October 19, 2006

Mr. David A. Christian
Senior Vice President and
Chief Nuclear Officer
Innsbrook Technical Center
5000 Dominion Boulevard
Glen Allen, VA 23060-6711

SUBJECT: KEWAUNEE POWER STATION
NRC INITIAL LICENSE EXAMINATION REPORT 050000305/2006301(DRS)

Dear Mr. Christian:

On August 9, 2006, members of your Kewaunee Power Station Training Department administered an NRC approved initial operator licensing written re-examination to two license applicants that failed the November 2005 initial license written examination. The enclosed report documents the results of the re-take examination. An initial de-brief was conducted on August 28, 2006, by telephone with Mr. G. F. Winks, Kewaunee Power Station Training Manager and other members of your staff to acknowledge receipt of the station's post examination comments and to discuss a grading time-line for the examinations. A second de-brief was conducted by telephone with Mr. Winks and others of your staff on September 1, 2006, to clarify some post examination comments and other related information. A final exit was conducted by telephone with Mr. P. Short, Initial License Training Supervisor, on September 29, 2006, to convey final NRC disposition of post examination comments, explanation that the examination did not meet NRC examination submission expectations, and final grades for the two re-take initial license applicants.

Two Senior Reactor Operator applicants were administered license re-take written examinations. The results of the examinations were finalized on September 25, 2006. Neither applicant passed the re-take examination; both were sent proposed license denial letters.

The proposed written examination submittal by your training staff was considered outside the acceptable quality range expected by the NRC in accordance with NUREG-1021, "Operator Licensing Examination Standards for Power Reactors," Revision 9. This determination was based on the observation that several questions in the Senior Reactor Operator portion of the written examination required replacement or significant modification. Future examination submittals should incorporate any lessons learned from your evaluation of this examination and testing issues.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter and its enclosure will be available electronically for public inspection in the NRC Public Document Room, or from the Publicly Available Records (PARS) component of NRC's document system (ADAMS), 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 re-take examination.

Sincerely,

/RA/

Hironori Peterson, Chief
Operations Branch
Division of Reactor Safety

Docket No. 50-305
License No. DPR-43

Enclosures:
1. Operator Licensing Examination
   Report 050000305/2006301(DRS)
   w/Attachment: Supplemental Information
2. Written Examinations and Answer
   Keys (RO and SRO)
3. Post Examination Comments and
   NRC Resolutions

cc w/encl 1:
L. Hartz, Site Vice President
C. Funderburk, Director, Nuclear Licensing
  and Operations Support
T. Breene, Manager, Nuclear Licensing
L. Cuoco, Esq., Senior Counsel
D. Zellner, Chairman, Town of Carlton
J. Kitsembel, Public Service Commission of Wisconsin
State Liaison Officer, State of Wisconsin

cc w/encls 1, 2, and 3: G. F. Winks, Training Manager, Kewaunee Power Station

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

Sincerely,

/RA/

Hironori Peterson, Chief
Operations Branch
Division of Reactor Safety

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L. Cuoco, Esq., Senior Counsel
D. Zellner, Chairman, Town of Carlton
J. Kitsembel, Public Service Commission of Wisconsin
State Liaison Officer, State of Wisconsin

cc w/encls 1, 2, and 3: G. F. Winks, Training Manager, Kewaunee Power Station

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REGION III

Docket No:	50-305
License No:	DPR-43
Report No:	050000305/2006301(DRS)
Licensee:	Dominion Energy Kewaunee, Inc.
Facility:	Kewaunee Power Station
Location:	Kewaunee, WI
Date:	August 9, 2006
Examiner:	D. McNeil, Senior Operations Engineer
Approved by:	Hironori Peterson, Chief Operations Branch Division of Reactor Safety

SUMMARY OF FINDINGS

ER 050000305/2006301(DRS), 08/09/06, Dominion Energy Kewaunee, Inc, Kewaunee Power Station. Initial License Examination Report.

The announced operator licensing initial re-take written examination was administered by Kewaunee Power Station trainers in accordance with the guidance of NUREG-1021, "Operator Licensing Examination Standards for Power Reactors," Revision 9.

Examination Summary:

- Neither applicant passed the written re-take examination; both were issued proposed license denial letters. (Section 4OA5.1)

- The Senior Reactor Operator portion of the proposed examination required significant modification, deletion, or acceptance of 2 correct answers for 10 of 25 questions. This exceeded the NRC's expectation that less than 20 percent of the submitted examination questions would be unsatisfactory. (Section 4OA5.1)

Report Details

4. OTHER ACTIVITIES (OA)

4OA5 Other

.1 Initial Licensing Examinations

a. Examination Scope

Kewaunee Power Station's training staff prepared the examination outline and developed the written examination. The Nuclear Regulatory Commission (NRC) examiners validated the proposed examination during the week of July 24, 2006, and authorized the Kewaunee Power Station Training Department to administer the written re-take examination during the week of August 7, 2006. Kewaunee Power Station training department staff members administered the written re-take examination to two license applicants on August 9, 2006. In all cases, the training staff and the NRC examiners used the guidance established in NUREG-1021, "Operator Licensing Examination Standards for Power Reactors," Revision 9, to prepare, validate, revise, administer, and grade the examination.

b. Findings

Written Examination

The initial in-office review of the examination revealed that 7 of 25 questions on the Senior Reactor Operator portion of the submitted examination were considered unsatisfactory. After post examination comment changes were made to the examination, it was found that 3 additional questions on the Senior Reactor Operator portion of the examination were considered unsatisfactory. The 10 of 25 unsatisfactory questions exceeded the NRC's expectation of less than 20 percent of proposed questions being unsatisfactory. After post examination comment resolution, the Reactor Operator portion of the examination had a total of 13 unsatisfactory questions. Based on these results, the examiners determined that the examination, as submitted by the licensee, was not within the range of acceptability expected for a proposed examination.

The NRC examiners graded the written examination on September 25, 2006, and conducted a review of each missed question to determine the accuracy and validity of the examination questions. The licensee submitted 13 post-examination comments for the written examination. The comments and the NRC's resolution of the comments is enclosed with this report (Enclosure 3, Post Examination Comments and NRC Resolutions).

Examination Results

Neither applicant passed the written re-take examination; both were issued proposed license denial letters.

.2 Examination Security

a. Scope

The NRC examiners reviewed and observed the licensee's implementation of examination security requirements during the examination preparation. The examiners used the guidelines provided in NUREG 1021 to determine acceptability of the licensee's examination security activities.

b. Findings

No findings of significance were identified.

4OA6 Meetings

Exit Meeting

The chief examiner presented the process for post examination comments, the examination grading process, and findings on August 28, 2006, to Mr. G. F. Winks, Kewaunee Power Station Training Manager and other members of the Kewaunee Power Station Training Department staff in a de-brief via telephone. No proprietary or sensitive information was identified during the examination.

A second de-brief was conducted by telephone with Mr. Winks and other members of the training staff on September 1, 2006, to acknowledge receipt of the post examination comments and clarify some of the comments.

