July 2, 1999

Mr. M. Wadley
President, Nuclear Generation
Northern States Power Company
414 Nicollet Mall
Minneapolis, MN 55401

SUBJECT: OPERATOR LICENSING EXAMINATION REPORTS 50-282/99301(OL); 50-306/99301(OL)

Dear Mr. Wadley:

The Nuclear Regulatory Commission examiners completed initial operator licensing examinations at your Prairie Island Nuclear Power Station on May 21, 1999. The license applicants' performance evaluations were finalized on June 18, 1999. The Nuclear Regulatory Commission examiners observed control room operations and reviewed several administrative and operating procedures during the examination validation and administration. The enclosed report presents the results of the examination and concurrent operations inspection.

Examiners administered operating tests and written examinations to three reactor operator applicants. Two applicants passed all sections of the examination and were issued reactor operator licenses to operate your Prairie Island Nuclear Power Station. The remaining applicant passed the operating test, but failed the written examination and was denied an operating license.

Prairie Island Station training department personnel developed the operating test and written examination in accordance with NUREG 1021, "Operator Licensing Examination Standards for Power Reactors," Interim Revision 8, January 1997, and administered the written examination. Nuclear Regulatory Commission examiners administered the operating test and observed portions of the administration of the written examination.

Your training department staff submitted an examination outline that was well prepared and conformed with the guidelines contained in NUREG 1021. After the NRC examiners approved the outline, your staff developed a written examination using the approved outline. The examiners reviewed the examination and determined that twelve of the developed questions were unsatisfactory and required modification or replacement. The examiners determined that the unsatisfactory examination questions could be modified in a timely manner and the examination could proceed as scheduled. Because of the high number of unsatisfactory questions (12) and the relatively high number of post examination comments (9), the examiners concluded that construction and validation of a written examination is an area in which your staff needs improvement.

It is our expectation that the Prairie Island Station training department instructors will use the applicant and examination weaknesses outlined in the accompanying report as feedback to improve the operator license training program in accordance with your Systematic Approach to Training program.

In accordance with 10 CFR 2.790 of the Commission's regulations, a copy of this letter and the enclosures to this letter will be placed in the NRC Public Document Room.

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

Sincerely

Original /s/ D. E. Hills

David E. Hills, Chief
Operations Branch

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

Enclosures:
1. Inspection Reports 50-282/99301(OL); 50-306/99301(OL)
2. Facility Post Written Examination Comments and NRC Resolution
3. Simulation Facility Report
4. Written Examination and Answer Keys (RO)

cc w/encls 1, 2 & 3: Site General Manager, Prairie Island
Plant Manager, Prairie Island
S. Minn, Commissioner, Minnesota
Department of Public Service
State Liaison Officer, State of Wisconsin
Tribal Council, Prairie Island Dakota Community

cc w/encls 1, 2, 3 & 4: Michael J. Ladd, Plant Training Manager

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DOCUMENT NAME: G:DRS\PRA99301.WPD

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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:	50-282/99301(OL); 50-306/99301(OL)
Licensee:	Northern States Power Company
Facility:	Prairie Island Nuclear Generating Plant
Location:	1717 Wakonade Dr. East Welch, MN 55089
Dates:	May 17-21, 1999
Examiners:	D. McNeil, Chief Examiner D. Muller, RIII Examiner
Approved by:	David E. Hills, Chief, Operator Licensing Branch Division of Reactor Safety

EXECUTIVE SUMMARY

Prairie Island Nuclear Power Station
NRC Examination Reports 50-282/99301(OL); 50-306/99301(OL)

A licensee developed and Nuclear Regulatory Commission approved initial operator licensing examination was administered to three Reactor Operator license applicants. The examiners observed a period of routine operations in the control room.

Results:

Three applicants were administered an initial license examination. Two applicants passed all portions of the examination and were issued Reactor Operator licenses. One applicant passed the operating test, but failed the written examination and was denied a license.

