March 11, 2005

Mr. Christopher M. Crane
President and CEO
AmerGen Energy Company, LLC
200 Exelon Way, KSA 3-E
Kennett Square, PA 19348

SUBJECT: OYSTER CREEK NUCLEAR GENERATING STATION - SENIOR REACTOR AND REACTOR OPERATOR INITIAL RETAKE EXAMINATION REPORT NO. 05000219/2004302

Dear Mr. Crane:

This report transmits the results of the Senior Reactor Operator (SRO) and Reactor Operator (RO) initial licensing written retake examinations administered on December 10, 2004. It also documents the results of a meeting conducted at your site on February 9, 2005. The examinations addressed areas important to public health and safety and were developed and administered using the guidelines of the “Examination Standards for Power Reactors” (NUREG-1021, Revision 9).

Based on the results of these examinations, one (out of three) Senior Reactor Operator upgrade applicant passed his written retake examination. The three applicants included one upgrade SRO, one instant SRO and one RO. On January 28, 2005, final examination results, were given during a telephone call between Mr. John G. Caruso and Mr. Jesse Hackenburg of your staff.

The meeting of February 9, 2005, was conducted in response to our letter of June 24, 2004, transmitting the results of an examination conducted in April of 2004. The retake applicants for the examination noted herein were failures on the April 2004 exam. The purpose of the meeting was twofold: 1) discuss the review of the relatively high number of post examination comments on an NRC developed written exam after respective reviews by NRC staff and your facility (then 12 comments, now 13 as a result of applicant appeals after our letter of June 2004 was issued); 2) your review of the relatively low average scores on the April 2004 examinations. Additional details are included herein. The meeting was of mutual benefit from which lessons learned were obtained.

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). These records include the final examination and are available in ADAMS (Master File - Accession Number **ML041730211**; RO and SRO Written - Accession Number **ML0502270025;** Post Exam Comment Submittal - Accession Number **ML050330431**, ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Should you have any questions regarding this examination, please contact me at (610) 337-5183 or by E-mail at RJC@NRC.GOV .

Sincerely,

/RA/

Richard J. Conte, Chief
Operational Safety Branch
Division of Reactor Safety

Docket No. 50-219
License No. DPR-16

Enclosure: Initial Examination Report No. 05000219/2004302

cc w/encl:
Chief Operating Officer, AmerGen
Site Vice President, Oyster Creek Nuclear Generating Station, AmerGen
Plant Manager, Oyster Creek Generating Station, AmerGen
Regulatory Assurance Manager Oyster Creek, AmerGen
Senior Vice President - Nuclear Services, AmerGen
Vice President - Mid-Atlantic Operations, AmerGen
Vice President - Operations Support, AmerGen
Vice President - Licensing and Regulatory Affairs, AmerGen
Director Licensing, AmerGen
Manager Licensing - Oyster Creek, AmerGen
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OFFICE	RI/DRS	RI/DRS	RI/DRP	RI/DRS
NAME	CBixler/***CJB***	JCaruso/***JGC***	ABurritt/***ALB***	RConte/***RJC***
DATE	02/15/05	02/15/05	02/15/05	03/11/05

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U. S. NUCLEAR REGULATORY COMMISSION

REGION I

Docket No.	50-219
License No.	DPR-16
Report No.	05000219/2004302
Licensee:	AmerGen Energy Company, LLC
Facility:	Oyster Creek
Location:	200 Exelon Way, KSA 3-E Kennett Square, PA 19348
Dates:	December 10, 2004 (Written Retake Examination Administration) December 11, 2004 - January 21, 2005 (Licensee Initial Grading and Comments on Written Examination) December 22, 2004 - January 25, 2005 (NRC Examination Grading and Evaluation of Comments) January 25, 2005 - Exam Period End, Receipt of Final Post Examination Comments from Facility February 9, 2005, Onsite Meeting related to April 2004 Examination
Examiners:	John G. Caruso, Senior Operations Engineer (Chief Examiner) Steve Dennis, Senior Operations Engineer
Approved by:	Richard J. Conte, Chief Operational Safety Branch Division of Reactor Safety

Enclosure

SUMMARY OF FINDINGS

IR 05000219/2004302; December 10, 2004; Oyster Creek; Initial Operator Licensing Written Retake Examination.