A final exit was conducted with Mr. P. Short, Initial License Training Supervisor, via telephone on September 29, 2006. The final disposition of post examination comments and final grades were discussed. Mr. Short was also informed that the submitted written examination did not meet NRC expectations for a facility-written examination.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee
S. Johnson, Examination Author
P. Short, Initial License Training Supervisor
G. Winks, Operations Training Manager

NRC
S. Burton, SRI, Kewaunee Power Station

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened, Closed, and Discussed

None

LIST OF DOCUMENTS REVIEWED

None

LIST OF ACRONYMS USED

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

WRITTEN EXAMINATIONS AND ANSWER KEYS (RO/SRO)

RO/SRO Initial Examination ADAMS Accession # ML062910307

Enclosure 2

Post Examination Comments and Resolutions

Question #8:

- The plant is operating at 100% power.
- Annunciator 47024-B, ACCUMULATOR A LEVEL HIGH/LOW alarms.
- Accumulator A level is 25%.
- A-SI-33, Abnormal Safety Injection Accumulator Level and Pressure, is implemented.
- SI Pump A is started to fill Accumulator A per A-SI-33.
- SI-101A/ CV-31247, SI Pump Makeup to Accumulator A is opened.
- A Design Basis LOCA occurs.

What is the postulated effect of these conditions on ECCS operation?

a. SI Pump A becomes gas bound from the filling operation and fails.

b. Cold leg injection flow is decreased due to the filling operation.

c. Accumulator A does NOT inject because it is isolated during the filling operation.

d. SI Pump A reaches runout conditions.

ANSWER
d.

Examinee Feedback:

Answer “b. Cold leg injection flow is decreased due to the filling operation.” is also correct. Cold leg flow would decrease while the Accumulator fills. When opening a parallel flow path, flow in the first flow path must go down.

The supplied Engineering Calculation/Evaluation also concludes that, “Pump run out should not be a concern for parallel pump operation if a Large Break LOCA were to occur during accumulator fill.”

Training/Operations Proposed Resolution:

The question is correct as written.

The calculation is correct with two SI Pumps running, however, the question premise provided the following condition, “A Design Basis LOCA occurs.” The question first gives that SI Pump A is running. USAR Section 14.3.2 describes the “Major RCS Pipe Ruptures (LOCA),” and subsection “Performance Criteria for ECCS” describes the ECCS systems response. On page 14.3-8 the following information is provided to define the design basis LOCA.

For the Best-Estimate large break LOCA analysis, one ECCS train, including one high head safety injection (HHSI) pump and one RHR (low-head) pump, starts and delivers flow through the injection lines. One branch of the HHSI injection line spills to the containment backpressure; the other branch connects to the intact loop cold leg accumulator line. The RHR injection line connects directly into the upper plenum. Both emergency diesel generators (EDGs) are assumed to start in the modeling of the containment fan coolers and spray pumps. Modeling full containment heat removal systems operation is required by Branch Technical Position CSB 6-1 (Reference 14) and is conservative for the large break LOCA.

The Technical Specification basis also supports the question as written. TS 3.3.b.5 provides protection from the possibility of one SI pump reaching runout condition during SI accumulator fill concurrent with a large break LOCA. CAP 028956 documents the reason the Step 4.3.5.f and h, for entering and exiting a 1 hour LCO to fill an SI Accumulator was added to procedure A-SI-33, Abnormal Safety Injection Accumulator Level and Pressure. This CAP references LER 97-004, which was submitted to report the existence of an unanalyzed condition during SI Accumulator filling operations. Specifically, if a LOCA occurred while filling an SI Accumulator, applying single-failure criteria to the Safety Injection pumps could result in only one SI pump delivering flow to both the RCS and the accumulator being filled, and a single SI pump providing flow to both the RCS and an accumulator could be subject to runout, which would result in no operable SI pumps during the accident.

Concerning the reduction in SI flow, the amount of water being injected to the RCS would be reduced by the amount being diverted to the accumulator. However, under the design basis accident, RCS pressure would be at Containment atmospheric pressure. With a maximum value of 46 psig for this pressure, the water flowing to the Accumulator would still be directed to the Cold Leg via the Accumulator outlet. If the flow indication (FI925) were assumed to the operator to be observed indication of Cold Leg Injection flow, then it would rise since the Accumulator fill line is downstream of the flow indicator.

<u>NRC Resolution</u>:

During the initial review of this question before the examination was administered, the NRC reviewer believed that there was adequate material in the question for an operator to determine distractor “d.” was the correct answer if they understood the complete definition of “Design Basis LOCA.” This answer was supported by the basis for a station technical specification limiting the amount of time that Safety Injection pumps could be aligned to fill the SI tanks. The basis stated that if a large break LOCA (Loss of Coolant Accident) occurred concurrent with the SI pump aligned to fill the SI tank, the SI pump would go into runout (distractor “d.”). However, the applicant also has a valid contention that distractor “b.” is correct. If water flow from the SI pump is going into the SI tank fill line, then cold leg injection flow must be less than it would have been if all of the injected water was going into the cold leg. If the water coming from the SI pump entering the cold leg through the SI tank injection line is not counted as “cold leg injection,” then distractor “b.” is correct. If the flow from the SI pump entering the cold leg through the SI tank injection line is counted as “cold leg injection,” then distractor “b.” is incorrect.

However, while reviewing this question, it was found that the term “Design Basis LOCA” was not defined for the applicant in the question stem and the applicant could make assumptions that would make both answers incorrect. Additionally, the term “Design Basis LOCA” is not defined in the station’s updated final safety analysis report. In the facility’s USAR it lists two types of reactor coolant system pipe ruptures as Condition III – Infrequent Faults, which comprises events involving small breaks, and Condition IV – Limiting Faults, which comprises events involving large breaks.

Section 14.3.1 of the USAR discussed analysis of the small break LOCA. The small break LOCA of concern is the 3-inch diameter break which results in most limiting break size in terms of highest peak clad temperature. Additional break sizes of 2-inch and 4-inch were also analyzed.

Section 14.3.2 of the USAR discussed the analysis specified in 10 CFR 50.46, Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors. The USAR presents this as a “Best-Estimate large break LOCA analysis.” The “LOCA Transient Analysis” identifies that the Cold Leg Split Break (70% of the area of the cold leg) is more limiting than the double-ended cold leg guillotine break.

The USAR does not refer to the “Best-Estimate” events as “Design Basis” events. Because of the lack of a definition of the term “Design Basis LOCA” in the question stem and in the facility’s updated final safety analysis report, the question was deleted from the examination and the answer key modified to remove the question from the examination.

Question # 19:

Given the following:

- The plant is operating at 100% power.
- A Reactor Trip has occurred.
- Safety Injection has actuated.
- The crew is implementing E-0, Reactor Trip or Safety Injection, Step 5 Verify Feedwater Isolation, when it is discovered that both the RED and GREEN Position Indicating Lights for FW-12B, Feedwater to Steam Generator B Isolation Valve are NOT lit on Mechanical Control Console A.

What action is taken at this time to verify that Feedwater Isolation is complete?

a. Dispatch operator to locally close FW-12B.

b. Check FW-12B status light DIM on SI Ready Status Panel.

c. Check FW-12B status light BRIGHT on CI Active Status Panel.

d. Check FW-12B status light BRIGHT on SI Active Status Panel.

ANSWER
d.

