance should either the SI or the procedure transition not be allowed. In this case D is also a correct answer. The NRC agrees with the facility licensee's recommendation and accepts answers B and D as correct.

<u>SIMULATION FACILITY REPORT</u>

Facility Licensee: Prairie Island Nuclear Generating Plant

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

Operating Tests Administered: September 8 -18, 2003

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	None.

WRITTEN EXAMINATIONS AND ANSWER KEYS (RO/SRO)

RO Initial Examination ADAMS Accession No. ML032970307
SRO Initial Examination ADAMS Accession No. ML032970331

Enclosure 4