One of three applicants passed the retake examination (i.e., one SRO upgrade passed).

The written examinations were administered by the facility. There were no inspection findings of significance associated with the examinations.

A. NRC-Identified and Self-Revealing Findings

None

B. Licensee-Identified Violations

None

REPORT DETAILS

1. REACTOR SAFETY

Mitigating Systems - Senior Reactor Operator (SRO) and Reactor Operator (RO) Initial License Written Retake Examinations

a. Scope of Review

The licensee developed the written examinations. The NRC together with site training and operations' personnel verified or ensured, as applicable, the following:

- The examinations were prepared and developed in accordance with the guidelines of Revision 9 of NUREG-1021, "Operator Licensing Examination Standards for Power Reactors." A review was conducted both in the Region I office. Final resolution of comments and incorporation of test revisions were completed.
- Simulation facility operation was proper.
- A test item analysis was completed on the written examination for feedback into the systems approach to training program.
- Examination security requirements were met.

The written examination was administered by the site training staff on December 10, 2004.

b. Findings

Grading and Results

One applicant (an SRO upgrade) passed the initial licensing written retake examination. The applicant had previously passed all other portions of the initial licensing examination and was granted a waiver for those sections.

The facility had three post-examination comments on the RO portion of the written examination. NRC resolution of these comments are attached. Based on these comment resolutions, the NRC regraded all three of the applicants' written examinations. The regrading of the written retake examinations resulted in one SRO upgrade applicant achieving a passing grade, rather than a failing grade, as originally determined by the licensee.

Examination Administration and Performance

During the exam administration there were no training deficiencies identified.

4OA6 <u>Meetings, including Exit</u>

<u>Related to the Retake Examination</u>

On January 28, 2005, the NRC provided conclusions and examination results to site management representatives via telephone. The license number for the one SRO upgrade applicant that had passed the examination was withheld pending any potential appeals on the written examination.

<u>Meeting of February 9, 2005</u>

The meeting of February 9, 2005, was conducted in response to NRC letter of June 24, 2004, transmitting the results of an examination conducted in April of 2004 for Oyster Creek. Attendees are noted herein. The retake applicants for the examination noted herein were failures on the April 2004 exam. The purpose of the meeting was twofold: 1) discuss the review of the relatively high number of post examination comments on an NRC developed written exam after respective reviews by NRC staff and your facility (then 12 comments, now 13 as a result of applicant appeals after NRC letter of June 2004 was issued); 2) facility review of the relatively low average scores on the April 2004 examinations.

Overall, the meeting was of mutual benefit from which lessons learned were obtained. The high number of post exam changes/comments reflected a certain number of problematic questions for which the validation review was not completely effective by both NRC staff and the facility. Enhancement actions related to effectiveness in the future are planned by both organizations; the focus is on improving the review of distractor quality and ensuring that the one correct answer is indeed correct. In a multiple choice written test, distractors are the three of four answer choices that are to be incorrect but plausible in order to enhance the question's discrimination value. The final examination with changes for April 2004 was viewed by NRC staff as valid and reliable.

With respect to the low average scores, the facility is reviewing needed improvements in the initial operator licensing training program including applicant screening, use of high cognitive questions, and factoring in additional lessons learned from the review of results for this retake examination. Planned improvements are also being considered for the licensed operator requalification program as applicable, such as in the area of validation reviews and the use of higher cognitive questions.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee Personnel

J. Hackenburg, Manager, Operations Training
G. Young, Supervisor of Operator Training

NRC Personnel

J. Caruso, Senior Operations Engineer
S. Dennis, Senior Operations Engineer

Attendees for Meeting of February 9, 2005

NRC Personnel
W. Lanning, Director, Division of Reactor Safety
R. Conte, Chief, Operational Safety Branch
D. Jackson, Senior Project Engineer, Projects Branch No. 2, Division of Reactor Projects

Licensee Personnel
B. Swenson, Site Vice President
D. McMillan, Director of Training
J. Hackenburg, Manager, Operations Training
J. Randich, Plant Manager
R. Detwiler, Director Operations
J. Kandasamy, Manager Regulatory Assurance.
V. Cwietniewicz, Exelon Corporate Training
H. Tritt, Operations Support Manager

State of New Jersey
R. Penna, Bureau of Nuclear Engineering.

