August 4, 2005

Mr. Christopher M. Crane
President and CNO
Exelon Nuclear
Exelon Generation Company, LLC
200 Exelon Way KSA 3-E
Kennett Square, PA 19348

SUBJECT: LIMERICK GENERATING STATION UNIT 1 AND UNIT 2 AND PEACH BOTTOM ATOMIC POWER STATION UNIT 2 AND UNIT 3 LIMITED SENIOR REACTOR OPERATOR INITIAL EXAMINATION REPORT NOS. 05000352/2005302, 0500353/2005302, 05000277/2005302 AND 05000278/2005302

Dear Mr. Crane:

This report transmits the results of the Limited Senior Reactor Operator (LSRO) licensing examinations conducted by the NRC during the period of June 13-15, 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 the examination, all three applicants passed all portions of the examination. The applicants included three LSROs. Examination results indicated that the applicants were well prepared for the examination. Mr. J. Caruso, Chief Examiner, discussed performance insights observed during the examination with Mr. C. Rich on June 15, 2005. On July 21, 2005, final examination results were given during a telephone call between Mr. J. Caruso and Mr. C. Rich of your training organization. The issuance of license numbers will be delayed pending receipt of written certification from Exelon to the U.S. Nuclear Regulatory Commission stating that the applicants have acquired all experience for which they were previously granted waivers.

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 ML042440215; LSRO Written - Accession Number ML052060104; LSRO Operating Sections A and B- Accession Number ML0052060114. Note for LSRO examinations there is no Operating Section C. 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
Operations Branch
Division of Reactor Safety

Docket Nos. 50-352, 50-353, 50-277, 50-278
License Nos. NPF-39, NPF-85, DPR-44, DPR-56

Enclosure: Initial Examination Report Nos. 05000352/2005302, 05000353/2005302, 05000277/2005302 and 05000278/2005302

cc w/encl:
Chief Operating Officer, Exelon Generation Company, LLC
Site Vice President, Peach Bottom Atomic Power Station
Plant Manager, Peach Bottom Atomic Power Station
Regulatory Assurance Manager - Peach Bottom
Senior Vice President, Mid-Atlantic Operations
Vice President - Operations Support
Vice President - Licensing and Regulatory Affairs
Director, Licensing and Regulatory Affairs, Exelon Generation Company, LLC
Vice-President, General Counsel and Secretary
J. Bradley Fewell, Assistant General Counsel, Exelon Nuclear
D. Quinlan, Manager, Financial Control, PSEG
R. McLean, Power Plant and Environmental Review Division (MD)
Director, Nuclear Training, Peach Bottom
Manager, Operations Training, Peach Bottom
Correspondence Control Desk
Site Vice President - Limerick Generating Station
Plant Manager, Limerick Generating Station
Regulatory Assurance Manager, Limerick
Senior Vice President - Nuclear Services
Vice President - Mid-Atlantic Operations
Manager, Licensing - Limerick Generating Station
Director, Training, Limerick Generating Station
Chairman, Board of Supervisors of Limerick Township
D. Allard, Director, Pennsylvania Bureau of Radiation Protection (SLO, PA)
Commonwealth of Pennsylvania (c/o R. Janati, Chief, Division of Nuclear Safety,
Pennsylvania Bureau of Radiation Protection)
R. Fletcher, Maryland Department of Environment
T. Snyder, Director, Air and Radiation Management Administration,
Maryland Department of the Environment (SLO, MD)

Public Service Commission of Maryland, Engineering Division
Board of Supervisors, Peach Bottom Township
B. Ruth, Council Administrator of Harford County Council
Mr. & Mrs. Dennis Hiebert, Peach Bottom Alliance
TMI - Alert (TMIA)
J. Johnsrud, National Energy Committee, Sierra Club
Mr. & Mrs. Kip Adams
S. Glenn, INPO

Distribution w/encl: **(via e-mail)**
S. Collins, RA
M. Dapas, DRA
A. Blough, DRS
R. Conte, DRS
J. Caruso, DRS
C. Bixler, DRS
M. Shanbaky, DRP
A. Burritt, DRP
F. Bower, DRP - NRC Senior Resident Inspector
D. Schroeder, DRP - NRC Resident Inspector
S. Schmitt, DRP - NRC Resident OA
S. Lee, RI OEDO
J. Clifford, NRR
D. Roberts, NRR
G. Wunder, PM, NRR
T. Tate, PM, NRR
S. Hansell, DRP - Senior Resident Inspector
C. Colantoni, DRP - Resident Inspector
T. Haverkamp - Resident OA
Region I Docket Room (with concurrences)
ROPreports@nrc.gov

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DATE	07/28/05		07/26/05		07/26/05		08/04/05	