Operations:

Operators performed their shift responsibilities in a professional, business-like manner during the observed period. Procedures reviewed by the examiners contained adequate guidance (cautions, precautions, notes, and steps) for the station operators to properly operate the station's systems. (Sections O1.1, O1.2)

Examination Summary:

The licensee submitted a well prepared initial examination outline. The outline conformed with the guidelines contained in NUREG 1021, Operator Licensing Examination Standards for Power Reactors, Interim Revision 8, January 1997. The submitted operating test was acceptable without modification. (Section O5.2)

The high number of unsatisfactory questions (12) submitted to the NRC, the relatively high number of post examination comments (9), and the five hour validation time of the written examination indicated that the development of a written examination was an area that needed improvement. (Sections O5.2 and O5.4)

Report Details

I. Operations

O1 Conduct of Operations

O1.1 General Comments

a. Scope (IP 71707)

A Nuclear Regulatory Commission (NRC) examiner observed routine control room activities during full power operations for a two-hour period using Inspection Procedure 71707, Plant Operations. The examiner observed a routine surveillance, verbal communications, annunciator responses, and control room panel attentiveness.

b. Observations and Findings

The lead Reactor Operator (RO) conducted a routine surveillance with an instrument and control technician during the observed period. The surveillance was conducted without mishap. The control room crew engaged in routine face-to-face discussions and consistently used three-way communications. The ROs responded to several annunciators by announcing them to the Senior Reactor Operators and stating if they were expected or unexpected. The operators referenced the annunciator procedures for unexpected alarms. The crew generally had at least one RO per unit maintaining visual contact with the control boards.

c. Conclusions

Operators performed their shift responsibilities in a professional, business-like manner during the observed period.

O3 Operations Procedures and Documentation

03.1 General Comments

a. Scope (71707)

Using Inspection Procedure 71707, the examiners reviewed selected administrative and operations procedures during the initial license examination validation.

b. Observations and Findings

The examiners reviewed the procedures used to develop the operating test. The procedures were logically organized and contained adequate instructions, notes, precautions and cautions to direct the operators in the execution of their responsibilities.

c. Conclusions

Procedures reviewed by the examiners contained adequate guidance (cautions, precautions, notes, and steps) for the station operators to properly operate the station's systems.

O5 Operator Training and Qualification

O5.1 General Comments

Nuclear Regulator Commission examiners administered operator initial license examinations at the Prairie Island Nuclear Power Station to three RO applicants during the week of May 17, 1999. Two RO applicants successfully passed all sections of the initial license examination and were issued RO licenses. The remaining applicant passed the operating test, but failed the written examination and was denied an RO license.

Prairie Island training department instructors used the guidance prescribed in NUREG 1021, Operator Licensing Examination Standards for Power Reactors, Interim Rev. 8, January 1997, to prepare the operating test and written examination. The training staff administered the written examination and NRC examiners administered the operating test and observed portions of the administration of the written examination.

O5.2 Pre-Examination Activities

a. Scope

Nuclear Regulatory Commission examiners reviewed the examination material submitted by the training department's examination developers using the guidance prescribed in NUREG 1021.

b. Observations and Findings

1. Examination Outline:

The initial outline submittal was timely and developed in accordance with the quantitative and qualitative requirements of NUREG 1021, ES-201-2, "Examination Outline Quality Assurance Checklist." No changes were necessary to the submitted outline.

2. Initial Submittal:

Written Examination

The examiners reviewed the written examination and determined that twelve submitted questions were unsatisfactory. The twelve questions were not written in accordance with NUREG 1021, Section D.2.b, in that they contained significant psychometric flaws. Eleven questions were unsatisfactory because an applicant taking the test would not be able to determine the correct answer when given the question stem. The twelfth question was unsatisfactory because it was non-discriminatory. The examiners determined that the unsatisfactory questions could be replaced or corrected in a timely manner, which would prevent a delay in the administration of the written examination. The NRC examiners also requested that the station instructors modify several other questions to correct grammar errors.

One week prior to the on-site validation week, station examination developers informed the NRC chief examiner that the examination had been validated as a five hour examination. This was contrary to the guidance provided in NUREG 1021, Appendix E, Section B.3, which stated that the examination was to be a four hour examination. The NRC examiners disagreed with the assessment of the facility instructors and directed the examination proctor to administer the examination in accordance with the NUREG 1021 guidelines. All three applicants finished the examination within the allotted four hours.

Operating Examination

The administrative Job Performance Measures (JPMs), operating JPMs and dynamic simulator scenarios were all acceptable as submitted by the training department's staff.

c. Conclusions

The initial examination outline was well prepared. The high number of unsatisfactory questions (12) submitted to the NRC and the five hour validation time of the written examination indicated that the development of a written examination was an area that needed improvement. The submitted operating test was acceptable without modification.