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened/Closed/Discussed

None

NRC RESOLUTION OF LICENSEE COMMENTS

Licensee's Post Written Examination Comments Publically Available
ADAMS Accession No. ML050330431

Note: The licensee's post exam comments regarding these three questions were received by the NRC on December 22, 2004. After NRC review, additional information was requested from the licensee and provided on January 3, 2005 and January 7, 2005. During the exam there was one question from one of the applicants regarding question #39. After further review by the NRC staff and discussions with the licensee, the licensee submitted a revised comment for RO question #2, dated January 11, 2005, and received in Region I on January 18, 2005. Finally, the licensee submitted a third and final revision to the Region I staff that combined and summarized all previous discussions and submittals for these three questions dated January 21, 2005, and received in Region I on January 25, 2005.

RO question 02:

The plant has experienced a loss of offsite power, and the following conditions exist:

Buses 1C and 1D are being supplied by their respective EDGs.
RPV pressure is being maintained at 935 psig with Isolation Condensers.
Oyster Creek has been informed that offsite power will be restored no sooner than 72 hours.

If a plant cooldown is commenced at the **MAXIMUM** allowable cooldown rate, what will be the **MINIMUM** time it takes to clear the shutdown cooling interlocks, assuming a constant cooldown rate?

A. 1.9 hours

B. 2.2 hours

C. 19 hours

D. 22 hours

<u>Submitted answer explanation</u>:

Maximum allowable cooldown rate during loss of AC power is 10 deg. F/hr.
Starting temperature @ 935 psig is 538 deg. F
SDC interlocks clear @ 350 deg. F
Required cooldown of 188 deg. F to clear SDC interlocks.

A assumes a Tech Spec allowable cooldown rate of 100 deg. F/hr
B assumes the administrative limit of 90 deg. F/hr.
C is the correct answer.
D is incorrect, but plausible if a math error is committed.

Final Revised answer explanation:

On a loss of offsite power, both non-vital buses (1A/1B) are de-energized and their input supply breakers open. Additionally, both vital buses (1C/1D) are separated from the non-vital buses, and both EDGs automatically start. After approximately 10 seconds, both vital buses are energized by their respective EDG, and USS 1A2, 1A3, 1B2 and 1B3 are energized from the vital buses. After a 60-second time delay, both CRD pumps (powered from USS 1A2/1B2) automatically start as part of the diesel loading logic on a loss of offsite power. This provides high-pressure makeup capability to the RPV.

Additionally, when the non-vital buses lose power, the Fire Protection Pond Pumps lose power. These pumps are designed to keep the fire header pressurized and available. When the pond pumps trip, fire header pressure drops below 75 psig, causing both fire diesels to start and provide makeup to the fire protections system to keep it pressurized. This happens immediately following the loss of the pond pumps.

The fire protection system serves as an alternate makeup source to the Isolation Condenser shells. The normal source of makeup comes from the Condensate Transfer pumps, powered from 1B32. The breaker supplying 1B32 must be manually reset and closed following a loss of power, as this breaker has a shunt trip device to trip the breaker on low voltage. With no operator actions taken to reset 1B32, there will be no Condensate Transfer pumps available for Isolation Condenser shell makeup. However, the automatic start of both fire diesels provides more than adequate makeup capability to the Isolation Condenser shells, and it occurs automatically.