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REGION I

Docket Nos.	05000352, 05000353, 05000277, 05000278
License Nos.	NPF-39, NPF-85, DPR-44, DPR-56
Report Nos.	05000352/2005302, 05000353/2005302, 05000277/205302, 05000278/2005302
Licensee:	Exelon Generating Company
Facility:	Limerick Generating Station and Peach Bottom Atomic Power Station
Dates:	June 15, 2005 (Written Examination Administration) June 13-14, 2005 (Operating Test Administration) June 22, 2005 (Licensee Grading Complete) June 16, 2005 - July 19, 2005 (Examination Grading)
Examiners:	J. Caruso, Senior Operations Engineer (Chief Examiner) G. Johnson, Operations Engineer
Approved by:	Richard J. Conte, Chief Operations Branch Division of Reactor Safety

SUMMARY OF FINDINGS

IR 05000352/2005302, 05000353/2005302, 05000277/2005302 and 05000278/2005302; June 13-15, 2005; Limerick Generating Station and Peach Bottom Atomic Power Station; Initial Limited Senior Operator Licensing (LSRO) Examination. Three of three applicants passed the examination (3 LSROs).

The written examinations were administered by the facility and the operating tests were administered by 2 NRC region-based examiners. There were no inspection findings of significance associated with the examinations.

A. Inspector Identified Findings

No findings of significance were identified.

B. Licensee Identified Findings

No findings of significance were identified.

REPORT DETAILS

1. REACTOR SAFETY

Mitigating Systems - Limited Senior Reactor Operator (LSRO) Initial License Examination

a. Scope of Review

The facility developed the written and operating initial examination and together with NRC Region I examiner staff verified or ensured, as applicable, the following:

- The examination was 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 and at the Limerick plant and training facility. Final resolution of comments and incorporation of test revisions were conducted during and following the onsite preparation week.

- 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 NRC examiners administered the operating portion of the examination to all applicants from June 13-14, 2005. Limerick training staff administered the written examination on June 15, 2005.

b. Findings

Grading and Results

Three of three applicants (3 LSROs) passed all portions of the initial licensing examination.

The facility had two post-examination comments. These are detailed in Attachment 2.

One procedure enhancement issue was identified during the administration of the Operating Test at Limerick. This was communicated to the licensee.

Examination Administration and Performance

No findings of significance were identified.

4OA6 Exit Meeting Summary

On July 21, 2005, the NRC provided conclusions and examination results to a Limerick management representative via the telephone. License numbers for the applicants were not provided at this time pending receipt of written certification from Exelon to the U.S. Nuclear Regulatory Commission stating that the applicants have acquired all experience for which they were previously granted waivers. The NRC expressed appreciation for the cooperation and assistance provided by the licensee's training staff during the preparation and administration of the examination.

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

<u>Licensee Personnel</u>
J. White, Director of Training
C. Rich, Manager, Operator Training
C. Fritz, Facility Exam Development Team
C. Goff, Facility Exam Development Team

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

ITEM NUMBER	TYPE	DESCRIPTION
NONE		

ATTACHMENT 2

Licensee's Post Written Examination Comments Publically Available
ADAMS Accession No. ML052060249

The licensee's post exam comments regarding Questions 26 and 37 were received by the NRC on June 17, 2005. After initial review, the NRC provided comments to and requested additional information from the licensee on June 20, 2005. After further review by the NRC staff and discussions with the licensee, the licensee submitted revised comments for both LSRO questions #26 and #37, dated and received in Region I on June 22, 2005 (see below). The NRC provided the licensee further comments on July 5, 2005 regarding question #37 and certain stated assumptions made by the applicants and the fact that under the conditions stated in the stem that there was no technical basis for assuming that Fuel Pool Cleanup was either NOT running at the onset, nor for assuming it was lost following the event. The licensee submitted a third and final revision for question #37 on July 11, 2005. During the exam there were no questions from the applicants regarding either LSRO question #26 or #37. The NRC's resolution for these two post exam comments is based on the independent reviews that were conducted by both NRC examiners assigned to the exam team (plus a third independent examiner for the final submittal on question #37) as well as the Branch Chief. There was some additional input from a DRP Engineer who was a former Peach Bottom SRO.

<u>Original LSRO Question 37</u>:

PBAPS Unit 2 plant conditions are as follows:

- Mode 5
- Core Shuffle Part 1 has just begun
- RBCCW is backing up TBCCW
- The CRD system is in service
- The RWCU system is in service in a normal lineup dumping 60 GPM to the Main Condenser
- A fire header break in the RBCCW system room has caused both RBCCW pumps to trip

WHICH ONE of the following describes the operational implications of this condition?