O5.3 Examination Activities

a. Scope

The NRC examiners administered the operating test (JPMs and dynamic scenarios) during the week of May 17, 1999. Station instructors administered the written examination on May 21, 1999. The tests were administered using the guidance prescribed in sections ES-302 and ES-402 of NUREG 1021.

b. Observations and Findings

Job Performance Measures

The examiners noted that the applicants used good self-checking techniques. The licensee training staff coordinated the arrival times of the applicants and provided escorts to maintain examination security during the operating test. The licensee's

simulator staff was timely and accurate in their daily setup and execution of the operating test during the validation and examination weeks.

Dynamic Simulator Scenarios

The applicants performed well during the simulator scenarios. They displayed good self-checking and a thorough knowledge of control room operating requirements. Some individual and generic communication deficiencies were identified. These deficiencies included the failure of applicants to acknowledge important information from other crew members and, in one case, providing incomplete information to other crew members.

Written Examination

The licensee administered the written examination concurrent with the management exit meeting on the last day of the examination week. The examination administrator accurately followed the guidelines in NUREG 1021 while administering the written examination. The examination was completed in the allotted four hours.

c. Conclusions

The applicants were well prepared for the operating test. In general, they displayed good self-checking and communications practices during the operating test. The facility training staff was well prepared to support the examination process.

O5.4 Post Examination Activities

a. Examination Scope

The NRC examiners evaluated individual applicant performance on the operating test and reviewed the licensee's grading of the written examination. The examiners also reviewed post examination comments submitted by the licensee. Examiners followed the guidelines contained in sections ES-303, ES-403, and ES-501, of NUREG 1021.

b. Observations and Findings

Job Performance Measures

Two generic knowledge deficiencies were discovered while grading the candidates' performance of the operating JPMs:

- Applicants were unable to recognize a valid entry into the station's emergency diesel generator fuel oil tank level technical specifications. The applicants were given a set of diesel generator fuel oil tank levels obtained from the turbine building operator and asked to determine if the plant had sufficient fuel oil to meet the station's technical specifications. The expected answer to the question was that there was insufficient diesel fuel oil available and the shift supervisor needed to enter a technical specification for inadequate diesel fuel oil. All three applicants failed to subtract the unusable fuel oil from the tank levels and determined that there was adequate fuel oil.

- Applicants were unable to determine the correct bank overlap unit reading under certain plant conditions. The applicants were given control rod positions followed by a series of switch manipulations by the reactor operator. They were then asked to provide the bank overlap unit value. All three applicants responded that the value would be zero. The applicants appeared to understand how the equipment worked but lacked the knowledge needed to arrive at the correct answer.

Dynamic Simulator Scenarios

The NRC examiners did not discover any significant or generic weaknesses during the review of the operating test results.

Written Examination

There were nineteen questions that were answered incorrectly by more than 50% of the applicants. These questions were considered potential generic knowledge deficiencies and are provided to the Prairie Island training staff for consideration and implementation into the Systematic Approach to Training-based program. The examiners considered these generic deficiencies as potential because of the small sampling size (3 applicants).

Question #	Knowledge Weakness
#4	Required communications practices when acknowledging directions given to operators outside the control room.
#6	Maximum allowed deviation between steam generator pressures during normal operation.
#10	Application of ALARA principles.
#19	Operation of DC Hold power during normal conditions and during a reactor trip.
#32	Conditions that will generate annunciator 47013-0507, "COMPUTER ALARM ROD DEVIATION/SEQUENCING."
#35	Incore instrumentation system response to a 30°F rise in ambient temperature around the reference junction boxes.
#36	Indications used to monitor a reactor coolant system cooldown under natural circulation conditions.
#45	Control of steam generator pressure from the hot shutdown panel.

