U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Report No. 50-282/OL-92-01(DRS)

Docket Nos. 50-282; 50-306 Licenses No. DPR 42; DPR 60

Licensee: Northern States Power Company
414 Nicollet Mall
Minneapolis, MN 55401

Facility Name: Prairie Island Nuclear Generating Plant

Examination Administered At: Prairie Island Nuclear Generating Plant

Examination Conducted: Week of May 4, 1992

RIII Examiners: J. Lennartz 6/2/92 Date

C. Osterholtz 6/2/92 Date

Approved By: T. Burdick, Chief 6/2/92 Date
Operator Licensing Section 2

Examination Summary

Examination administered during the week of May 4, 1992 (Report No. 50-282/OL-92-01(DRS)). Initial written and operating examinations were administered to five reactor operator (RO) candidates, three senior reactor operator (SRO) upgrade candidates, and one senior reactor operator (SRO) instant candidate.

Results: All of the candidates passed the operating examination. One RO candidate failed the written examination; all the other candidates passed the written examination. During the administration of the simulator examinations some errant cues were given to the candidates by the training personnel operating the simulator. The apparent cause for some of these errant cues was having different persons operating the simulator than were utilized during scenario validation. This problem has occurred on previous NRC exams. See Report 50-282/OL-91-01(DRS).

REPORT DETAILS

1. Examiners

*C. Osterholtz, NRC
J. Lennartz, NRC
K. Parkinson, Sonalysts

*Chief Examiner

2. Exit Meeting

An exit meeting was held on May 8, 1992, between the NRC and licensee representatives to discuss the examiner observations as described in this report.

NRC representatives in attendance were:

M. Dupah, Senior Resident Inspector
J. Hansen, Examiner, Observer
J. Lennartz, Examiner
C. Osterholtz, Examiner

Licensee representatives in attendance were:

S. Gheen, Prairie Island Trainer
M. Hall, Prairie Island Trainer
M. Ladd, Prairie Island Trainer
M. Lawrence, Prairie Island Trainer
D. Reynolds, Prairie Island Operations Training Supervisor
M. Wadley, Prairie Island General Superintendent, Plant Operations
L. Waldinger, Director, Training Power Supply
T. Wellumson, Monticello Trainer
D. Westphal, Prairie Island Trainer

The licensee representatives acknowledged the examiner observations discussed in Section 3 of this report as well as the items identified in Enclosure 4, the Simulation Facility Report.

3. Examiner Observations

a. Examination Development

The licensee training staff provided the NRC excellent support during validation of simulator scenarios and job performance measures. In addition, the facility's pre-review of the written examination was very thorough and considered very valuable in the development of a plant specific valid examination. During this review it was identified that the facility system descriptions did not always accurately describe present plant configurations. This deficiency hindered the written examination development and review. Additionally, some minor procedural deficiencies were identified by the NRC examiners and provided to the facility. None of these deficiencies were considered safety significant.

b. Operating Examination Administration

During the administration of the operating examinations, the NRC examiners observed both strengths and deficiencies on the part of the senior reactor operator (SRO) and reactor operator (RO) candidates.

The following strengths were observed:

- The ability to effectively communicate information between crew members.
- The ability to utilize plant piping and instrumentation diagrams.
- The ability to utilize Annunciator Response guidance.

The following deficiencies were observed:

- Knowledge in the fundamentals of radiation theory, including shielding requirements to guard against neutron radiation.
- Relying only on verification of damper position to determine the status of the associated ventilation fan (running, not running), while responding to an abnormal radiation level during a waste gas release, rather than verifying actual fan status indications.
- Leaving pressurizer heaters energized during a loss of heat sink event which contributed to the unnecessary cycling of the pressurizer Power Operated Relief Valves (PORV).

During the dynamic scenario portion of the operating examinations, some errant cues were given to the candidates by the simulator operators. The following are specific examples of problems associated with simulator operation:

- During an RHR break with pressurizer level decreasing, no reports of steam or water being present in the RHR pit was provided to the candidates at the appropriate time as was discussed during scenario validation. This caused a delay in diagnosis of the casualty by the crew.

- The RHR to letdown isolation valve, MV 32234, was left closed for a scenario which required it to be open in accordance with procedure C15, step 5.1.13. This caused confusion among the candidates as to plant status.

- An incorrect IC was installed in the simulator for a scenario. This delayed scenario initiation for the Group 2 candidates.

The apparent cause for some of these difficulties was having different persons operating the simulator during the examinations than during scenario validation.

