October 26, 2007

Mr. Christopher M. Crane
President and Chief Nuclear Officer
Exelon Nuclear
Exelon Generation Company, LLC
4300 Winfield Road
Warrenville, IL 60555

SUBJECT: CLINTON POWER STATION
NRC INITIAL LICENSE EXAMINATION REPORT 05000461/2007301(DRS)

Dear Mr. Crane:

On August 23, 2007, Nuclear Regulatory Commission (NRC) examiners completed initial operator licensing examinations at your Clinton Power Station. The enclosed report documents the results of the examination. An initial de-brief of the examination was presented on August 23, 2007, with Mr. B. C. Hanson and other members of your staff. An exit meeting was conducted by telephone on September 18, 2007, between Mr. M. Helton, Initial License Training Lead of your staff and Mr. D. McNeil, Senior Operations Engineer, to review the NRC's resolution of the station's post examination comments and the proposed final grading of the written examination for the license applicants.

The NRC examiners administered an initial license examination operating test during the weeks of August 13 and August 20, 2007. The written examination was administered by NRC examiners and Clinton Power Station training department personnel on August 23, 2007. Seven Senior Reactor Operator (SRO) and five Reactor Operator (RO) applicants were administered license examinations. Two of the Senior Reactor Operator applicants were previously licensed Reactor Operators at the Clinton Power Station. The results of the examination were finalized on October 12, 2007. Four applicants (1 SRO and 3 RO) failed the written examination and were issued proposed license denial letters. Five applicants (4 SRO and 1 RO) passed all sections of their respective examinations and were issued applicable operator licenses. In accordance with NRC policy, the remaining three applicants (2 SRO and 1 RO) passed all sections of their respective examinations, but because their final written examination grade was less than 82 percent, their licenses are being withheld pending the outcome of any written examination appeal that may be initiated by the applicants that failed the written examination.

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 examination.

Sincerely,

/RA/

Hironori Peterson, Chief
Operations Branch
Division of Reactor Safety

Docket No. 50-461
License No. NPF-62

Enclosures:
1. Operator Licensing Examination Report 05000461/2007301(DRS) w/Attachment: Supplemental Information
2. Simulation Facility Report
3. Post Examination Comments w/ NRC Resolution
4. Written Examinations and Answer Keys (RO/SRO)

cc w/encls 1 & 2:
Site Vice President - Clinton Power Station
Plant Manager - Clinton Power Station
Regulatory Assurance Manager - Clinton Power Station
Chief Operating Officer
Senior Vice President - Nuclear Services
Vice President - Operations Support
Vice President - Licensing and Regulatory Affairs
Manager Licensing - Clinton Power Station
Senior Counsel, Nuclear, Mid-West Regional Operating Group
Document Control Desk - Licensing
Assistant Attorney General
Illinois Emergency Management Agency
State Liaison Officer, State of Illinois
Chairman, Illinois Commerce Commission

cc w/encls 1, 2, 3, & 4: J. Lindsey, Training Director, Clinton Power Station

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Vice President - Licensing and Regulatory Affairs
Manager Licensing - Clinton Power Station
Senior Counsel, Nuclear, Mid-West Regional Operating Group
Document Control Desk - Licensing
Assistant Attorney General
Illinois Emergency Management Agency
State Liaison Officer, State of Illinois
Chairman, Illinois Commerce Commission

cc w/encls 1, 2, 3, & 4: J. Lindsey, Training Director, Clinton Power Station

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Letter to Christopher M. Crane from Hironori Peterson dated October 26, 2007.

SUBJECT: CLINTON POWER STATION
NRC INITIAL LICENSE EXAMINATION REPORT 05000461/2007301(DRS)

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REGION III

Docket No:	50-461
License No:	NPF-62
Report No:	05000461/2007301(DRS)
Licensee:	AmerGen Energy Company, LLC
Facility:	Clinton Power Station
Location:	Clinton, Illinois
Dates:	August 13, 2007, through August 23, 2007
Examiners:	D. McNeil, Senior Operations Engineer C. Zoia, Operations Engineer C. Moore, Operations Engineer
Approved by:	Hironori Peterson, Chief Operations Branch Division of Reactor Safety

SUMMARY OF FINDINGS

ER 05000461/2007301(DRS); 08/13/2007 - 08/23/2007; AmerGen Energy Company, LLC, Clinton Power Station. Initial License Examination Report.

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

Examination Summary:

- Four applicants (1 Senior Reactor Operator and 3 Reactor Operators) failed the written examination and were issued proposed license denials. Eight applicants passed all sections of their respective examinations. Five applicants (4 Senior Reactor Operators and 1 Reactor Operator) were issued applicable operator licenses. The remaining three licenses (2 Senior Reactor Operators and 1 Reactor Operator) may be issued pending the outcome of any written examination appeal. (Section 4OA5.1).

REPORT DETAILS

4. OTHER ACTIVITIES (OA)

4OA5 Other

.1 Initial Licensing Examinations

a. Examination Scope

Members of the Clinton Power Station Training Department prepared the examination outline and developed the written examination and operating test. The NRC examiners validated the proposed examination during the week of July 23, 2007, at the Clinton Power Station Training Building with the assistance of members of the licensee training staff. During the on-site validation week of July 23, 2007, the examiners audited two license applications for accuracy. The NRC examiners conducted the operating portion of the initial license examination during the weeks of August 13, 2007, and August 20, 2007. The NRC examiners and members of the Clinton Power Station training department staff administered the written examination on August 23, 2007. The station's examination developers 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

During the validation of the written examination several questions were modified or replaced as requested by the NRC examiners. Changes made to the written examination were documented on Form ES-401-9, "Written Examination Review Worksheet," which is available electronically in the NRC Public Document Room or from the Publicly Available Records component of NRC's document system (ADAMS). Less than 20 percent of the examination was considered unsatisfactory for examination use and therefore, met the NRC's expectations for a submitted examination. The licensee submitted sixteen written examination post-examination comments for consideration by the NRC examiners when grading the written examination. Twelve of the submitted post-examination comments were comments from the applicants and were not supported by the facility's training staff. Based on the significant number of written examination failures, additional reviews were conducted with the licensee with respect to post-examination comments. The post-examination comments and the NRC resolution for the post-examination comments are contained in Enclosure 3, "Post Examination Comments and Resolutions." The NRC examiners completed the final grading of the written examination on October 10, 2007, and conducted a review of each missed question to determine the accuracy and validity of the examination questions.