#51	Auxiliary feedwater system response to a trip of 12 main feedwater pump concurrent with a loss of buses 11 and 12.
#57	The response of the reactor coolant drain tank system to a system HI/LO tank level with pumps and valves in automatic.
#63	Response of the fire protection system when flow is initiated in the system with the 121 motor driven fire pump disabled.
#64	The status of the containment vacuum breakers following a manual initiation of safety injection.
#67	Expected RCS conditions as natural circulation develops following a loss of power to the reactor coolant pumps.
#82	Conditions that may lead to pressurized thermal shock during a steam generator tube rupture event.
#86	Results of a white bus (111) power failure.
#90	Response of the 12 charging pump under certain conditions when the control switch at the hot shutdown panel is taken to LOCAL.
#92	Effect of placing the cation demineralizer in service after a fuel pin begins leaking.
#98	The best method to collapse an apparent void in the RCS when a natural circulation cooldown is occurring.
#100	The basis for maintaining a minimum feed flow of 40 gpm to each steam generator following an uncontrolled depressurization of both steam generators.

The licensee submitted nine post examination comments which were reviewed by the NRC examiners. The licensee's comments and NRC resolution of the comments are detailed in Enclosure 2, "Facility Post Written Examination Comments and NRC Resolution." Six comments were accepted, resulting in four questions being deleted and answer key changes for two questions. The comments for the remaining three questions were not accepted.

c. Conclusions

The relatively high number of post examination comments (9) on the written examination indicated a deficiency in developing a written examination that conformed to the guidelines of NUREG 1021. The potential generic knowledge deficiencies from the operating test and written examination were provided as feedback for the licensee's training program.

O5.5 Simulator Fidelity

The simulator performed well during the validation week, during the operating JPM test and during the dynamic simulator scenarios with no noted deficiencies. This was documented in Enclosure 3, Simulation Facility Report.

V. Management Meetings

X1 Exit Meeting Summary

The chief examiner presented the examination team's observations and findings to members of the licensee's management on May 21, 1999. The licensee acknowledged the findings presented and indicated that no proprietary information had been identified during the examination or at the exit meeting.

PARTIAL LIST OF PERSONS CONTACTED

Licensee

T. Amundson, General Superintendent, Engineering
D. Cedergren, RO Class Coordinator
M. Gardzinski, Simulator Engineer
J. Lash, Instructor
B. Mather, Training Shift Manager
D. Schuelke, Plant Manager
J. Sorensen, NSP Site General Manager
D. Westphal, Operations Training Superintendent

NRC

S. Ray, Senior Resident Inspector, Prairie Island Nuclear Plant

INSPECTION PROCEDURES USED

IP 71707, “Plant Operations”

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

None

Closed

None

Discussed

None

LIST OF ACRONYMS USED

CFR	Code of Federal Regulations
DRS	Division of Reactor Safety
IP	Inspection Procedure
JPM	Job Performance Measure
NRC	Nuclear Regulatory Commission
OL	Operator Licensing
RO	Reactor Operator

Facility Post Written Examination Comments and NRC Resolution

1. Question #4

Which of the following describes the correct usage of repeat back when directing an APEO in the field to perform valve manipulations?

The APEO should...

a. provide verbatim repeat back of orders.
b. repeat back paraphrasing the orders.
c. repeat back only that portion of the orders that are NOT completely understood.
d. repeat back the orders only if the directions were provided over the phone or Plant Page system.

Answer: b.

Comment:

The question asks for the correct usage of repeat backs when directing the APEO to perform valve manipulations in the field. 5AWI 3.15.6, "Site Communications Standard" requires a paraphrased repeat back of communications of operational significance.

The question did not ask the minimum standard for communicating. The minimum standard is a paraphrased repeat-back. However, a verbatim repeat-back exceeds the requirements of a paraphrased wording. Additionally, it is a common practice to use verbatim repeat backs when acknowledging orders to position valves.

The facility recommends that both answers (a) and (b) be accepted as correct choices.

NRC Resolution:

The station's communications procedure, 5AWI 3.15.6, Section 6.1, **General Requirements**, stated that for operationally significant communications: "The receiver **SHALL acknowledge** receipt by a paraphrased repeat back of the received information." (Bold face words were bold face in the procedure.) The procedure had no mention or allowance for verbatim repeat backs. During the review of the written examination, licensee instructors rejected distractor a. as a correct answer by stating that, "Verbatim repetition is NOT desired since this may not indicate understanding of directions." The facility's instructors were unable to demonstrate that a. should be accepted as a correct answer. Answer choice b. was retained as the only correct answer. No change was made to the answer key.

2. Question #5

Throttle valve SI-15-7 was adjusted and requires independent verification (IV).