Since both fire diesels automatically start following the loss of power, there is adequate makeup capability to the Isolation Condenser shells. Since both CRD pumps automatically start following the loss of power, this ensures adequate RPV high pressure makeup capability, since use of the Isolation Condensers does NOT result in a loss of inventory. Therefore, these two automatic actions satisfy the requirement of a high pressure injection source to the RPV and adequate makeup to the Isolation Condenser shells, which then allows a maximum cooldown rate of 90 deg. F/hr., IAW ABN-36, step 3.1.7, page 5 (previously provided). FSAR sections/pages 15.2.6.2.2 (page 15.2-5), 15.2.7.2 (page 15.2-7), page 6.3-4, 6.3.1.1.2 (page 6.3-8), page 6.3-9 also support this. Therefore, the maximum cooldown rate is 90 deg. F/hr and the correct answer is B.

Maximum allowable cooldown rate during loss of AC power meeting the conditions in ABN-36, step 3.1.7 is 90 deg. F/hr.
Starting temperature @ 935 psig is 538 deg. F
SDC interlocks clear @ 350 deg. F
Required cooldown of 188 deg. F to clear SDC interlocks.

A is incorrect, as it assumes a Tech Spec allowable cooldown rate of 100 deg. F/hr
B while the administrative limit of 90 deg. F/hr is correct for this condition, a calculational rounding error was committed (see below), which makes this answer incorrect.
C is incorrect, using 10 deg. F/hr cooldown rate.
D is incorrect, but plausible if a math error is committed using 10 deg. F/hr cooldown rate.

Calculation of time to reach SDC interlocks is 188 deg. F divided by 90 deg. F/hr which equals 2.088 hrs. Even when rounded to the nearest tenth, the calculated minimum time is 2.1 hours.

Since none of the distractors state this particular answer, there is no correct answer for this question.

Therefore, Oyster Creek recommends deleting this question.

<u>NRC Resolution</u>:

The NRC staff reviewed the documentation provided by the licensee in support of their proposed change to the original correct answer. Specifically, the information contained in the FSAR sections and pages noted above, ABN-36, "Loss of Offsite Power," alarm response procedures, and station electrical load diagrams. The examiners noted the following:

FSAR section 15.2.6.2.2. "Loss of All AC Power - Event Description - Assumptions," states, in part, "The relief valves and Isolation Condensers would be available for decay heat removal" and "A control rod drive (CRD) hydraulic pump, powered from the diesel generators, can supply 110 gpm makeup flow to the reactor." The CRD pump would be a high pressure makeup system available to the reactor pressure vessel (RPV).

Alarm Response Procedures N-2-a, N-3-a, N-2-b, and N-3-b, for the Fire Protection System describe the automatic actions which occur upon a loss of the fire pond pumps which would occur under a loss of offsite power (LOP) condition. As stated in the automatic actions section of the alarm response, if a LOP were to occur causing the pond pumps to trip, the diesel driven fire pumps would auto start to maintain fire water header pressure.

ABN-36, step 3.1.3 states, "Maintain the cooldown rate less than 10 degrees per hour." However, ABN-36, step 3.1.7 states, "When adequate high pressure makeup is available to the RPV and Isolation Condenser shells, then MAINTAIN RPV cooldown rate #90 degrees per hour."

Based upon the information in the FSAR and alarm response, the requirements, as stated in ABN-36, for a cooldown rate of 90 degrees per hour would be met. A high pressure makeup system for the RPV (CRD) and for the Isolation Condenser shells (Fire Water via the Diesel Fire Pumps) would be available given the conditions stated in question stem. However, given a 90 degree per hour cooldown rate and the required cooldown of 188 degrees to clear the shutdown cooling interlocks, the correct answer would be 2.1 hours (rounded up from 2.088) which is not a choice available from the possible answers provided.