4. Written Examination Administration

The post examination review of the written examination by the NRC identified the following deficiencies in the candidates' knowledge as evidenced by the majority of the candidates failing to provide the correct response for each particular knowledge area examined. This information is being provided as input to the licensee's system approach to training (SAT) process:

- The amount of time that must elapse to ensure decay heat generated is less than 1% of rated power following 100 days of operation at 100% power. (SRO and RO Question 26)

- Identifying a charging piping leak using various CVCS system indications and component status. (SRO and RO Question 38)

- The technical specification basis for the minimum required level in the fuel oil storage tanks. (SRO and RO Question 67)

- The normal demineralizer/heat exchanger lineup during spent fuel pool cooling system operation. (SRO and RO Question 68)

- The electrical power sources to the instrument busses in order of priority. (SRO and RO Question 69)

- System indications (annunciators) upon placing the spare battery charger in service. (SRO and RO Question 70)

- The means to determine that the 21 motor driven cooling pump is properly primed. (SRO and RO Question 73)

- The basis for isolating seal leakoff following a complete failure of the No. 1 seal on an RCP. (RO Question 82)

5. Written Examination Review

Licensee representatives reviewed the written examination prior to administration with appropriate changes being incorporated into the examinations at that time. Following the administration of the written examinations, the facility was given a copy of the RO and SRO examinations and answer keys for review. The facility's post examination comments and the NRC resolutions are contained in Enclosure 2 of this report.

ENCLOSURE 2

Facility Comments and NRC Resolution of Comments

SRO and RO Question 54

While operating at 100% power, a rupture of the main feedwater system occurs inside containment upstream of the check valve. Which of the following initiates a main feedwater pump trip in response to this rupture?

a. Reactor trip initiated by a low-low steam generator level.

b. Reactor trip initiated by a steam flow/feed flow mismatch.

c. Low steam generator pressure safeguards actuation.

d. High containment pressure safeguards actuation.

ANSWER: d

REFERENCE: C7E, FW21

PI COMMENT/RECOMMENDATION:

During a feedwater rupture inside containment, containment pressure will increase, as suggested in answer d, and SI will actuate, tripping both feedwater pumps. However, steam generator level will decrease in the affected steam generator. If level reaches 13%, a reactor trip/turbine trip occurs, tripping one of the feedwater pumps.

The Cause and Effects document identified containment pressure as causing a reactor trip/SI for this particular malfunction. However, not all severities and locations of feedwater breaks are included, thus there may be certain severities or locations where the reactor trip occurs due to low-low-steam generator level. In those cases, one of the feedwater pumps will trip due to the turbine trip.

The latest Simulator Certification testing (4/92) for this malfunction shows that, at 100% severity, SI actuation due to containment pressure and steam generator low-low level occur within one second of each other.

Thus answers a. and d. are both correct.

NRC Resolution

Comment Accepted. The SRO and RO examination answer keys have been modified to indicate that answer a or d is correct.

SRO and RO Question 69

Which of the following describes the electrical power sources to the instrument busses in order of priority? (From most preferred to least preferred.)

a. 120 VAC, 480 VAC, 125 VDC
b. 120 VAC, 125 VDC, 480 VAC
c. 480 VAC, 125 VDC, 120 VAC
d. 480 VAC, 120 VAC, 125 VDC

ANSWER: c

REFERENCE: C20.8, PG 3, B20.8, PG 2

PI COMMENT/RECOMMENDATION:

The question is confusing in that it asks for the source of power to the instrument busses which is always 120 VAC. However, this 120 VAC can be supplied four (4) different ways:

1. From a 480 VAC MCC through a step down transformer to 120 VAC through an inverter.

2. From a 125 VDC panel through an inverter.

3. From a 480 VAC MCC through a step down transformer to 120 VAC through a static switch in the inverter.

4. From a 120 VAC panel.

Because the answers do not clarify which 480 or 120 VAC is being referred to, there is no clear correct answer to the question and the question should be deleted.

PI REFERENCE: Drawing ED-321

NRC Resolution:

The candidates that were confused by the question wording were allowed to ask for clarifications from the examination proctor. When the proctor was questioned as to where the sample point was, the proctor clarified that the sample points for the power sources in question were prior to the inverter input which makes choices "a" and "b" clearly wrong. Additionally, since 125 VDC is a higher priority power source than 120 VAC, the only correct response is choice "c". Therefore, this comment is not accepted.

Enclosure 4

SIMULATION FACILITY REPORT

Facility Licensee: Prairie Island Nuclear Generating Plant

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

Operating Tests Administered On: Week of May 4, 1992

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

ITEM	DESCRIPTION
1.	RHR pit rad monitors are not modeled for an RCS to RHR leak.
2.	The simulator locked up prior to the initiation of two different scenarios, causing a delay in scenario initiation.
3.	R-53, SI pump area radiation monitor, errantly alarmed when R-26, RHR cubicle air monitor, failed high.
4.	The simulator had an ERCS computer operator aid which was utilized during the examinations that is not available in the control room.
5.	Pressure indicator 135 did not change when pressure transmitter 135 failed low.