<u>Examinee Feedback</u>:

There is no correct answer. E-O Step 5 does not direct you to check the SI ACTIVE Status Panel at this point. Step 17 has you check these items, and has you manually align those items.

<u>Training/Operations Proposed Resolution</u>:

The question is correct as written.

The use of alternate indications for status verification is supported by Site and Operations standards.

Dominion Nuclear Operations Standard, DNOS - 0302, Control Board Monitoring, provides the following Expectation, "Operators monitor control board indications closely to detect problem situations early." The Standard states, in part, "The OATC ("operator at the controls") shall use all direct and alternative indications for verification of status."

GNP-03.30.02, Conduct of Operations, Revision E, also supports the above by reference to the expectations and standards for control board monitoring contained in DNOS - 0302.

Step 5, sub-step c states, "FW-12A and B, Feedwater To Steam Generator A(B) Isolation Valves - CLOSED. The Contingency Action sub-step c states, "Manually close the valves." BKG E-O describes actions as, Determine if the valves are closed and close valves as necessary. Also the instrumentation given is "Position indications for" these valves.

<u>NRC Resolution</u>:

The applicant contended that there was no correct answer to this question. He stated that E-O Step 5 does not direct you to check the SI ACTIVE Status Panel, but that Step 17 has you check these items, and has you manually align those items at that time. However, the stem of the question stated that when the crew was implementing E-0, Reactor Trip or Safety Injection, Step 5 Verify Feedwater Isolation, they discovered that both the RED and GREEN Position Indicating Lights for FW-12B, Feedwater to Steam Generator B Isolation Valve were NOT lit on Mechanical Control Console A. Whether the item was discovered at Step 5 or Step 17 is not the question. According to the question stem the valve position was discovered while executing Step 5, and then asks, “What action is taken at this time to verify that Feedwater Isolation is complete?” The answer is provided in Step 5, sub-step c, which stated, "FW-12A and B, Feedwater To Steam Generator A(B) Isolation Valves - CLOSED. Then the Contingency Action for sub-step c stated, "Manually close the valves." In further support of the answer, the BKG E-O described the response actions as: Determine if the valves are closed and close valves as necessary. The NRC agrees with the facility’s proposed resolution that the operator should have taken the contingency action described in distractor “d.” (correct answer). No change was made to the answer key; distractor “d.” was retained as the only correct answer.

Question Number 25:

Given the following:

- Core Burnup is 9000 MWD/MTU.
- RCS boron concentration is 740 PPM.
- The Reactivity Placard for BORATE reads "1 PPM 2.7 gallons boron/ppm."

Which of the following conditions will require the LARGEST emergency boration (most gallons of boric acid)? (Reactor Data Manual Figures RD-6.6 and RD-6.7 provided)

a. Plant at Hot Shutdown, RCS Tavg is 540°F and rapidly decreasing, Annunciator 47061-B, S/G A SF>FF, is lit.

b. Reactor is critical at $1X10^4$ CPS, Control Bank C is at 65 steps, Annunciator 47042-R, CONTROL BANK LOW LOW LIMIT, is lit.

c. Reactor is tripped, and Control Rod G-3 Rod Bottom Light is dark and its associated IRPI indicator is NOT indicating.

d. Reactor is tripped, and Annunciator 47101-J, BUS 62 LOCKOUT, is lit.

ANSWER
a.

<u>Examinee Feedback</u>:

Accept “D” as the correct answer. Answer “A” will cause an SI thus emergency boration will not be required. That leaves answer “D” as the correct answer.

<u>Training/Operations Proposed Resolution</u>:

Accept either answer “A” or “D” as the correct answer.

Not enough information in either the stem or selection "A" to preclude the assumption that a Safety Injection is NOT actuated due to the condition. Based upon the given information, Selection "A" with given indications could result in Safety Injection actuation on low Przr pressure (less than 1830 psig) or low steamline pressure (less than 514 psig). This would provide the necessary boration via SI Pump injection into the RCS, rather than by direction of ECVC-35. The progression through the IPEOPs will be from E-O, Reactor Trip and Safety Injection, to E-2, Faulted Steam Generator Isolation, if the SG pressure is still dropping at Step 23 of E-O; or to ES-1.1, SI Termination, at Step 26 of E-O if the blowdown of the SG is complete or stopped by the time Step 23 of E-O is reached.

In E-2, Faulted Steam Generator Isolation, steps will be taken to isolate the affected SG and transition would be made to E-1, Loss of Reactor Or Secondary Coolant, at Step 9 of E-2. In E1, Loss of Reactor Or Secondary Coolant, transition to ES-1.1, SI

Termination, would be made at Step 12, if conditions supported, or by direction of E-1 QRF (Quick Reference Foldout), Item 1, SI Termination Criteria. If conditions did not support SI termination, then the transition would be made to ES-1.2, Post LOCA Cooldown And Depressurization, at Step 18.b of E-1.

In ES-1.2, Post LOCA Cooldown And Depressurization, charging pumps are started and aligned to the RWST, and then at Step 6 RCS cooldown to Cold Shutdown is initiated. In NOTE 1 prior to this step reads,

> NOTE: During the RCS cooldown, RCS Boron Concentration should be monitored to verify Cold Shutdown Boron Concentration per RD-6.7.

ES-1.1, SI Termination, also directs starting of charging pumps, but directs establishing normal charging and letdown. ES-1.1 Step 12 has the operator set makeup for automatic control with the Boric Acid Controller set to 11.0. The operator is directed in ES-1.1 Step 25 to establish stable plant conditions, which includes maintaining RCS temperature at existing value. This would place the plant in INTERMEDIATE SHUTDOWN. ES-1.1 then directs the operator to go to the appropriate Plant Procedure, which would be N-O-05, Plant Cooldown from Hot Shutdown to Cold Shutdown Condition. N-O-05, Step 4.1.1 and 4.1.2 direct the operator to determine the Cold Shutdown boron concentration and borate the RCS to this value. However, in both circumstances, the boration is performed using the normal boration controls for the Reactor Makeup Control System. Therefore, emergency boration is not used.

Selection "D" is definitive in its conditions and requires an emergency boration to the Hot Shutdown boron concentration. This emergency boration is the largest amount required (648 gallons) from the remaining selections.

Selection "A" remains an acceptable answer if it is assumed an SI does not occur. This is plausible if based on conditions as allowed in procedures (N-RC-36C, Steps 4.2.7.8.b and 4.4.3) when RCS pressure is below 2000 psig and Safety Injection is blocked (Pressurizer SI Blocked). In this case SI would not occur and the appropriate action would be emergency boration. This emergency boration would require 2416 gallons to reach the Cold Shutdown boron concentration.