Operating Test

During the validation of the operating test, minor changes to the operating test were made to the JPMs and to the dynamic simulator scenarios. One JPM was replaced because it lacked discriminatory value. Less than 20 percent of the operating test was considered unsatisfactory for use and therefore, met the NRC's expectations for a submitted examination. Changes made to the operating test were documented in a document titled, "Operating Test Comments," which is available electronically in the NRC Public Document Room or from the Publicly Available Records component of NRC's document system (ADAMS). The NRC examiners completed operating test grading on October 9, 2007.

Examination Results

Seven applicants at the Senior Reactor Operator (SRO) level and five applicants at the Reactor Operator (RO) level were administered written and operating tests. Two of the SRO applicants were previously licensed as ROs at Clinton Power Station. Four applicants (1 SRO and 3 ROs) failed the written examination and were issued proposed license denials. Five applicants (4 SRO and 1 RO) passed all portions of their examinations and were issued applicable operating licenses. Three applicants (2 SRO and 1 RO) passed all portions of the license examination, but received a written test grade below 82 percent. In accordance with NRC policy, those applicants' licenses will be withheld until any written examination appeal possibilities by other applicants have been resolved. If an applicant's grade is still equal to or greater than 80 percent after any appeal resolution, the applicant will be issued an operating license. If the applicant's grade has declined below 80 percent, the applicant will be issued a proposed license denial letter and offered the opportunity to appeal any questions the applicant feels were graded incorrectly.

.2 Examination Security

a. Scope

The NRC examiners reviewed and observed the licensee's implementation of examination security requirements during the examination validation and administration to assure compliance with 10 CFR 55.49, "Integrity of Examinations and Tests." The examiners used the guidelines provided in NUREG 1021, "Operator Licensing Examination Standards for Power Reactors," to determine acceptability of the licensee's examination security activities.

b. Findings

No findings of significance were observed in the area of examination security.

4OA6 Meetings

Debrief

The chief examiner presented the examination team's preliminary observations and findings on August 23, 2007, to B. Hanson and other members of the Clinton Power Station Operations and Training Department staff. The examiners asked the licensee whether any of the material used to develop or administer the examination should be considered proprietary. No proprietary or sensitive information was identified during the examination or debrief meetings.

Exit Meeting

The chief examiner conducted an exit meeting on September 18, 2007, with Mr. M. Helton, Clinton Power Station Initial License Training Lead, by telephone. The NRC's proposed disposition of the station's post-examination comments were disclosed and revised preliminary written examination results were provided to Mr. Helton during the telephone discussion.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee
B. Hanson, Site Vice President
F. Kearney, Plant Manager
J. Lindsey, Training Director
D. Shavey, Operations Director
T. Chalmers, Shift Operations Supervisor
A. Bailey, Operations Training Manager
M. Helton, Initial License Training Lead

NRC
D. McNeil, Senior Operations Engineer
B. Dickson, Senior Resident Inspector
A. Koonce, Acting Resident Inspector

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
ALARA	As-Low-As-Reasonably-Achievable
DRS	Division of Reactor Safety
IR	Inspection Report
NRC	Nuclear Regulatory Commission
RO	Reactor Operator
SRO	Senior Reactor Operator

SIMULATION FACILITY REPORT

Facility Licensee: Clinton Power Station

Facility Docket No: 50-461

Operating Tests Administered: August 13 - 23, 2007

The following documents observations made by the NRC examination team during the initial operator license examination. These observations do not constitute audit or inspection findings and are not, without further verification and review, indicative of non-compliance with 10 CFR 55.45(b). These observations do not affect NRC certification or approval of the simulation facility other than to provide information which may be used in future evaluations. No licensee action is required in response to these observations.

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

ITEM	DESCRIPTION
Annunciators	The NOT FULLY CLOSED CDSR EMERG O/FLOW VLV 1CD020 annunciator repeatedly alarmed during the scenarios. The issue has been investigated before, and the simulator problem report closed. The annunciator was treated as a transient (nuisance) alarm during the administration of the operating test.

POST EXAMINATION COMMENTS AND RESOLUTIONS

RO Question 2:

The unit is operating at rated power. Component Cooling Water (CC) pump 1A is out of service for repair. Annunciator 5040-1B, AUTO TRIP PUMP/MOTOR, actuates when CC pump 1C trips. A few seconds later annunciator 5003-3D, RECIRC MTR A WDG CLG WTR FLOW LO actuates. Three minutes later annunciator 5003-1K, RECIRC PMP MTR A OR B TEMP HI actuates.

How can the control room operator determine which Reactor Recirculation (RR) Pump component temperature is causing the alarm?

a. Direct the Equipment Operator to check RR temperatures at local panel 1B33-P001A(B).

b. Direct the Equipment Operator to check RR temperatures at local HCU panel.

c. Check Recirc pump temperatures on Control Room panel 1H13-P614 recorder.

d. Check Recirc pump temperatures on DCS.

ANSWER: (c.)

<u>Applicant Comment:</u>

The applicant stated that: “The question as stated asks for ways the operator can check RR pump temperatures. The ROs routinely use Display Screen (DCS) #10 for RR PMS points on Group Point #7.”

<u>Facility Proposed Resolution</u>:

The facility stated that: “Answer (c.) is the only correct answer. Even though RR pump temperatures are available on DCS Screen #10, the RO must change this screen to select Group Point #7 on PMS.”

<u>NRC Resolution</u>:

When the exam author originally proposed this question, he stated that, “RR pump temperatures can be monitored on PMS but not on DCS. The RR DCS displays contain a lot of information but do not have indication of pump temperatures.” For this reason the NRC chief examiner accepted distractor (d.) as an incorrect answer. In the facility’s proposed resolution to this applicant comment, the facility maintained that there is only one correct answer because the operator does not monitor RR pump temperatures from DCS, he must change screen #10 to select Group Point #7 on PMS. Associated with CRT #10 there are two pushbuttons. One pushbutton selects the Display Control System (DCS); the other pushbutton selects the Plant Monitoring System (PMS). In order for an operator to monitor reactor recirculation pump component temperatures he or she must select PMS on CRT #10. The component temperatures needed to do the analysis required in the question are not provided by PMS to DCS. Therefore, the operator is required to switch from the DCS to PMS to retrieve the necessary component temperatures. Depressing the PMS pushbutton associated with CRT #10 disconnects the screen from DCS and connects it to the PMS computers. At this point the operator is no longer viewing information from the DCS, but from PMS. To retrieve the correct data, the operator has to select Group Point #7 in PMS. The NRC agreed with the facility's proposed resolution that distractor (d.) was not a correct answer. Since the temperatures necessary to make the required diagnosis are not retrieved from DCS, but are available from PMS, the answer key was not changed and distractor (c.) was retained as the only correct answer.