Conclusion

In conclusion, the NRC staff accepts the licensee comment that there was no correct answer for question # 02 and it should be deleted.
RO question 35:

During startup, when the reactor is critical, SRM detectors are initially withdrawn when _____(1)_____ and SRM period will _____(2)_____ as the detectors **INITIALLY** start moving

A. (1) all SRMs are greater than 1 E5 cps
(2) become longer

B. (1) all SRMs are greater than 1 E5 cps
(2) become shorter

C. (1) three IRMs in each RPS system read 50% on range 1
(2) become longer

D. (1) three IRMs in each RPS system read 50% on range 1
(2) become shorter

Submitted answer explanation:

As SRM detectors are initially withdrawn, the move into a higher flux area of the core causing period to become shorter. Period will become longer/go negative only after the detectors pass beyond the high flux area.

A is incorrect, detector withdrawal is dictated by IRM indication and period will initially go shorter
B is incorrect, detector withdrawal is dictated by IRM indication
C is incorrect, as period will initially become shorter
D is correct

Revised answer explanation:

When this question was developed, past knowledge of the BWR-5 product line was used as a basis for the question. In BWR-5s, the SRM detector position when fully inserted is 3 feet (36 inches) above the core mid-plane position. This puts the SRM detectors closer to, and in most cases, within the area of highest flux when taking the core critical. If the SRM detectors are within the high flux area, pulling the detectors will result in the detectors “seeing” more flux per unit time as the detectors are pulled (a Doppler effect), causing period to become shorter. However, in BWR-2s, the SRM detectors are only 1.5 feet (18 inches) above the core mid-plane. Since our rod sequence sheets are arranged such that criticality occurs nominally between notch position 8 and 12, the area of highest flux will be ABOVE the SRM detectors. As the detectors are being withdrawn, each detector will see a reduction of counts per unit time (again, a Doppler effect), causing period to become longer (trend toward infinity.)

After a lengthy conversation with the Reactor Engineering Manager at Oyster Creek, we realize our original thought process was wrong. Our pull sheets are designed to achieve criticality in the range of Notch position 8 to 12. Since the Oyster Creek SRM detectors are located 18 inches above the core mid-plane when fully inserted, this corresponds to Notch position 18 when fully inserted. Therefore, the highest area of flux will be **above** the SRM detectors when we go critical. Pulling the SRM detectors will therefore cause SRM period to become longer as the detectors are withdrawn, since the detectors are seeing less flux per unit time as they are being withdrawn away from the high flux area above them.

A review of SRM data from the most recent reactor startup following our refueling outage supports this fact. When IRMs were above 50% of range 1 per procedure, all 4 SRM detectors were withdrawn. As soon as the SRM detectors began moving outward, SRM periods immediately became longer, trending toward infinity. Additionally, a simulator startup was performed to replicate the question conditions. The reactor went critical with controlling group rods between notch position 10 and 12. When all 8 IRMs had been ranged up to range 2, all 4 SRM detectors were withdrawn. As soon as detector movement was initiated in the outward direction, and all 4 detector periods immediately became longer. As SRM detectors are initially withdrawn, they move further away from a higher flux area of the core, causing period to become longer.

Therefore, corrected explanations are:

A is incorrect, since detector withdrawal is dictated by IRM indication
B is incorrect, since detector withdrawal is dictated by IRM indication and period will initially go longer
C is correct
D is incorrect, as period will initially become longer

Oyster Creek recommends changing the correct answer to C.

NRC Resolution:

In certain model BWRs, dependent on core loading, as SRM detectors are initially withdrawn, they move into a higher flux area of the core causing period to initially become shorter. Period will become longer/go negative only after the detectors pass beyond the high flux area. Based upon that information the initial answer was correct. However, given the loading of the Oyster Creek core, the location of the high flux area during startup, and the location of the SRMs in the Oyster Creek BWR-2 reactor, the licensee stated that the correct answer was actually choice **C,** that reactor period would initially become longer.

The NRC staff reviewed information provided by the licensee which documented graphically how the SRMs responded during the most recent startup. Additionally, the staff reviewed data tables documenting SRM response in the simulator startup which very closely modeled actual plant response. In all documents reviewed by the staff, SRM response initially became longer, which supports the licensee position that the correct answer should have been **C**.

Conclusion

In conclusion, the NRC staff accepts the licensee comment that the correct answer to question # 35 is **C**.