<u>NRC Resolution</u>:

In the rules for taking the written examination the applicants were told, “When answering a question, do not make assumptions regarding conditions that are not specified in the question unless they occur as a consequence of other conditions that are stated in the question. For example, you should not assume that any alarm has activated unless the question so states or the alarm is expected to activate as a result of the conditions that are stated in the question. Similarly, you should assume that no operator actions have been taken, unless the stem of the question or the answer choices specifically state otherwise.” The stem of the question asks which of the distractors would require the largest emergency boration and does not specify that an operator blocked SI. Under distractor “a.”’s conditions, procedures call for boration to 1% Cold Shutdown concentration. This was calculated to be 2416.5 gallons of borated water. An alternate definition of emergency boration is found in E-CVC-35, Emergency Boration, Step 4.2, where one of the alternate emergency boration paths is described as follows: “IF RCS pressure is less than 2211 psig, THEN Safety Injection Pumps can be used to inject boric acid from the RWST to RCS.” Step 4.4 then describes the conditions to be met for stopping the boration flow, including specific values for volume added from the RWST. Since safety injection is considered a form of emergency boration, the boron injected by the Safety Injection pumps counts toward the total gallons of boric acid added to the reactor coolant system. Since the requirement exists to emergency borate, and SI is considered a form of emergency boration, the NRC believes distractor “a.” was the only correct answer. Additionally, the phrase “most gallons of boric acid” was inserted into the stem to ensure applicants understood the intent of the question. Since no clarifying questions were asked concerning the intent of this question the NRC must assume the applicants understood the intent of the question. Because distractor “a.” required the most gallons of borated water be injected and SI is an alternate form of emergency boration, the NRC made no changes to the answer key for question number 25. Distractor “a.” was retained as the only correct answer.

Question Number 52:

Given the following:

- The plant is operating at 100% power.
- RCS activity level is normal.
- A 10 gpm tube leak occurs in S/G B.

Which process radiation monitors will indicate the release of activity from S/G B to the environment?

a. R-33 Steam Line B , R-15 Air Ejector Exhaust, R-14 Aux Bldg Vent Exhaust, R-36 Aux Bldg Vent Stack.

b. R-34 Steam Line B, R-43 S/G B N-16 monitor, R-15 Air Ejector Exhaust, R-35 Aux Bldg Vent Stack.

c. R-33 Steam Line B, R-43 S/G B N-16 monitor, R-15 Air Ejector Exhaust, R-13 Aux Bldg Vent Exhaust.

d. R-34 Steam Line B, R-15 Air Ejector Exhaust, R-19 S/G Blowdown Liquid, R-13 Aux Bldg Vent Exhaust.

ANSWER
c.

<u>Examinee Feedback</u>:

All of the given monitors will indicate the release of activity from the "B" steam generator.

<u>Training/Operations Proposed Resolution</u>:

The question is correct as written.

In "A," R-36 is a high-range monitor for the Aux Building Stack; In "B" and "D," R-34 is a high-range monitor for the steam line from SG B.

Both R-34 and R-36 have a sensitivity of 1 R/hr with an indicated range of 1 R/hr to 10,000 R/hr. With a leak of 10 gpm, the expected radiation indication would range about several mR/hr at the main steamline. This is about one one-hundredth the sensitivity of the detectors and would not be expected to be displayed for the monitors.

<u>NRC Resolution</u>:

Upon review of the station's Integrated Logic Diagram Radiation Monitoring (e-2021, Revision AA) it was determined that R-34 and R-36 have a sensitivity of 1 R/hr with an indicated range of 1 R/hr to 10,000 R/hr. Because of this high sensitivity, the small RCS tube leak would not be detected by these radiation monitors. The NRC agreed with the facility's contention that distractors “a.,” “b.”, and “d.” are incorrect. The answer key was not modified.

Question Number 55:

Given the following:

- The plant is operating at 100% power.
- IA-101, Instrument Air to Containment Isol, inadvertently closes due to a failure.
- The GREEN indicating light bulb for IA-101 position indication is burned out, so the CLOSED position is NOT indicated.

What will alert the operator to this failure?

a. Charging flow decreases to ZERO due to CVC-11 failing closed.

b. Reactor Trip on Low Pressurizer Pressure.

c. Annunciator 47051-I, STATION AND INSTR AIR SYSTEM FAULT.

d. Low pressure is indicated on PI-4150103, Rx Bldg Header Pressure indicator.

ANSWER
d.

Examinee Feedback:

Answer "A" is also correct. No time frame was given for the question. CVC-11 fails closes.

Training/Operations Proposed Resolution:

Accept either "A” or "D" as correct.

The question does not provide the time frame for selecting the answer. CVC-11 does have an air accumulator in its supply line and will remain in its current position for a period of time. However, if air is not restored the accumulator will become depleted and CVC-11 will fail closed.

The line for CVC-11 is a 2" line and the bypass line containing CVC-13 and CVC-14 is a 3/4" line. The purpose of CVC-14 is to provide a relief path due to differential pressure in the line upstream and downstream of the regenerative heat exchanger when CVC-11 is closed. The reduction in flow when CVC-11 closes would be apparent to the operator. Running this condition on the simulator shows that the bypass line only passes approximately 0.6 gpm for FI128 (node point measurement on bypass line). This is below the accuracy of FI-128 meter, so for the operator flow goes to zero. The seal injection pathway flow provides the other flow path for charging.

NRC Resolution:

Low pressure on PI-4150103, Rx Bldg Header Pressure indicator (distractor “d.”) would be the first indication of the failure of the indicating light. The accumulator associated with CVC-11 would maintain the valve’s position for approximately 4 hours before there would be inadequate gas to keep the valve open and when the valve fails shut, distractor “a.” becomes a correct answer. Since the question stem did not provide a time limit nor ask which one occurs first, the NRC agreed with the applicant’s contention that there are two correct answers to this question. The answer key was modified to accept distractors “a.” and “d.” as correct answers.

Question Number 75:

Given the following:

- The plant was operating at 100% power.
- A LOCA inside containment has occurred.
- E-1, Loss of Reactor or Secondary Coolant is in progress.
- Containment pressure is 2.5 psig.
- Containment Spray Pumps have been stopped.
- Containment radiation is 13 R/hr.
- Decision is made to transition to FR-Z.3, Response to Containment High Radiation Level.

What action should be taken to address the high containment radiation?

a. Verify RHR is supplying containment spray to reduce containment pressure to atmospheric.

b. Start Containment Spray Pumps to provide Iodine scrubbing of containment atmosphere.

c. Start Post-LOCA Hydrogen Control System to dilute containment airborne radioactivity.

d. Start Containment Fancoil Units to provide some filtration of containment particulate radioactivity.

Answer:
d.

<u>Examinee Feedback</u>:

No correct answer. In the given conditions, SI has occurred. This starts all 4 fan coil units. No indication that any have been secured. Step 2 of FR-Z.3 has you "verify" fan coil units are running. The answer is "yes," thus you do not "Start containment fan coil units..." They are running.

<u>Training/Operations Proposed Resolution</u>:

The question is correct as written.

The premise does not provide the status of the Containment Fancoil units; however, there is nothing that precludes the units from being stopped or not running at this point in the procedure. The correct action is to verify they are running and if not start the Fancoil Units. If the situation was that the Fancoil Units are always running after SI and continued to run at this point, there would be no need for the Contingency Action statement. The correct action is to start the Fan Coil Units for the purpose provided.