RO Question 12:

Which ONE of the following describes the LOWEST power level that will result in a ROD OUT BLOCK initiation from the Source Range Monitoring (SRM) and Intermediate Range (IRM) Systems?

(NOTE: Choices are listed from LOWEST to HIGHEST power level.)

a. SRM A: 5 cps, SRM B: 3 cps, SRM C: 5 cps, SRM D: 4 cps,
   ALL SRMs are FULLY INSERTED.
   ALL IRMS on Range 1.

b. SRM A: 120 cps, SRM B: 100 cps, SRM C: 95 cps, SRM D: 140 cps,
   ALL SRMs are PARTIALLY WITHDRAWN.
   ALL IRMS on Range 1.

c. SRM A: 600 cps, SRM B: 750 cps, SRM C: 700 cps, SRM D: 650 cps,
   ALL SRMs are PARTIALLY WITHDRAWN.
   ALL IRMS on Range 1.

d. SRM A: $1.0x10^5$ cps, SRM B: $1.2x10^5$ cps, SRM C: $1.1x10^5$ cps, SRM D: $1.2x10^5$ cps,
   ALL SRMs are PARTIALLY WITHDRAWN.
   ALL IRMS on Ranges 2 or 3.

ANSWER: (a.)

Applicant Comment:

The applicant argued that: “Distractor (a.) is based on the assumption that a Rod Block occurs at 3.0 cps however conflicting information in CPS No. 3306.01, SOURCE/INTERMEDIATE RANGE MONITORS (SRM/IRM), section 2.7 & 6.2, as well as in CPS No. 3304.02, ROD CONTROL AND INFORMATION SYSTEM (RC&IS), TABLE 1: ROD BLOCK TROUBLESHOOTING GUIDE, Page 27. Actual Rod Block on low SRM count occurs when SRM count is ≤ 3.0 cps.

Facility Proposed Resolution:

The facility agreed that (a.) and (b.) were both correct answers for the reasons provided by the applicant.

NRC Resolution:

In the facility’s proposed examination, the facility stated that distractor (b.) would cause a rod block, but that the distractor was incorrect because it was not the lowest power level listed that would cause a rod block. The examination author stated that distractor (a.) would cause a rod block and was the lowest power level provided in the four distractors. For this reason the NRC examiners accepted distractor (a.) as the correct answer and distractor (b.) as an incorrect answer. The question is a recognition question seeking to determine if an applicant can recognize when a rod block will be initiated due to low SRM counts per second (cps). The NRC reviewed the applicant's contention, the facility's proposed resolution, and the documentation accompanying the facility's proposed resolution. The NRC examiners noted that there were references provided to support two correct answers. The following was reviewed:

Procedure/Reference	Statement	Justification
CPS No. 2204.02, RC&IS	Any SRM downscale (<3 cps)	Distractor b. is correct
CPS No. 3306.01, SRM/IRM, Step 2.7.2	SRM downscale at 3 cps	Distractor a. is correct
CPS No. 5005-3K, SRM DNSC (Annunciator)	SETPOINT <3 cps	Distractor b. is correct
Technical Specifications SR 3.3.1.2.4	Verify count rate is ≥3 cps	Distractor a. is correct
CPS ORM, Table 3.2.2-1.d	Downscale setpoint ≥ 3 cps	Distractor a. is correct

The examiners noted that there appears to be multiple correct answers. However, the station's Technical Specifications is part of the facility license and is the governing document. The Technical Specifications state that a rod block will occur at a count rate between 1.8 and 4.27 cps. This would mean that a rod block could have occurred with the conditions established in distractor (a.), making distractor (a.) a correct answer. Finally, the NRC asked for and reviewed, CPS No. 9431.13, Source Range Monitor C51-K600A(B,C,D) Channel Calibration, Revision 44, Section 8.4, Source Range Monitor Post Calibration Check. At Step 8.4.25 the SRM post-calibration values are recorded. The table stated that 3.0 cps is the desired value, 1.8 cps is the Technical Specification minimum, and 4.27 cps is the allowed maximum cps for the trip setpoint. It was found that the actual setpoint for the plant was 3.0 cps. Since the 3.0 cps given in distractor (a.) is within the technical specification requirements, and is the actual setpoint for the SRMs, it is reasonable to expect that a rod block would have occurred under the conditions given in distractor (a.). Since distractor (a.) is the lowest power level of the four distractors that would initiate a rod block, it is the only correct answer. The answer key was not changed and distractor (a.) was retained as the only correct answer.

RO Question 18:

The plant is operating at rated power with the following:

- Weekly control rod exercising is in progress.
- The next control rod to be exercised is rod 28-21 which is currently at position 48.

Which ONE of the following identifies the REQUIREMENT regarding where the Control Room Supervisor (CRS) MUST be positioned in order to adequately supervise the performance of this rod movement?

a. Anywhere within the “At the Controls” area.

b. Immediately behind the Reactor Operator.

c. Anywhere within the Main Control Room.

d. In proximity to the Reactor Operator.

ANSWER: (a.)

Applicant Comment:

The applicant argued that answer (d.) was also correct based on the Main Control Room (MCR) layout and procedure OP-AB-300-1001, Section 4.5 criteria which specifically stated: “… in proximity to the Reactor Operator…” This was a SHOULD statement, which must be treated as a SHALL statement unless waived by shift management. Answer (a.) was based on a MAY statement which was encompassed by distractor (d.).

Facility Proposed Resolution:

Only answer (a.) was correct. For rods fully inserted or fully withdrawn, the reactivity change is minimal. Therefore, the Control Room Supervisor (CRS) may adequately supervise control rod exercises from anywhere in the “At the Controls” area.