You may IV this valve by observing...

a. and counting the number of turns while CLOSING the valve fully and then REOPENING the valve the same number of turns independent of the initial verifier.

b. and counting the number of turns while this valve is OPENED fully and then RECLOSED the same number of turns by the initial verifier.

c. proper system flow through the valve.

d. proper stem position on the valve.

Answer: c.

Comment:

The question asks for the correct method of performing an independent verification of a specific valve. 5AWI 3.10.1, “Methods of Performing Independent Verification” provides generic guidance for throttled valves, however the specific method of independently verifying SI-15-7 is not done per any of the distracters.

The actual method of positioning and independent verification of this valve is completed under a work order issued by Engineering (last done under WO# 9600886) when the acceptance criteria of surveillance procedure (SP1092A) is not met. Once positioned, it is sealed and the operator independently verifies the position of this valve by ensuring a block and seal installed on the valve as shown on Checklist C1.1.18-1; p. 12 of 17. IV of this valve is an exception to 5AWI 3.10.1 and is controlled under the Work Order.

The answer key lists “proper system flow through the valve” as the answer. There is no flow measurement device, for a person independently verifying the position, to accomplish this (Flow diagrams X-HIAW-1-44 and 45 show only a total cold leg flow measurement as seen in the control room on flow indicator FI-925.). 5AWI 3.10.1 warns, per step 6.1.14.c, that “flow does not necessarily prove that flow is going to the correct location”.

The facility recommends that this question be deleted, as there is no correct answer.

NRC Resolution:

An originally submitted question was replaced during the examination validation week with this question. Based on the licensee’s procedures, it appeared that answer c. was the correct answer. The examiners reviewed the SI surveillance procedure performed under work order 9600886, and determined that this valve was not positioned using the generic guidance for independently verifying throttle valves. This valve was throttled closed from the full open position until the correct flow was achieved, and then it was blocked and tagged in that position. The flow instruments used to verify correct flow were subsequently removed and the IV only verified that the block and seal was installed. The comment was accepted. None of the answer choices was completely correct regarding the independent verification of valve SI-15-7. The question was deleted from the examination and the answer key amended.

3. Question #32

Which of the following sets of conditions will generate annunciator 47013-0507 "COMPUTER ALARM ROD DEVIATION/SEQUENCING"?

	Rods	Bank D Step counter	Rod G-3 RPI
a.	Moving	20 Steps	0 Steps
b.	Stationary	180 Steps	194 Steps
c.	Moving	190 Steps	204 Steps
d.	Stationary	228 Steps	214 Steps

Answer: b.

Comment:

The question asks about knowledge of annunciators. A recent Tech Spec revision (T.S. 3.10.F) required a change to the setpoint of the alarm. Previously, knowledge of whether the rods were moving or stationary was needed to ascertain when a rod deviation would occur.

The annunciator alarms, with the group step counter between 30 and 215, at a >12 step deviation. Below 30 and above 215 steps the alarm comes in with a >24 step deviation.

Recently revised Surveillance Procedure SP 1319 states that when the rod-to-bank deviation limit is exceeded. "RPI to bank demand difference greater than 12 steps with rod bank position between 30 and 215 steps." Because rod motion is no longer a concern, the choices for this question had three answers with a rod deviation of 14 steps. Two choices were in the 30 to 215 range. With multiple choices appearing to the candidates as the same answer (14 steps) they may have eliminated the choices because they were, as far as the annunciator goes, the same answer. Two candidates chose a greater deviation of 20 steps (selection a).

The facility recommends that this question be deleted from the test, due to the fact that the question was constructed with two selections being exactly the same.

NRC Resolution:

The licensee stated that because rod motion no longer has an input to this alarm, answers b. and c. were correct and essentially identical. The licensee agreed that distractors a. and d. were incorrect. The NRC's policy was to delete a question when there was three correct answers. Since there was only two correct answers, the question was a valid question. The facility's comment was rejected and the answer key was amended to accept both b. and c. as correct answers.

4. Question #40

Which of the following can be used as an indication of proper Hydrogen Recombiner operation following a LOCA event in which the containment hydrogen concentration was initially determined to be 2.8%? (Assume all other containment parameters remain constant for the period.)

a. Test Thermocouple temperatures will indicate a ramped decrease below 625°F with constant power input.

b. Test Thermocouple temperatures will indicate a ramped increase above 1225°F with constant power input.

c. Recombiner power output will increase from its initial setting without operation of the Pwr Adjust potentiometer.

d. Recombiner power output will decrease from its initial setting without operation of the Pwr Adjust potentiometer.