NRC Resolution:

In FR-Z.3, Step 2.a, the operator is directed to check Fan Coil Units running and if not, start the Fan Coil Units. The step was written this way to cover all possible plant conditions in which one could enter FR-Z.3 (including a normal Rx trip with no SI). With high containment radiation, or events in which SI activated, or in which the Rad in containment occurs a long way into the event, some of the Fan Coils may have been stopped by normal progression through the emergency procedures. However, for the conditions in the stem of this question, there is a LOCA and as such the Fan Coil Units should already be running due to the SI. One would have to assume that Fan Coil Units were shutdown or tripped (even though containment pressure is high at 2.5 psig) for distractor “d.” to be correct. Since the applicant is not allowed by examination rules to make such an assumption, the applicant’s correct action would be to “verify” the Fan Coil Units are running and if not, “start” the Fan Coil Units. Since no distractor required this action, there was no correct answer provided for the question. The answer key was modified to delete question # 75 from the examination.

Question Number 78:

Given the following:

- The plant is operating at 100% power with normal conditions established.
- A 138KV Transmission line failure causes a turbine load reduction.
- Bus 6 lockout also occurs just as the load reduction begins.
- Plant conditions following the transient are:
  - Reactor power is 97%.
  - Control Bank D position is 198 steps.
  - Control Rod Bank Selector is in MANUAL.

What actions are required based on the given plant conditions following the transient?

(Figure RD 4.1 and Technical Specifications pages 3.10-5 and 3.10-6 provided.)

a. Restore Control Rods above LOW-LOW insertion limit within 2 hours.

b. Verify position of control rods using incore moveable detectors within 4 hours.

c. Trip the reactor and emergency borate 360 gallons for each non-indicating control rod.

d. Restore inoperable IRPI indications within 12 hours or reduce power to less than 50%.

ANSWER
b.

<u>Examinee Feedback</u>:

The question does not take into consideration the full effects of the Bus 6 lockout outside of Bus 62. On a bus 6 lockout from 100% power, the secondary plant transient would lead to a condition that would require a reactor trip to place the unit in a safe condition. The loss of power to Bus 6 has various effects but most notably at 100% power are the loss of both Heater Drain Pumps and the loss of Service Water Train B.

Both Heater Drain Pumps are lost due to de-energization of MCC-62C and MCC-62E, leading to the loss of power to the Heater Drain Pump magnetic couplings and de-coupling. Per A-CD-03, the loss of two Heater Drain Pumps requires turbine impulse pressure to be reduced to less than or equal to 425 psig or approximately 70% using VPL Lower to maintain adequate suction pressure to the Main Feedwater Pumps. This is not accounted for in the question considering the stem has the plant at 97% following the transient which is not possible based on the previous discussion.

Additional consideration needs to be made based on the loss of Service Water. The Bus 6 lockout causes Service Water Pumps 1B1 and 1B2 to be lost due to the loss of power. Header pressure would be expected to drop to less than 72 psig in the B header quickly closing SW-3B if not already closed (in-plant condition). With the Turbine

Building SW Selector Switch in the B position, all Service Water flow would be lost to the Turbine Building. Temperatures on various equipment and components such as Feedwater and Condensate Pump bearings and slot gas temperatures would rise very quickly. Temperature limits would be exceeding requiring equipment stopped/tripped within a matter of minutes including a trip of the main turbine. At the onset of the transient the following procedures need to be addressed A-CRD-49, A-EHV-39, ASW-02, A-CD-03, and various other abnormals associated with high temperatures on equipment in the Turbine Building. During review and performance of the listed procedures trip criteria would be met prior to opening of SW-4A in A-SW-02 which is late in the procedural actions.

Therefore, based on the above discussion, the loss of Bus 6 puts the unit in a transient that is more severe than is stated in the question. Based on actual plant performance and response, the trip of the unit is the correct conservative response to make given this situation.

Answer C is correct.

Training/Operations Proposed Resolution:

There is no correct answer.

The question is to be deleted from the exam.

The loss of the Heater Drain Pumps was not addressed in development, review or validation of the examination, only the affect on IRPIs (rod indication). The action is correct to reduce load in accordance with A-CD-03, Condensate System Abnormal Operation. The plant would not be stabilized at 97% power; therefore, the premise of the question is invalid. This makes the question incorrect. As stated above, A-CD-03 requires with no Heater Drain Pump running that turbine load be reduced to less than or equal to 425 psig impulse pressure, or approximately 72% power.

The loss of Service Water is not considered since the question does not specify the Turbine Building SW Header is selected to either train. Assumption that the Turbine Building SW Header is aligned to Train B is not prudent since the header is just as likely to be aligned to Train A. This should have been addressed during the examination as a question on alignment of SW.

"C" cannot be a correct answer. Even if the reactor is tripped due to the given conditions, the emergency boration of 360 gallons for each control rod is not required. This makes the selection also incorrect.

NRC Resolution:

Distractor “a.” is incorrect because of the loss of Bus 6 will cause the LOW-LOW LIMIT annunciator - the control rods are above the insertion limit. The NRC does not agree with the applicant that distractor “c.” is a correct answer. This was a past procedural step from E-CDC-35 that has been discontinued and is no longer required at Prairie

Island Nuclear Generating Plant. This step was deleted because on the loss of individual rod position indication, the boration at 360 gallons per rod for all rods would result in a boron concentration significantly above hot shutdown boron concentration. The practice was discontinued and boration to hot shutdown concentration was inserted to limit the amount of boric acid added to the RCS. Distractor “d.” is incorrect as it is not a procedural or technical specification requirement. Concerning distractor “b.” - during validation of this question, the NRC reviewer discussed the short and long-term effects of the Bus 6 lockout with the question author. The question author stated that the intent of the question was to assess applicant knowledge of what to do under the given plant conditions with Control Bank D position at 198 steps, not determine what to do with the power plant. The selected K/A referred to operator actions with a loss of individual rod position indication. Assessing only the need concerning rod position with the given plant conditions leaves distractor “b.” as a correct answer. However, after a thorough review of the question the NRC has determined that several flaws exist with the question. Under the conditions stated in the stem, reactor power should have been greater than 100% after the transient because of the loss of heater drain pump flow. Reactor power at 97% would require operator intervention which is not stated in the stem of the question. Further, an applicant’s attention would be drawn to the immediate actions to resolve the plant conditions and reduce reactor power below 72% in accordance with procedures, not evaluate a technical specification that has a 4-hour time limit. Because the stem of the question does not provide realistic plant conditions, the NRC agreed with the facility resolution and the answer key was modified to delete question #78 from the examination.

Question Number 80:

Given the following:

- Reactor Trip and Safety Injection have occurred.
- A LOCA has occurred.
- RHR Pump A has failed.
- ES-1.3, Transfer to Containment Sump Recirculation is being implemented.
- CC-400B, Component Cooling to RHR HX B will NOT open.
- Efforts are in progress to locally open CC-400B.
- It is estimated that repair of CC-400B will take 90 minutes.

What is the impact of these conditions upon establishing Containment Sump Recirculation and upon removal of Core Decay Heat?

a. Containment Sump Recirculation can be established without CCW.
Core decay heat is transferred to the Containment and will be removed by Containment Fan Coil Units.

b. Containment Sump Recirculation can be established without CCW.
Transition to Core Cooling FR Procedures to provide Core Cooling.

c. Containment Sump Recirculation can NOT be established without CCW.
Place RHR in service per A-RHR-34B, Residual Heat Removal Split-Train Mode to provide Core Cooling.

d. Containment Sump Recirculation can NOT be established without CCW.
Transition to ECA-1.1, Loss of Emergency Coolant Recirculation to provide Core Cooling.

ANSWER
a.