NRC Resolution:

Upon review of the question, applicant comment, and facility proposed resolution, the NRC determined that the question had three correct answers. The question was designed to test the applicant’s understanding of the change in supervisory requirement when a fully inserted or fully withdrawn control rod was exercised. When other control rods (not fully inserted or withdrawn) are being moved, the CRS must be “in proximity to the Reactor Operator,” (OP-AB-300-1001, Step 4.5). However, the CRS is not required to be “in proximity to the Reactor Operator,” but only in the “at the controls area,” when fully withdrawn or fully inserted control rods are being exercised (also OP-AB-300-1001, Step 4.5). However, “In proximity to the Reactor Operator (d.),” and “Immediately behind the Reactor Operator (b.),” are subsets of “Anywhere within the “At the Controls” area (a.).” This makes distractors (a.), (b.), and (d.) correct answers. Additionally, the question stem asked for the REQUIREMENT for where the CRS must stand. The procedure used a SHOULD statement, indicating the location of the CRS during rod movements is not a requirement. Considering this information, there is no correct answer to the question because there is no required location that the CRS must stand. Because there were three correct answers, or in the case of REQUIREMENT, no correct answers, the answer key was modified to delete this question from the examination.

RO Question 27:

The unit is operating in normal, full power lineup when a loss of off-site power occurs and the indicating lights for all breakers on the 6.9KV and 4.16KV 1A panels extinguish.

Which ONE of the following components will likely be damaged?

a. Main Turbine

b. Recirculation Pump A Motor

c. Circulating Water Pump B Motor

d. Motor Driven Reactor Feedwater Pump

ANSWER: a.

Applicant Comment:

The applicant argued that: “Distractor b. is plausible. There is no direct indication or cause in the question stem that would lead the examinee to assume the Turbine Generator tripped. Therefore, based on the conditions provided, the UAT would continue to power the non-safety related busses. This plant response would then make the RR pump (distractor b.) the most likely component to be damaged by this transient of the choices provided.”

Facility Proposed Resolution:

“Answer a. is the only correct answer. This scenario was performed on the simulator with the end result being that the Turbine tripped. With the loss of DC power, the Emergency Bearing Oil Pump would not be available and damage would result to the Main Turbine.”

NRC Resolution:

Upon review of applicant’s comment and the facility’s proposed resolution, the NRC determined that the original answer (a.) was the only correct answer. With a loss of off-site power, there will be no place for the power being generated by the main generator to be sent, resulting in a turbine trip, most likely caused by a power-load unbalance trip. The facility used the station’s simulator to verify that a trip does occur. The UATs will still be powered as the turbine coasts down since the turbine output breakers to the UATs have lost control power and cannot open. The loss of breaker indicating lights given in the question stem provided an indication of the loss of DC power from MCC 1E. This MCC also powers the Main Turbine Emergency Bearing Oil Pump (EBOP). When the main turbine trips and begins coasting down, lube oil from the shaft driven lube oil pump will decrease in pressure, leading to reduced lube oil flow and possible damage to the main turbine bearings because the EBOP cannot start. This makes distractor (a.) a correct answer. Distractor (b.) indicated that the RR pump ‘A’ motor could be damaged. If there was a loss of CCW the damage to the pump could occur. However, a loss of DC MCC 1E will not affect CCW and no damage occurs to the RR pump ‘A’ motor. Since there is no damage to the RR pump ‘A’ motor, distractor (b.) was rejected as a correct answer. The answer key was not changed and distractor (a.) was retained as the only correct answer.

RO Question 36:

A reactor coolant pressure boundary leak has occurred inside the Drywell.

- All control rods are fully inserted
- Drywell pressure is 4.6 psig and rising slowly
- Reactor water level has just reached -145.5 inches and is dropping slowly
- Reactor pressure is 885 psig and rising slowly
- CRD is the only high pressure injection source available
- All low pressure ECCS pumps have started

What is the status of the Automatic Depressurization System (ADS) and what procedural actions are required to mitigate the consequences of these conditions?

a. ADS has automatically initiated. Verify the seven (7) ADS valves are open.

b. ADS will automatically initiate in 105 seconds. Inhibit ADS at the direction of the CRS.

c. ADS will automatically initiate in 105 seconds. Immediately inhibit ADS and notify the CRS.

d. ADS will automatically initiate in 6 minutes. Reset the RPV water level low ADS seal-in logic.

ANSWER: (b.)

Applicant Comment:

The applicant argued that: “Distractor c. is plausible based on EOP-1 direction. The CRS has directed (given permission) to the RO to “inhibit ADS if the timer initiates.” Based on the stem conditions, it is the CRS would be [SIC] at the step to “WAIT until TAF”’ and the ADS inhibit direction would have already been issued. This would then allow the RO to “immediately” inhibit ADS and notify the CRS. At this time the CRS would not be expected to once again give the RO direction to inhibit ADS. The CRC [SIC] would expect the RO to take actions immediately.”

Facility Proposed Resolution:

“Answer b. is the only correct answer. To take EOP actions requires direction of the CRS. Pre-direction from the CRS was not included in the question stem.”

NRC Resolution:

Upon review of the question, the applicant’s comment, and the facility’s proposed resolution, the NRC determined that answer (b.), was the only correct answer. The applicant had assumed in his answer to the question that the CRS would provide early permission to inhibit ADS if the timer initiates before reaching the top of active fuel (TAF). This assumption is not allowed by NUREG 1021, Operator Licensing Examination Standards for Power Reactors, Revision 9, Appendix E., Policies and Guidelines for Taking NRC Examinations. The applicants were briefed on the NRC's rules concerning taking NRC examinations. Item number 7 for written examinations states: "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 this question did not state that the SRO had pre-approved any action concerning ADS. The facility's proposed response indicated that there was only one correct answer to this question and provided OP-CL-101-111-1001, Revision 4, to provide expectations for its control room operators. On page 4, Step B.3) of the procedure it stated, "Get the CRS permission PRIOR to defeating an automatic initiation of an ECCS component." Based on this requirement to get permission prior to defeating an automatic initiation, and the lack of pre-approval by the CRS to take EOP actions, the NRC has determined that there was only one correct answer provided for this question. The answer key was not modified and distractor (b.) was retained as the only correct answer for this question.

RO Question 46:

The plant is operating at 25% power. The Main EHC System Emergency Trip System pressure drops to 350 psig.

Using the attached DCS displays.

What will be the final steady state Feedwater temperature entering the Reactor.

a. 87°F

b. 90°F

c. 286°F

d. 312°F

ANSWER: (b.)

<u>Applicant Comment:</u>

The applicant argued that: "Distractor a. is technically correct when the stem condition "final" is applied to the expected operator actions for the transient. Per CPS No. 4005.01, a loss of FW heating >100°F may occur requiring a SCRAM. CPS No. 3006.01 directs removing SJAEs; therefore "final" FW temp would be 87°F."

<u>Facility Proposed Resolution</u>:

"The only correct answer is b. When ETS pressure drops below the turbine trip setpoint the Main Turbine trips and steam flow is shut off to all FW heaters. This results in "final steady state" FW temperature being approximately the temperature at the discharge of the Condensate-Booster pumps. Based on the provided figures that temperature is 90°F."