Answer: b.

Comment:

The answer key lists (b) as the correct choice. The given reference, C19.8 does state that a ramp change in temperature will occur. The problem with answer (b) is that it states that there will be a "ramped increase". A ramp change and a ramped increase are not the same. C19.8 Figure 2, shows that there is a ramp change as the temperature nears the recombination temperature of 1225°F. In other words, the ramp rate changes from a steep increase to a lesser rate at the higher temperature.

Surveillance Procedure SP 1255 substantiates this. Steps 7.1.6 through 7.1.12 have the operator heatup the recombiner. Step 7.1.12 states, "temperature should stabilize at approximately 1225°F within 3 to 4 hours." This is affirmed in step 7.1.13.

Because none of the choices adequately answer this question, the facility recommends that this question be deleted from the test.

NRC Resolution:

NRC examination reviewers agreed that there were no answers that correctly responded to the question stem. The post exam comment was accepted. The question was deleted from the examination and the answer key amended to reflect the deletion of the question.

5. Question #51

Given the following conditions on Unit 1:

- Reactor power 65%
- Control power to 12 Main FW pump was lost
- Immediately thereafter buses 11 and 12 were lost due to breaker faults
- Both SG NR levels decreased to 20%

What is the status of the AFW Pumps IMMEDIATELY following this event?

	11 TD AFW Pump	12 MD AFW Pump
a.	Stopped	Stopped
b.	Stopped	Running
c.	Running	Stopped
d.	Running	Running

Answer: c.

Comment:

This question asks about the cause-effect relationship between main and auxiliary feedwater. Due to a new diverse scram system (AMSAC/DSS), the response of the plant will be different.

The answer key lists (c.) as the answer, which is correct before the recent modification. However, AMSAC/DSS will actuate due to the loss of busses 11 and 12, which will trip both 11 and 12 RCPs. The RCPs will generate a protection signal which will start both AFW pumps.

The facility recommends changing the answer key from (c.) to (d.)

NRC Resolution:

The post exam comment was accepted. The answer key was amended to accept only d. as the correct answer.

6. Question #56

Given the following conditions on Unit 1:

- A release is in progress from the 121 ADT Monitor Tank using the programmable controller
- Power is subsequently lost to bus 13 and the bus is deenergized

What is the effect on the release? The release will...

a. continue as normal.

b. continue but requires control directly from the programmable controller using the programming panel or I&C laptop computer.

c. terminate due to loss of power to the control room instrument for R-18 radwaste liquid release radiation monitor.

d. terminate due to loss of power to the programmable controller which closes the release valve and stops the pumps.

Answer: d.

Facility Comment:

This question requires the operator to recognize how a liquid release is affected by a sustained loss of non-safeguards bus 13. The reference for this question, (C21.1.2, Section 3.0) makes a generic statement regarding various systems, which does not fully apply to this situation. C21.1.3.1 AOP1 states that only the valves will close.

The power supply to the programmable controller is MCC 3A1, which is powered from bus 13 via bus 310. The power supply to the ADT monitor tank pumps is MCC 1RW2 which is powered from bus 290, which in turn is powered from bus 24 (a unit 2 power supply). The loss of bus 13 will have no effect on the ADT monitor tank pumps. Additionally, the programmable controller only has control over the release valves and does not have control over the 121 ADT Monitor Tank Pump.

Upon losing power to bus 13, bus 310 will undergo a voltage restoration and be immediately repowered from bus 410 via the breaker 3141 bus tie. This restores power to the programmable controller. The answer key choice of (d) states that the programmable controller will lose power (which is temporary), however it will not have any effect on the "pumps" (ADT monitor tank pumps).

Checklist C21.1-5.1; step 5.1.4, shows that the pump will recirc with the release valve closed.

Confusion with the candidates was apparent, as one asked, "Will backup power supplies "kick in?" No clarification could be given during the exam. Because the correct answer has an incorrect statement, the facility recommends that this question be deleted from the test.

NRC Resolution:

The licensee's post exam comment was correct. The pumps would not stop. The post exam comment was accepted. The question was deleted and the answer key amended.