Examinee Feedback:

Not enough information. If the "recirculation fluid temperature and pressure" exceed design conditions, then answer "D" is correct.

Training/Operations Proposed Resolution:

The question is correct as written.

The premise that the feedback addresses is based on the precaution and limitation from the system normal operating procedure, N-RHR-34, Residual Heat Removal System Operation. This P&L is applicable to normal operation conditions only. The conditions given in the question is post-accident. Thus the limitations for the systems are the "design" limits. Consideration must be then given to RHR, SI and Containment Spray systems that may be supplied on recirculation flow. From the USAR the following design values are:

	Design Temperature	Design Pressure
RHR Pump (also for valves & pipes)	400°F	600 psig
RHR Hx	400° (tube) / 350°F (shell)	600 psig (tube) / 150 psig (Shell)
SI Pump	300°F	2485 psig
ICS Pump	300°F	500 psig

Based on the accident analysis the maximum expected Containment Sump temperature is approximately 280°F. This correlates to a maximum Containment pressure of 46.0 psig. These valves fall within the acceptable design limits for the associated systems that may receive the recirculated fluid under accident conditions. As shown by the graph, the actual temperature in the Containment Sump at the time ES-1.3 is being performed (approximately 20 minutes following the initiation of the accident) will be closer to 250°F (maximum). This is confirmed in USAR Table 6.2-12 that gives 250°F as the maximum operating conditions for RHR components during post LOCA recirculation.

NRC Resolution:

The NRC reviewed the applicant's contention and the response provided by the facility's Training Department. The NRC agreed with the facility's position concerning this question. The stem clearly showed that an accident had occurred and design temperatures and pressures were values to be used when considering an answer for this question. Technical Specification Bases 3.3-4, Amendment No. 178 stated that: "One component cooling water pump together with one component cooling heat exchanger can accommodate the heat removal load either following a loss-of-coolant accident or during normal plant shutdown. If during the post-accident phase, the component cooling water supply were lost, core and containment cooling could be maintained until repair were effected." This eliminated distractors "c." and "d." The correct answer then becomes a choice between distractors a. and b. Distractor "b." requires a transition to Core Cooling FRGs to provide Core Cooling. This is not necessary. As indicated in the Training/Operations Proposed Resolution, adequate cooling is provided without entering Core Cooling FRGs. The background document for ES-1.3, Step: "Establish Component Cooling Flow to the RHR Heat Exchangers," stated: "If CC cannot be established to one heat exchanger, the remaining procedure can be performed as listed provided that the un-cooled recirculation fluid temperatures and pressure do not exceed equipment design conditions." The station's USAR stated (p 9.3-13): "During this period, no heat removal from the containment by the Residual Heat Removal System is required since the containment fan coil units capability using service water exceeds decay heat generation." Putting all of these items together means that containment sump recirculation can be established without CCW, and core decay heat is transferred to the containment and removed by containment fan coil units (distractor “a.”) without a transition to core cooling FRGs required by distractor “b.” The NRC believed there was enough information in the stem of the question to correctly answer the question with the above background information applied. No change was made to the answer key; distractor “d.” was retained as the only correct answer.

Question Number 85:

Given the following:

- The plant is operating at 100% power.
- ONLY the following annunciators are in alarm:
- 47104-A, BATTERY CHARGER A TROUBLE
- 47101-A, BRA-102 DC VOLTAGE LOW

What actions are required based on these conditions?

a. Go to E-EDC-38A, Loss of Train A DC Power.

b. Declare D/G A inoperable.

c. Verify BRA-102 voltage greater than 105VDC for battery OPERABILITY.

d. Cross-connect BRA-102 and BRB-102 per A-EDC-38.

ANSWER
c.

Examinee Feedback:

Answer "A" is correct based on procedure use and adherence rules. E-EDC-38A Symptoms Section 2.1 lists 47101-A, DC VOLTAGE LOW, as an entry condition for procedure. Procedure E-EDC-38A may be implemented due to a symptom being met, therefore answer "A" is correct.

Training/Operations Proposed Resolution:

The question is correct as written.

The single symptom, "BRA-102 DC VOLTAGE LOW (47101-A)," is not indicative of a loss of power to BRA-104, which is the purpose of E-EDC-38A. It would not be appropriate to perform E-EDC-38A for these conditions. The Alarm Response Procedure in and of itself directs, "GO TO A-EDC-38."

Other than the first Subsequent Action of E-EDC-38A, 4.1, "Contact Plant Electricians to investigate cause for loss of power," the remaining steps of E-EDC-38A address actions for systems lost or affected by the loss of DC. These do not address the conditions provided in the premise. A-EDC-38, which also has the same symptom, does directly address actions to address the specific conditions by investigating and checking local indications and components.

The UG-O provided section on Procedure Entry refers to dual-column format procedures only, as indicated by the hierarchy.

1. Performance of all dual-column format procedures shall start at Section 4.0. Detailed Procedure, Step 1.
    a. Section 1.0. Introduction, and Section 2.0. Symptoms, may be used as necessary to determine if the procedure is applicable.

E-EDC-38A and A-EDC-38 are single-format procedures. The condition may apply to these procedures, but the key statement is "to determine if the procedure is applicable."

NRC Resolution:

Upon review of this question and associated references, the NRC determined that distractor “a.” “Go to E-EDC-38A, Loss of Train A DC Power” is not a correct answer. This procedure is entered upon loss of power to BRA-104 which is not indicated by the two annunciators provided in the question stem. The NRC agrees with the facility’s contention that the question is correct as written. No change was made to the answer key. Distractor “c.” was retained as the only correct answer to question #85.

Question Number 86:

Given the following:

- The plant was operating at 100% power.
- A total Loss of Feedwater occurred.
- The reactor was tripped.
- All Auxiliary Feedwater Pumps failed to start.
- FR-H.1, Response to Loss of Secondary Heat Sink was implemented.
- RCS Bleed and Feed was initiated.
- Pressurizer PORV PR-2B failed to open.
- RCS Wide Range Hot Leg temperatures and Core Exit Thermocouple temperatures are INCREASING.
- SG A Wide Range level indicates 2%.
- SG B Wide Range level indicates 3%.
- Repairs have made Feedwater Pump A available.
- SI is reset.

What mitigation strategy is required?

a. Reset FW Isolation,
Open FW Isolation Valves,
Fast start Main FW Pump A,
Open FW-10A and B, SG A and B Main Feedwater Bypass Flow Control Valve to establish 60 to 100 gpm feed flow.

b. Open FW isolation Valves,
Fast start Main FW Pump A,
Open FW-10A or FW-10B, SG A or B Main Feedwater Bypass Flow Control Valve to establish maximum feed flow to both SGs.

c. Open FW Isolation Valves,
Fast start Main FW Pump A,
Open FW-10A and B, SG A and B Main Feedwater Bypass Flow Control Valves to establish 60 to 100 gpm feed flow.

d. Reset FW Isolation,
Open FW Isolation Valves,
Fast start Main FW Pump A,
Open FW-10A or B, SG A or B Main Feedwater Bypass Flow Control Valve to establish maximum feed flow.

ANSWER
d.