<u>NRC Resolution</u>:

Upon review of the question, the applicant's comment, and the facility's proposed resolution, the NRC determined that the original answer, (b.), was the only correct answer. For the temperature to be reduced to 87°F, SJAEs would have to be shut down. This action was not specified in the stem of the question. The applicant made an assumption that an operator took action to remove the SJAEs from service. NUREG-1021, Appendix E, "Policies and Guidelines for Taking NRC Examinations," specifically prohibits applicants from making assumptions not specified in the stem of the question. Appendix E, Item #7 states, in part, 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 assume that no operator actions have been taken, unless the stem of the question or the answer choices specifically state otherwise. Because no direction was provided in the question stem that the SJAEs were secured, distractor (a.) is not a correct answer. The answer key was not amended; distractor (b.) was retained as the only correct answer.

RO Question 51:

Irradiated fuel is being transferred from the Containment fuel storage pool to the Fuel Building spent fuel storage pool.

1RIX-AR016 SPENT FUEL STORAGE FB 755' AH-117 ALERT and HIGH alarms simultaneously actuate.

The alarm procedure directs you to determine if the alarm is due to a spike.

How is this accomplished?

a. Direct Radiation Protection to survey the area.

b. Access DNA History Plot for the affected instrument.

c. Access the digital recorder history for the affected instrument.

d. Direct the Equipment Operator to verify radiation level on local instrument.

ANSWER: (b.)

Applicant Comment:

The applicant argued that: "Distractor a. is plausible. CPS No. 3315.03, "Radiation Monitoring (AR/PR)," Step 2.2.12 directs RP to survey the area to validate a "spike" and CPS No. 5140.08, AR/PR Annunciator – Spent Fuel Storage – 1RIX-AR016, directs RP to survey if alarm is due to a "spike.""

Facility Proposed Resolution:

"The only correct answer is b. Question asks what actions to take to confirm if the alarm is due to a "spike". The question is not asking what actions to take if it is a spike."

NRC Resolution:

Upon review of the question, the applicant's comment, and the facility's proposed resolution, the NRC determined that the original answer (b.) was the only correct answer. The question specifically asked how an applicant was to determine if an alarm associated with spent fuel storage was due to a spike. The applicant provided CPS 5140.08, Spent Fuel Storage – 1R1X-AR016, Step 1, under the sub-heading, "Operator Actions." The procedure states that if the alarm is due to a spike, then direct RP to perform survey [sic] in the monitor vicinity. This direction is provided as a response after it has been determined a spike occurred, but does not answer the question of how to determine if a spike occurred. The answer to the question is provided in a preceding procedure step, i.e, "Review monitor history to determine if alarm is due to a spike." Distractor (a.) does not answer the question and is, therefore, an incorrect answer. The answer key was not modified; distractor (b.) was retained as the only correct answer.

RO Question 54:

The plant was at near rated power when a LOCA occurred. The following plant conditions exist:

- All Control Rods are fully inserted.
- Reactor Pressure is 200 psig.
- Reactor Water Level is -155 inches on Wide Range and steady.
- The Reserve Auxiliary Transformer has locked out.
- The only ECCS pumps that will run are LPCI A and B which are injecting at maximum.
- Containment Temperature is 180°F and rising.

You have been directed to initiate Containment Spray to maintain Containment integrity. What action should you take?

a. Initiate ONE loop of Containment Spray only.

b. Initiate BOTH loops of Containment Spray.

c. Do NOT initiate either loop of Containment Sprays.

d. Initiate Shutdown Service Water in the Containment Spray mode of operation.

ANSWER: (c.)

Applicant Comment:

The applicant stated that: "The stem of the question places the examinee in the role of Reactor Operator (RO). The RO is required to follow the SRO orders. The correct action in this situation was to "push-back/challenge" the CRS on the directed action. This was not a listed answer. Therefore distractor b. could be considered correct based on the Chain of Command during implementation of EOPs."

Facility Proposed Resolution:

"The only correct answer is c. EOP-6 states to start Containment Sprays but the "finger" states "Do not use RHR you need for core cooling." The plant conditions given indicate both loops are needed for core cooling.

NRC Resolution:

The applicant correctly stated that the stem of the question placed the examinee in the role of a Reactor Operator. The question then provided a set of conditions designed to determine if the Reactor Operator examinee could recognize the correct instructions he or she would be given for a given set of plant conditions. Using systems knowledge (Low Pressure Coolant Injection/Containment Spray) and a basic understanding of radioactive release barriers, the Reactor Operator applicant should have been able to determine the correct instruction to be given for the provided plant conditions. Upon review of the applicant's comment and the facility's proposed resolution, the NRC determined that the original answer, (c.) was the only correct answer. The question stem states that reactor water level is -155 inches on wide range and steady, and that the only ECCS pumps that will run are LPCI A and B which are injecting at maximum. With water level steady and both LPCI A and LPCI B injecting at maximum, they are required for core cooling. In accordance with the CPS Emergency Operating Procedures (EOPs), an operator is not to divert any system to spray the containment when it is necessary for core cooling. A knowledgeable operator would recognize the LPCI pumps were required for injection and were not to be diverted for containment spray. Therefore, the applicant's proposed correct answer ("push-back/challenge" the CRS on the directed action) may be a correct action; however, it was not one of the choices. Distractors (a.) and (b.) (diverting a LPCI pump(s) to spray the containment) are incorrect actions under the given conditions; however, the applicant chose distractor (b.). The only appropriate action was not to initiate sprays. The answer key was not modified; distractor (c.) was retained as the only correct answer.

RO Question 58:

Following entry into EOP-1A, ATWS RPV Control, CPS 4411.08, Alternate Control Rod Insertion, directs defeating ATWS interlocks as required.

What is accomplished by defeating ATWS interlocks?

a. It bypasses RPS logic trips to drain the scram discharge volume and insert control rods.

b. It bypasses the scram signal to allow all rods to be fully inserted using normal insertion with CRD.

c. It vents the over piston area and enables control rod insertion using individual control rod scram switches.

d. Allows the scram to be reset, the scram discharge volume drained, and the CRD accumulators recharged for subsequent manual scram attempts.

ANSWER: (d.)