7. Question #73

Given the following conditions on Unit 1:

- The letdown heat exchanger is out of service
- Excess letdown is in service
- 11 Charging pump is running at minimum speed
- An ATWS has been diagnosed

Which of the following values is the highest boration rate that the RO would establish using the normal boration flowpath?

a. 20 gpm
b. 15 gpm
c. 12 gpm
d. 8 gpm

Answer: b.

Comment:

This question asks the operator to determine boration rate during an ATWS, given several plant conditions associated with the CVCS as stated above.

Based upon their answers, all students recognized the limit of allowing only 75% boric acid flow of total charging flow. The question asks what flow should be established, "given the following conditions." The conditions listed, have the running charging pump at minimum speed, and therefore the total charging flow is 16 gpm. With the plant in an abnormal lineup, the candidates did not know if 11 charging pump speed could be increased. The question stem does not suggest that the speed could be increased.

Although the basis for step 4 of FR-S.1 says, "it may be necessary to increase the speed controller of the charging pumps." it also states, "the operator should borate at the maximum rate available based upon current conditions..." The current conditions given in the question are such that initially the operator would establish 12 gpm boric acid flow (16 gpm x 75%).

The facility recommends changing the answer key from (b) to (c).

NRC Resolution:

The answer choice of b. assumed that the RO will increase charging pump speed to maximize the boration rate. However, there was no information in the question that would lead one to believe that charging pump speed could not be increased which would lead to the 12 gpm boric acid flow. The post exam comment was accepted. The answer key was amended to change the correct answer from b. to c.

8. Question #82

Which condition may lead to a Pressurized Thermal Shock condition in a steam generator tube rupture event?

a. SI flow to one cold leg is isolated.

b. SI to reactor vessel valves are opened.

c. The RCP trips in the loop with the ruptured SG

d. Both RCPs have stopped.

Answer: d.

Comment:

The question asks about conditions which may lead to a PTS condition. Selection (d) is true specifically for a SGTR event.

According to the SI system description (B-18A), the SI to reactor vessel valves are normally closed to prevent unnecessary thermal shock to the reactor vessel in the event of a spurious SI actuation. Opening of these valves with SI actuated (choice b) would therefore, lead to a PTS condition from the thermal shock/cooling to the reactor vessel. This action is outside the EOPs.

The facility recommends that both answers (b) and (d) be accepted as correct choices.

NRC Resolution:

The question specifically asked for a condition (answer) during a steam generator tube rupture. The SI vessel isolation valve concern (distractor d.) applied during a spurious SI actuation. The question stem did not include the condition of a spurious SI actuation. The comment has not provided sufficient information to require a change to the examination answer key. The post exam comment was not accepted, the answer key was not amended.

9. Question #85

Given the following conditions on Unit 1:

- A reactor trip has occurred due to a loss of offsite power.
- The actions of 1ES-0.1 “REACTOR TRIP RESPONSE” are being performed for verifying natural circulation flow.

Which of the following correctly explains why the SG levels are maintained in the normal post-trip control band during natural circulation?

a. Ensures RCS cooling remains symmetrical.

b. Prevents voiding from occurring in the reactor vessel head.

c. Ensures SG tubes are covered to verify natural circulation.

d. Prevents a complete loss of RCS flow due to voiding in a single loop.

Answer: a.

Comment:

The question asks the knowledge found in the basis for an EOP step regarding conditions for natural circulation.

The suggested answer does not answer the question. The question asks why levels are maintained in a band. The low end of the band is to ensure SG tube coverage and provides symmetric cooling if both SG levels are above the SG tubes. However, the high end of the band is not addressed. The high end of the control band is 40% which prevents having too much mass in the SG in case of a steamline break accident (see references C1.3; Figure C1-39; and USAR Section 14).

Because none of the choices answer the stem of the question, the facility recommends that this question be deleted from the test.

NRC Resolution:

The comment was accepted. The actual reason for maintaining both the high end and low end of SG level was not included in any of the answer choices. The question was deleted and the answer key amended.

Enclosure 3

SIMULATION FACILITY REPORT

Facility Licensee: Prairie Island Nuclear Station

Facility Licensee Docket Nos: 50-282; 50-306

Operating Tests Administered: May 17-20, 1999

The following documents observations made by the NRC examination team during the initial 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

1. None