Examinee Feedback:

Following establishment of Bleed and Feed, the PORV relieving to the PRT causes the rupture disc to rupture. It is anticipated that Containment may reach Adverse Conditions of 4 psig. Step 27 of FR-H.1 has criteria for a hot, dry SG at 5% [20% for Adverse]. In either case this is met and FW flow should be limited to 60 to 100 gpm until level is greater than 5% [20% Adverse]. Levels per the question are 2% and 3%. Answer "A" is correct.

Training/Operations Proposed Resolution:

The question is correct as written.

Hot Dry SG conditions do exist. However, with the RCS temperatures rising, the urgency of establishing a heat sink predominates, and full feedwater flow should be established to one SG. As indicated in the BKG FR-H.1, "...it is advisable to reestablish feedwater to only one steam generator regardless of the size of the plant or number of loops. Thus, if a failure occurs due to excessive thermal stresses, the failure is isolated to one steam generator."

This last statement makes "A" incorrect since it establishes flow to both SGs.

NRC Resolution:

The NRC reviewed this question and determined that a FW Isolation Reset was required. This eliminated distractors “b.” and “c.” The difference between distractors “a.” and “d.” is the feedwater flow rate that should be established to the steam generators and the number of generators that should receive feed water. Step 26 of FR-H.1 (Response to Loss of Secondary Heat Sink) asks, “Check RCS Temperature: Core Exit TCs DECREASING.” When that question is answered “NO,” because the question conditions indicated that core exit thermocouple temperatures were increasing, the contingency action must be executed. The contingency action stated: “Decrease RCS temperatures by establishing maximum feed flow to one SG AND GO TO Step 25.” This established distractor “d.” as the correct answer as feed is directed to one SG at maximum rate. The answer key was not changed; distractor “d.” was retained as the only correct answer.

Question Number 91:

Given the following:

- The plant is operating at 100% power.
- SP-36-082, Reactor Coolant Leak Rate Check, has just been completed.
- Based on VCT level change, the total RCS leak rate is indicated at 1.11 gpm.
- Annunciator 47015-L, RXCP B STANDPIPE HIGH/LOW has alarmed.
- The NCO has determined that Standpipe level is HIGH.
- Reactor Coolant Pump B #1 Seal Leakoff is 2.8 gpm.
- Auxiliary Operator reports that RCDT level increased from 20% to 34% in 1 hour.
- Auxiliary Operator reports that RCDT level normally increases 3% over 4 hours.

What is the required course of action for the above conditions? (Operator Aid 89-6 provided.)

a. NO additional action is required since the unidentified RCS leakage is less than 1 gpm.

b. Leakage to the RCDT must be evaluated as safe for continued operation, and the investigation into source of the unidentified leakage must continue.

c. The maximum allowed leakage rate is exceeded, the reactor is to be placed in HOT SHUTDOWN condition within 12 hours.

d. The reactor must be shutdown, and the plant taken to COLD SHUTDOWN within 24 hours due to a non-isolable fault.

ANSWER
b.

Examinee Feedback:

Answer "A" is correct. The premise of the question states that SP-36-082 is completed with a leak rate of 1.11gpm of total RCS leakage. Also given are the combination of the RXCP annunciator and RCDT level change both of which are indicative of a Number 2 Seal issue on the B RXCP. Using the provided Operator Aid the leakage from the Number 2 seal can be calculated to be approximately 0.75gpm. (change of the last hour minus normal leakage based on RCDT level change or 0.8 gpm - 0.04gpm = -0.75 gpm) This leakage is now considered identified leakage. It is known where it is coming from, that it is contained, and that it is within the limitations of the RCDT. Being that the total leakage identified by SP-36-082 was 1.11 gpm and 0.75gpm has been determined to be identified leaves the remaining 0.36gpm as unidentified. Technical Specifications allows not in excess of 1gpm unidentified leakage and not in excess of 10 gpm identified leakage. Neither of these thresholds have been met. Therefore no actions are required since unidentified leakage remains less than 1gpm per Technical Specifications.

Training/Operations Proposed Resolution:

Accept either "A" or "B" as correct.
The premise does state, "SP-36-082, Reactor Coolant Leak Rate Check, has just been completed." The intent of this was to signify that the Mass balance Leakrate Calculation had been completed. In this event, the investigation is required. This was apparent during development, review and validation, since no comments were received that disputed the selected correct action.

However, the statement in the premise can be interpreted to mean the entire procedure is complete, which includes Section 6.3, the Investigation and Evaluation. If this is performed and the RCDT level change is attributed to the RXCP #2 seal problem, then no other action is required. Data Sheet 4, INVESTIGATION AND EVALUATION, would have been completed to describe the source of the leak, the effect on plant operation and the determination the plant operation may safely continue. This also agrees with the actions of A-RC-36C, Attachment b, Response to Abnormal #2 Seal Leakoff, with #2 seal leakoff flow greater then 0.5 gpm but less than 1.1 gpm.

NRC Resolution:

The NRC agreed with the applicant and the facility that there are two correct answers for this question for the reasons stated above. The answer key was modified to accept distractors “a.” and “b.” as correct answers to question #91.

Question Number 95:

Given the following:

- The plant is in REFUELING SHUTDOWN.
- Irradiated fuel movement is in progress.
- Containment Purge is in service
- Annunciator 47013-A, RAD MONITOR SAMPLING FLOW HIGH/LOW is alarming.
- R11/12, Containment Particulate/Gas Monitor, Sample Pump has stopped and will NOT restart.

What action is required?

a. Verify Containment Ventilation Isolation Train A has occurred.

b. Restore R11/12 Sample flow to normal within 4 hours or stop the Containment Purge.

c. Verify R21 is operating and aligned to sample Containment.

d. Immediately suspend fuel movement.

ANSWER
c.

<u>Examinee Feedback</u>:

Accept "D" as a correct answer.

N-FH-53CLD, Refueling Daily Checklist, has you check R-12 operating. If the Checklist or an item on the Checklist is not verified, you cannot/should not move fuel. The action is conservative.

<u>Training/Operations Proposed Resolution</u>:

Accept either "C" or "D" as correct.

The ODCM requires either R-12 or R-21 to be OPERABLE and addresses action for a purge in progress, but does not specifically address Fuel Handling. Likewise the normal procedure NRM-45, Radiation Monitoring System, addresses the removal of R-12 from service, and ensuring R-21 is operating and aligned to sample the appropriate location.

Using Conduct of Operations and the Standard DNOS - 0101 for Nuclear Safety and Conservative Decision Making, it is not unreasonable to stop fuel movement while verifying the status of required components. The operations guidance for fuel handling, N-FH-53-CLC, Pre-Refueling Checklist, and N-FH-53-CLD, Refueling Daily Checklist, both require R-12 and R-21 to be operating during fuel movement. There is no specific guidance on actions to take if one of the monitors fails. Therefore, Conservative Decision Making is applied, and fuel movement should be stopped. This is also supported by the General Notes of RF-03.01, Fuel Movement During a Refueling Outage," 2.7 which states, in part, "Each member of the refueling team needs to understand that they have the authority to stop refueling activities to resolve an issue."