Applicant Comment:

The applicant argued that: "distractor (a.) is plausible based on procedure CPS 4410.00C012, DEFEATING ATWS INTERLOCKS, Section 1.2, heading which encompasses distractor (d.)." He further stated that, "Distractor a. as written is true. By defeating ATWS Interlocks [sic] results in interlocks be [sic] reset that allow subsequent scrams and insertion of control rods. Question stem [sic] does not ask for why the RPS Interlocks are defeated but for the ATWS Interlocks."

Facility Proposed Resolution:

"The only correct answer is (d.). After defeating the RPS Logic Trips the Div 1 through 4 RPS automatic scram signals are bypassed but not the actions to insert control rods. The question asks purpose [sic] for defeating all ATWS interlocks does [sic]."

NRC Resolution:

Upon review of the question, the applicant's comment, and the facility's proposed resolution, the NRC determined that the original answer (d.) was the only correct answer. The applicant argued that answer (a.) was also correct based on procedure CPS No. 4410.00C0012, Goal 1.2, which states, "Allow the scram to be reset, the scram discharge volume drained, and the CRD accumulators recharged for subsequent manual scram attempts." It should be noted that Goal 1.2 cited by the applicant in his argument that distractor (a.) is correct is the actual correct answer to this question. Distractor (a.) states, "It bypasses RPS logic trips to drain the scram discharge volume and insert control rods." Distractor (a.) is an incomplete response to the question stem. It states that RPS logic trips are bypassed, but does not mention resetting the reactor scram. Both distractors refer to draining the scram discharge volume, but distractor (a.) refers to inserting control rods while distractor (d.) correctly refers to the actual action taken, i.e., scram-reset-scram (subsequent manual scram attempts). Because distractor (a.) contains misleading information and is only partially correct, it was determined that distractor (a.) is not a correct answer. The answer key was not modified; distractor (d.) was retained as the only correct answer.

RO Question 66:

The plant is performing a down power to shift Rod patterns.

- Annunciator 5007-4D, High Vibr Turb Shaft (Pre Trip), alarms.
- NO other annunciators are LIT on this panel.
- Bearing # 9 indicates 9 mils vibration and increasing.

(1) If NO operator action is taken, what will be the impact on the Main Turbine?
(2) What operator actions should be taken at this time?

a. (1) The Main Turbine will trip.
   (2) Scram the reactor, trip the Main Turbine and fully open the vacuum breaker until Turbine speed is less than 1200 RPM.

b. (1) The Main Turbine will NOT trip.
   (2) Scram the reactor, trip the Main Turbine and fully open the vacuum breaker until Turbine speed is less than 1200 RPM.

c. (1) The Main Turbine will trip.
   (2) Scram the reactor, trip the Main Turbine and reduce vacuum to no lower than 24" Hg.

d. (1) The Main Turbine will NOT trip.
   (2) Scram the reactor, trip the Main Turbine and reduce vacuum to no lower than 24" Hg.

ANSWER: (c.)

Applicant Comment:

The applicant argued that: "Distractor (a.) also technically correct based on CPS No. 3112.01, Condenser Vacuum (CA), Limitation 6.1 which directs opening Vacuum Breaker until turbine speed is <1200 rpm. The determination of the extent of the vibration emergency is subjective and would allow both a. and c. to be correct."

Facility Proposed Resolution:

"The only correct answer is c. The Main Turbine does trip on High Vibrations. If the high vibration trip was disabled, the "TROUBLE TSI CAB" alarm would be lit. The vibration trip may be bypassed for plant evolutions but vacuum should still not be lowered to less than 24" Hg until Main Turbine speed is less than 1200 RPM."

NRC Resolution:

Upon review of the question, the applicant's comment, and the facility's proposed resolution, the NRC determined that the original answer (c.) was the only correct answer. The provided procedures stated the following: CPS 3105.01, Step 6.3.1, "The turbine should not be operated above 1200 rpm with a condenser vacuum <24" Hg (6" Hg abs)." Step 6.3.2 stated, "Vacuum should not be broken until the turbine has reached 1200 rpm, unless an EMERGENCY CONDITION exists that requires the turbine to be slowed as rapidly as possible, which includes, but is not limited to: " … "High main turbine vibration/load metallic noise." It further stated, "When desired to slow the turbine more quickly while >1200 rpm, vacuum may be reduced to as low as 23" Hg to allow a more rapid reduction of turbine speed, then vacuum can be fully broken once turbine speed is less than 1200 rpm. Another procedure, CPS 3112.01, states approximately the same information as that cited from CPS 3105.01 above. Distractor (a.) does not include any instruction to control vacuum at or above 24" Hg that is recommended in CPS 3105.01, Step 6.3.1, but only states the operator can fully open the vacuum breaker valve. Without this information, an operator could break vacuum without regard to that limitation and may possibly cause turbine damage. This makes distractor (a.) an incorrect answer. Distractor (d.), however, contains the instructions to control main condenser vacuum, and is, therefore, the correct answer. The answer key was not modified and distractor (d.) was retained as the only correct answer.

RO Question 67:

The unit is operating at near rated conditions.

Select the statement that describes the effect on the Condensate (CD) system of turning the SJAE CDSR 1A/1B selector switch clockwise one position from the "BOTH" position and the reason for that effect.

a. CD flow and pressure reduction causes low Feedwater pump suction pressure due to SJAE Condenser/OG Recombiner inlet isolation valve 1CD006B closure.

b. CD flow and pressure remain stable as SJAE Condenser/OG Condenser inlet isolation valve 1CD006A closes and the SJAE Condenser/OG Recombiner bypass valve 1CD066A opens.

c. SJAE Condenser/OG Recombiner inlet isolation valve 1CD006A closes, which lowers CD flow and CB pressure because response of the bypass valve 1CD066A is too slow to adequately compensate.

d. CD flow and pressure are reduced as SJAE Condenser/OG Recombiner inlet isolation valve 1CD006B closes then returns to previous values as the SJAE Condenser/OG Recombiner bypass valve 1CD066B opens.

ANSWER: (c.)

<u>Applicant Comment:</u>

The applicant argued that ". . . distractor (b.) is also technically correct based on CPS No. 3104.01, Condensate/Condensate Booster (CD/CB), Discussion 2.6.1 which acknowledges that dP will not drop very much, implying that the evolution is virtually seamless and plant transient is minimal."

<u>Facility Proposed Resolution</u>:

"The only correct answer is (c.). Based on plant experience CPS No. 3104.01, CONDENSATE/CONDENSATE BOOSTER (CD/CB), discusses the fact that the valve breaker is cycled to allow the system stabilize based on valve CD066A response time."