RO question 39:

Given the following:

The plant is shutdown
RPV pressure is 700 psig
An I&C tech causes drywell pressure transmitter RE04A to fail upscale
RWCU system isolates
RE04A has been returned to service and drywell pressure is 1.2 psig
RWCU pressure is 140 psig
Filter bypass valve V-16-83 is OPEN

Based on the above, the RWCU system **SHOULD** _____(1)_____ and
the following actions are required, **in the stated sequence**, to open V-16-1: _____(2)_____

A. (1) have isolated
(2) Depress DW ISOLATION RESET pushbutton on 4F, and reduce RWCU pressure to approximately 80 psig.

B. (1) **NOT** have isolated
(2) Reduce RWCU pressure to approximately 80 psig, and depress DW ISOLATION RESET pushbutton on 4F.

C. (1) have isolated
(2) Depress DW ISOLATION RESET pushbutton on 4F, reset the redundant high pressure isolation keylock in A/B Battery Room, and reduce RWCU pressure to approximately 80 psig.

D. (1) **NOT** have isolated
(2) Reduce RWCU pressure to approximately 80 psig, depress DW ISOLATION RESET pushbutton on 4F, and reset the redundant high pressure isolation keylock in A/B Battery Room.

Submitted answer explanation:

In order for the RWCU system to isolate on either high drywell pressure or low-low RPV level, it takes two (one-out-of-two twice) logic to cause this isolation, as the isolation circuitry is part of the RPS system and uses the same detectors the RPS circuitry uses for these isolation and initiation signals. However, a high system pressure causes the system to isolate. There are two isolation circuits for high pressure: the primary circuit, and the Appendix R redundant

circuit. Both circuits are currently set at 130 psig for the isolation function, and either one will cause the system to isolate. Therefore:

A is incorrect, as isolation should not have occurred.
B is incorrect, as the redundant high pressure trip must be reset also.
C is incorrect, as isolation should not have occurred.
D is correct

Revised answer explanation:

Besides the containment isolation signals (high drywell pressure or low-low RPV level), high system pressure also causes the system to isolate. There are two isolation circuits for high pressure: the primary circuit, and the Appendix R redundant circuit. Both circuits are currently set at 130 psig for the isolation function, and either one will cause the system to isolate.

Based on the conditions stated in the stem, RWCU system pressure of 140 psig would trip the system on both high pressure trip signals. Since there is no timeline or indication that the RWCU pressure rose to 140 psig SUBSEQUENT to the isolation on high drywell pressure, it is valid that the candidates would choose an answer stating the system “SHOULD have isolated,” since a valid isolation signal is present in the question stem. However, neither final part of the possible answers depicting the system should have isolated contains the correct sequence of operation needed to recover the system. Therefore, since neither of the distracters (A & C) that state the system “SHOULD have isolated” has the subsequent actions in the correct order, there is no correct answer for this question.

Oyster Creek recommends deleting this question.

NRC Resolution:

The NRC staff reviewed the conditions stated in the question stem and also reviewed the criteria for a Reactor Water Cleanup Pump trip as stated in ARP D-3-b, “Cleanup System.” The ARP stated that the system would isolate with system pressure > 130 psig and, therefore, a pump trip would occur on low suction pressure. Therefore, it is a valid answer, for the initial part of the question, that the system “SHOULD have isolated,” since a valid isolation signal is present in the question stem. Therefore, the only distractors which answer the first part of the question correctly are in distractors A and C.

The second portion of the question asks what is the correct sequence to perform followup actions to open valve V-16-1. The NRC staff reviewed the guidance contained in Procedure No. 303 Rev.88, “Reactor Cleanup Demineralizer System,” step 4.3.6, for reopening valve V-16-1 and found that distractors A and C did not state the correct method or sequence to open the valve.

Therefore, given that distractors B and D were incorrect from the first part of the question and distractors A and C were incorrect for the last part of the question, no complete correct answer was provided for the question.

Conclusion

In conclusion, the NRC staff accepts the licensee comment that there is no correct answer for question #39 and it should be deleted.