NRC Resolution:

The NRC disagrees with the applicant's and the facility's proposed resolution to question #95. The stem of the question asks, "What action is required?" (Emphasis added.) While suspending fuel movement is a conservative action, it is not required at this point. This makes distractor "d." incorrect. Distractor "c.," however, is also incorrect. Kewaunee Power Station Operating Procedure N-RM-45, Step 4.3.10.a stated: "IF Containment Purge/Vent is in progress, THEN VERIFY R-21 operating and aligned to sample stack, OR STOP Purge/Vent." Distractor "c.," which is the proposed correct distractor has the operator verify R21 is operating and aligned to sample Containment, not the sample stack as required by the procedure. Since there was no correct answer provided for this question, the question was deleted from the examination. The answer key was modified to show question #95 was deleted.

Question Number 98:

Given the following:

- A Loss of All AC Power occurred.
- ECA-0.0, Loss of All AC Power has been implemented.
- At Step 18, while performing actions to prepare to tie Bus 52 to Bus 46, Maintenance reports DG B is ready for starting.
- Diesel Generator B has been manually started.
- Bus 6 has been energized.

The NCO reports Core Exit Thermocouples are reading 725°F and increasing. The NCO requests that Safety Injection Pump B be started to restore Core Cooling.

What is the proper action?

a. Wait until it is directed in FR-C.2, Response to Degraded Core Cooling.

b. Direct the start of Safety Injection Pump B.

c. Wait until it is directed by ECA-0.2, Loss of All AC Power Recovery with SI Required.

d. Direct manual initiation of Safety Injection, and verify Safety Injection Pump B starts.

*ANSWER
c.

<u>Examinee Feedback</u>:

Answer "B" is correct answer. When DG is started it is done per A-DGM-10B, and Bus 6 is energized per Step 4.6. Step 4.6.4 states "Sequentially start safeguards equipment as required." This guidance can be used to start the SI Pump B. No conflicting guidance is given in ECA-0.0

<u>Training/Operations Proposed Resolution</u>:

The question is correct as written.

Step 7 of ECA-0.0 directs the operator to place specific equipment in PULLOUT, including the SI Pumps. A NOTE prior to Step 9 (Dispatch personnel to locally restore Emergency AC Power) reads. "Pre-planning of power restoration efforts based on the event and available sources is required." When power is restored to the Bus the operator is directed to continue actions of ECA-0.0 at Step 37. Three CAUTIONS exist prior to Step 38. The two applicable CAUTIONS read,” The loads placed on the energized emergency AC Bus should not exceed the capacity of the power source," and, "If an SI signal exists or if an SI signal is actuated during this procedure, it should be reset to permit manual loading of equipment on an emergency AC bus." At Step 40, the operator transitions to the appropriate recovery procedure (ECA-0.2 in this case). The NOTE at the beginning of the procedure states (Likewise in ECA-0.0), "CSF Status Trees should be monitored for information only. Function Restoration Procedures should not be implemented prior to completion of Step 10. At Step 5 in ECA-0.0, the operator is directed to manually load safeguards equipment on AC Emergency Bus, including SI Pumps.

BKG ECA-0.0 specifically states in 1. Introduction,

> If plant conditions have deteriorated significantly, the operator may have insufficient or conflicting indications as to plant status and a concurrent event may be contributing to the deterioration of RCS conditions. Under these RCS conditions, the operator is instructed to implement IPEOP ECA-0.2 and initiate plant recovery utilizing Safety Injection (SI) operational systems. IPEOP ECA-0.2 functions to start safe-guards equipment as appropriate and then directs the operator to go to IPEOP E-1...)

Also (Section 3)

> The loss of all ac power procedures are unique within the IPEOP set. With the exception of these procedures, all IPEOPs are written on the premise that at least one ac emergency bus is energized and associated equipment can be powered from the energized ac emergency bus. Consequently, the guidance provided in other procedures in the IPEOP set is not applicable following a loss of all ac power. Therefore, ECA-0.0 has priority over all other procedures in the IPEOP set.
>
> (3.1.4)
> Following restoration of ac power, the operator is instructed to stabilize steam generator pressures, if secondary depressurization is in progress, and to evaluate the status of the energized ac bus. These actions verify that certain select equipment has automatically loaded on the ac emergency bus and provides the operator with information that will aid him in loading subsequent equipment on the energized ac emergency bus in recovery procedures.

Step 6 CAUTION 1 (for the transition to Step 37) basis reads:

> To minimize the deterioration of plant conditions, recovery actions should be started as soon as ac power is restored. Procedure ECA-0.0 is written such that recovery actions step can be entered from any step that follows this CAUTION. Procedures ECA-0.0, ECA-0.1 and ECA-0.2 are written to establish the appropriate systems operation and alignments before transitioning the operator to other IPEOPs.

UG-0, User's Guide For Emergency and Abnormal Procedures, sets the priority for implementing procedures. Section 6.2 identifies the general order of priority: 1) FRPs; 2) ORPs; 3) EOPs; and 4) AOPs. As mentioned above ECA-0 series is special in that it takes priority over FRPs. In this case the direction of ECA-0 series should take the highest priority in actions to be performed.

The action in "B" will start the SI Pump without having a Component Cooling Water Pump running for support.

NRC Resolution:

The NRC reviewed the Westinghouse Emergency Guidelines and agreed with the facility's argument that there is a restoration sequence for the event described in the question stem, and that distractor "b." does not adhere to that restoration sequence. Guideline ECA-0.0, Loss of All AC Power, provided procedural guidance for a loss of all ac power as an initiating event or as a coincident occurrence in combination with a loss of reactor coolant, loss of secondary coolant or steam generator tube rupture. Following the loss of ac power, optimal recovery cannot be initiated until ac power is restored to at least one emergency ac bus. Consequently the major objective of guideline ECA-0.0, following the immediate actions and the actions to check RCS isolation and secondary heat sink availability, is to restore ac emergency power as soon as possible. Guideline ECA-0.0 instructs the operator to restore ac emergency power from the control room. If ac power can be restored, RCS conditions will not have deteriorated significantly and the operator is directed to return to the guideline and step in effect when ac power was lost. If ac power cannot be restored from the control room, the operator is instructed to dispatch personnel to locally restore ac power. The operator then proceeds to maintain plant conditions until ac power is restored. The last action category in ECA-0.0 is the selection of the recovery guideline based on existing RCS conditions. The criteria for recovery guideline selection include: (1) existence of RCS subcooling, (2) existence of pressurizer level, and (3) verification that SI equipment has not automatically actuated upon restoration of ac power. If RCS conditions have not deteriorated significantly (i.e., all criteria are satisfied), the operator is directed to guideline ECA-0.1 to recover the plant using normal operational systems. If RCS conditions have deteriorated significantly, (i.e., any criterion not satisfied), the operator is directed to guideline ECA-0.2 to recover the plant using safeguards systems.

The question stem stated that ECA-0.0 had been implemented. The stem stated that ac power restoration took place outside the control room, indicating a more serious condition existed. Step 40 of ECA-0.0 directed operators to select an appropriate recovery procedure. Since the core exit thermocouples are reading 725°F and rising (given in question stem), the operator should select ECA-0.2 as the correct recovery procedure (distractor "c."). The NRC did not agree with the applicant's assertion that distractor "b." was a correct answer, especially in light of the fact that the SI pump would be started without having a component cooling water pump started to support SI pump operations. The answer key was not amended; distractor "c." was retained as the only correct answer for this question.