<u>NRC Resolution</u>:

Upon review of the applicant's comment, the facility provided procedures, and the facility's proposed resolution, the NRC determined that the original answer (c.) was the only correct answer. Previous plant experience has been included in CPS No. 3104.01 where it states the following at Step 2.6: "During performance of section 8.2.4, Isolating a Steam Jet Air Ejector Condenser/Off-Gas Recombiner Condenser Train, the system is expected to respond as follows: 1.) As 1CD006A/B is going shut, THEN the 1CD066B/A is going open which allows the DP to not drop very much. The 1CD066A/B was going shut as the flow in that leg went down. An operator could be stationed at 1PA05J to verify that the 1CD066A and B are behaving properly." At Step 8.2.4.1 it provides steps for opening valve 1CD006A, stating the valve control switch had to be cycled repeatedly; five 20-second steps followed by several 4-second steps, in order to control its closure to maintain stable plant conditions. This guidance showed that this evolution would cause unstable plant conditions, showing that distractor (b.) is not a correct answer. Significant, additional operator actions must be taken to maintain the plant in a stable condition for this transfer to take place. Per the examination instructions provided to each applicant, no other operator actions were to be assumed except those specifically stated in the question. Because additional actions are required to keep the plant stable during the evolution, distractor (b.) is not correct. The answer key was not modified and distractor (c.) was retained as the correct answer.

RO Question 69:

A plant cool down is in progress with Reactor pressure at 750 psig.

- The MDRFP is unavailable and Reactor water level is being controlled with the A TDRFP.
- The A TDRFP is being controlled with M/A Station in manual.
- The A TDRFP is at 2370 rpm and Reactor water level is slowly increasing.

What action must be taken to maintain Reactor water level in the normal operating band?

a. Shift Feedwater level control to the Condensate Booster Pumps.

b. Depress the DECREASE push button on the A TDRFP M/A Station.

c. Shift the TDRFP to the Startup Level Controller and DECREASE push button.

d. Shift the A TDRFP to the Manual Potentiometer and turn the potentiometer counter clockwise.

ANSWER: (d.)

Applicant Comment:

The applicant argued that "Lowering pressure to shift FW level control to CD/CB per distractor a. would be an operationally sound action and should be considered as a correct choice." The applicant goes on to say, "Implicit in aligning CD/CB for feedwater injection is to ensure that Reactor pressure is within the capacity of the Condensate Booster pumps. Guidance in CPS No. 4411.09 is provided to lower pressure to <725# to make CD/CB injection available. The question stem states that Reactor water level was trending upward, which demonstrates adequate inventory is available to support depressurization. Additionally, the TDRFP is already at its minimum setting on its controller, so sufficient makeup capacity exists. Thus, the strategy to lower pressure and align CD/CB for injection is a viable option. The action to lower pressure also depletes inventory and would compensate for rising level. There is no information in the stem that would indicate that lowering pressure within the capability of the Condensate Booster pumps would exceed the required cooldown rate (<100°F/hr.)."

Facility Proposed Resolution:

Answers (a.) and (d.) were both correct based on agreement with the candidate's challenge.

NRC Resolution:

The NRC examiners reviewed the question, the applicant's contention and the proposed facility resolution and disagreed that distractor (a.) is a correct option under the conditions presented in the question. The applicant stated that distractor (a.) was correct based on CPS No. 4411.09, RPV PRESSURE CONTROL SOURCES. This procedure is entered when directed by the EOP/SAGs. The question stem does not indicate an emergency condition exists nor has an EOP has been entered. Further, distractor (a.) does not include any statement to lower reactor pressure vessel pressure below CB discharge pressure. The submitted examination question stated that distractor (a.) was incorrect because,"the 750 psig is above the shutoff head of the Condensate Booster Pumps." For this reason the NRC examiners accepted distractor (d.) as an incorrect answer. Transferring to CD/CB under the conditions specified in the question could leave the plant in a hot shutdown condition without the ability to feed the reactor should the need arise. If turbine bypass valves are opened to reduce reactor pressure to a point where CD/CB could inject, it may cause reactor vessel level to swell to a point where the TDRFP trips. This would put the plant in a condition where pressure is too high for CD/CB injection, but no feed pump to put water in the reactor pressure vessel if needed. The question stem states that reactor pressure vessel level is rising. This is because the discharge pressure of the operating turbine driven feedwater pump (TDRFP) is higher than reactor vessel pressure and water is somehow feeding in to the vessel. In order to control (stop) the vessel level rise, the TDRFP's discharge pressure must be reduced. This is accomplished by reducing the TDRFP's speed by either tripping the TDRFP or by lowering the setting on the manual potentiometer. Tripping the TDRFP reduces the pump's speed, but leaves the plant in a condition where no feedwater can be injected into the vessel because the CD pump discharge pressure is below reactor pressure vessel pressure. Reducing the turbine's speed using the manual potentiometer is the correct answer provided in distractor (d.). Further, Procedure CPS 3006.01, Cooldown with Main Condenser, Step 8.8.3, indicates transferring to CD/CB will occur at 500-600 psig, not the 750 psig specified in the conditions of the question. The NRC does not believe it correct to accept an answer that would intentionally put the plant in a hot condition where the reactor pressure vessel does not have a ready injection source. Because reactor pressure vessel pressure is above the shutoff head of the CD pumps and no provisions are contained in the distractor that provide for reactor pressure vessel depressurization, distractor (a.) is not a correct answer. The answer key was not modified and distractor (d.) was retained as the only correct answer.

RO Question 71:

The plant is operating at near rated conditions when the following annunciators alarm:

- 5000-2A F-D INFL CNDCT HI-LO
- 5000-2B F-D EFL CNDCT HI-LO
- 5000-2C F-D System Trouble

The following RWCU parameters indicate:

- RWCU Filter Demin A flow indicates 0 gpm.
- RWCU Filter Demin B flow indicates 250 gpm
- RWCU Filter Demin B effluent conductivity indicates 1.0 μs/cm

Which one of the following would occur as a result of this event?

a. Reactor power increases and Main Steam Line radiation increases.

b. Reactor power decreases and Main Steam Line radiation increases.

c. Reactor power decreases and Main Steam Line radiation decreases.

d. Reactor power remains stable and Main Steam Line radiation remains stable.

ANSWER: (b.)

Applicant Comment:

The applicant argued that answer (a.) was also correct because the question stem did not provide a specific timeline or include the amount of resin that entered the Reactor Vessel, and industry experience has shown that resin intrusion events can result in both power increases and decreases.

Facility Proposed Resolution:

Answers (a.) and (b.) were both correct based on agreement with the candidate's challenge.

NRC Resolution:

The proposed question stated distractor (a.) was incorrect because power would decrease if there was a resin intrusion. The NRC examiners were unaware that there were operating experience reports indicating that power could increase on resin intrusion and accepted the question as written with one correct answer. Upon review of the question, the applicant's comment, the facility's proposed resolution, and two operating experience reports, the NRC determined that all of the distractors could be correct answers based on the amount of resin intrusion, the type of resin entering the reactor pressure vessel, and the current reactor plant conditions. Since resin intrusion could cause power to either increase or decrease, and since Main Steam Line radiation is a function of both power and resin quantity, an argument could be made that distractors (a.) (c.) and (d.) could also be correct. Because the question has potentially 4 correct answers, it is unacceptable and was deleted from the written exam. The answer key was modified to remove question number 71 from the examination.

RO Question 75:

The plant is operating at near rated power. HPCS is running "tank to tank" mode per CPS 9051.01, HPCS Pump and HPCS Water Leg Pump Operability.

A spurious HPCS initiation signal is received.

NO other systems are affected.

What effect if any does this have on Reactor Pressure and Reactor Power?

a. Reactor Pressure increases, Reactor Power increases.

b. Reactor Pressure decreases, Reactor Power increases.

c. Reactor Pressure decreases, Reactor Power decreases.

d. Reactor Pressure and Reactor Power remain the same.

ANSWER: (c.)

Applicant Comment:

The applicant argued that answer (b.) was also correct because the question stem did not provide a specific time-frame.

Facility Proposed Resolution:

Answers (c.) and (b.) were both correct based on agreement with the candidate's challenge.

NRC Resolution:

The proposed examination question reference stated that HPCS will shift from "tank to tank" mode to injection mode. When HPCS spray begins in the reactor pressure vessel, this will cause reactor pressure and power to decrease. For this reason the NRC examiners accepted distractor (c.) as the only correct answer. The examiners felt the stem of the question provided sufficient information to arrive at the correct answer. However, the wording in the question stem was insufficient for some applicants to determine the intent of the question. Upon review of the question, the applicant's comment, an operating experience report, and the facility's proposed resolution, the NRC determined that distractors (b.) and (c.) were correct answers. The facility provided operating experience (OpEx) stated that reactor pressure decreased from its normal operating pressure to a value about seven psig below the normal pressure, but then increased back to the plant's normal operating pressure. Reactor power decreased from 100% to approximately 93%, then increased back to 100% during the same transient. Because the time frame required to correctly answer this question was not provided, an applicant could arrive at either distractor (b.) or (c.) as a correct answer. Distractor (a.) is incorrect since reactor pressure will not increase, and distractor (d.) is incorrect because reactor pressure and power do not remain the same. The answer key was modified to accept distractors (b.) and (c.) as correct answers for question number 75.

SRO Question 3:

A reasonable expectation of operability does NOT exist for piece of equipment and the onshift SRO determines that an OPERABILITY determination evaluation is required in accordance with OP-AA-108-115, Operability Determinations.

Which ONE of the following procedures is PRIMARILY be used to track the status of the equipment?

a. LS-AA-125 Corrective Action Program Procedure

b. OP-AA-108-104 Technical Specification Compliance

c. OP-AA-108-105 Equipment Deficiency Identification and Documentation

d. OP-AA-108-111 Adverse Condition Monitoring and Contingency Planning

ANSWER: (b.)

<u>Applicant Comment:</u>

The applicant argued that "distractor (a.) is also a possible correct answer based on OP-AA-108-115, "Operability Determinations," Step 1.4, that cites the Corrective Action program for tracking actions."

<u>Facility Proposed Resolution</u>:

"The only correct answer is b. The equipment should be declared inoperable immediately. Actual LCO tracking is controlled by OP-AA-108-104, "Operability Determinations, for Technical Specification Compliance.""

<u>NRC Resolution</u>:

In the proposed examination, it stated that distractor (a.) was incorrect in this question because LS-AA-125, "Corrective Action Program," was used to track corrective actions and actions to prevent recurrence but not the actual Limiting Conditions for Operation (LCO). For this reason the NRC examiners accepted distractor (a.) as an incorrect answer. Upon review of the question, the applicant's comment, and facility's proposed resolution, the NRC determined that the original answer (b.) was the only correct answer. Because the facility stated LCO tracking is controlled by OP-AA-108-104, the answer key was not modified and distractor (b.) was retained as the only correct answer.

SRO Question 7:

The plant is operating at near rated power. The A RO reports parameters that have a potential for core shroud cracking and that NSED should be notified.

Who else must be notified?

a. NRC

b. Reactor Engineering

c. Reactor Engineering and Radiation Protection

d. Reactor Engineering, Radiation Protection and Chemistry

ANSWER: (b.)

Applicant Comment:

The applicant argued that distractor (a.) was also correct based upon procedure CPS 4008.02, "Core Shroud Cracking," Section 4.1, which stated that the Reactor Engineer (RE) was to be notified. In addition, this event was reportable per LS-AA-1110, "Reportable Event," SAF 1.4; therefore, the NRC should be notified as well.

Facility Proposed Resolution:

Distractor (b.) was the only correct answer. The indicated symptoms were indicative of a Core Shroud Cracking event, and per procedure CPS 4008.02, NSED and the RE were to be notified. Directions to perform Subsequent Actions of an Off-Normal procedure are provided by the Control Room Supervisor (CRS).

NRC Resolution:

The proposed question stated that distractor (a.) was incorrect because Reactor Engineering must be notified, not the NRC. Upon review of the question, applicant's comment, and the facility's proposed resolution, the NRC determined that the original answer, (b.) was the only correct answer for the question. The facility's procedure, CPS No. 4008.02, stated that engineering and reactor engineering must be notified; it does not address nor require a notification to the NRC. If it was determined that there was core shroud cracking, then the required notifications would be determined by the CRS after the actions of procedure CPS No. 4008.02 were completed. The NRC would then be notified. A courtesy call could be made while the actions were being conducted, but the notification is not required until core shroud cracking is verified. Because notification to the NRC was not addressed in the applicable procedure, distractor (a.) was not accepted as a correct answer. The answer key was not modified for SRO question #7; distractor (b.) was retained as the only correct answer.

WRITTEN EXAMINATIONS AND ANSWER KEYS (RO/SRO)

RO/SRO Initial Examination ADAMS Accession # ML072960319.