A. Higher than normal plant dose rates.

B. Loss of Instrument Air to the Refueling Bridge

C. Reactor Cavity and fuel pool visibility will degrade

D. Reactor Cavity and Fuel Pool water level will begin to lower.

A2-2

Originally designated correct answer “D” <u>Submitted answer explanation (original):</u>

	Justification
a.	Incorrect - RWCU will not isolate on high temperature. The isolation temperature is 200 degrees F but the Reactor Cavity temperature will be between 110 F and 130 F during refueling operations. Generally, the temperature is maintained well below 110 F; around 90 F.
b.	Incorrect - The Refueling Bridge at PBAPS has an air compressor mounted on the bridge and is, therefore, independent of station air systems.
c.	Incorrect- RWCU will not isolate on high temperature. The clarity of the Reactor Cavity will not change due to this event.
d.	Correct- With RBCCW supplying TBCCW loads, RBCCW is supplying cooling water to the CRD pump lube oil coolers and thrust bearings. The CRD pumps will trip after loss of RBCCW. The loss of 60 GPM from the CRD system into the reactor cavity will cause Reactor Cavity and Fuel Pool Levels to slowly lower. RWCU Dump Flow is still in service and would lower cavity level at the rate of 60 GPM. This question tests differences between LGS and PBAPS.

<u>Licensee’s Original Justification for Change</u>

Question Number:

37 (Missed by all three candidates)

Facility Regrade Request:

Change the correct answer to “c”

Justification:

It also provides that Control Rod Drive System (CRD) is in service, which provides a normal flow of approximately 60 gpm to the reactor vessel. With RWCU dump flow out of the reactor compensating for CRD flow into the reactor, reactor cavity/spent fuel pool level will be stable.

The question then states that both RBCCW pumps trip. The answer key indicates that the operational implication of this would be that Reactor Cavity and Fuel Pool water level will begin to lower, which is answer “d”. The justification for this on the answer key states that “The CRD pumps will trip after a loss of RBCCW.” If this were the case and the running RWCU pump remains running, then reactor cavity/spent fuel pool level would lower.

However, there is no direct trip of the CRD pump due to a loss of RBCCW flow. The PBAPS Initial Licensed Operator Training lesson plan for Control Rod Drive Hydraulic System, PLOT-5003A states on page 18 of 24, under “Interlocks”, that the pump will trip on low suction pressure and various electrical malfunctions. This is also supported by the Annunciator Response Card for CRD WATER PUMP TRIP (ARC-211, F-1 and G-1) that lists only “Low suction pressure” and “Motor overcurrent” as the automatic trips of the CRD pumps. The lesson plan also states (on page 20 of 24) that "A loss of TBCCW and RBCCW will cause the CRD pump to overheat." Therefore, the CRD pump will remain running even with a loss of TBCCW and RBCCW.

Since RBCCW also cools the RWCU pump motor coolers (see Design Basis Documents P-S-33 for RBCCW and P-S-36 for RWCU) a loss of RBCCW will result in an automatic trip of the RWCU pump due to high temperature in the RWCU pump motor windings at a setpoint of 149 deg. F. This is supported by ARC-215, A-2 and B-2, which are provided. In addition, the RWCU System Manager at PBAPS, Luis Feliu (717-456-3634) indicated that this trip would occur "fairly soon" after a loss of RBCCW at rated conditions. He also indicated that with the reactor shutdown and cooled down to a temperature typically seen during a refueling outage, the high temperature trip setpoint may take longer to reach due to the absence of heat conduction input from the system, but would still reach the high temperature trip setpoint due to the heat generated due to the motor winding current. This was also confirmed by the alternate RWCU System Manager at PBAPS, who was the previous RWCU System Manager, as well as engineering personnel at LGS, which has similar RWCU pumps. The System Managers also indicated even with the RPV flooded up to normal level for refueling operations, no dump flow would be expected after RWCU pump trip, due to the lack of RPV pressure and the high headloss of the circuitous RWCU dump flowpath (RPV pressure will be approximately 0 psig since the reactor is in Mode 5 with Core Shuffle Part 1 in progress). When the author of this question was asked why RWCU was assumed to remain in service, he responded that he overlooked the high motor winding trip for the RWCU pumps.

The candidates were interviewed to determine why answer "c" was chosen. The candidates indicated that they assumed the Fuel Pool Cleanup System was not in service because it was not stated in the question stem. The candidates also indicated that they were aware of operational configurations where the Fuel Pool Cleanup System is not in service. The Fuel Pool Cleanup System is routinely removed from service for other outage related activities (i.e., Service Water System maintenance). When the Fuel Pool Cleanup System is removed from service, the Residual Heat Removal (RHR) System must be placed in service to remove the decay heat from the reactor cavity and spent fuel pool, as specified in P-S-09, Residual Heat Removal System. When the RHR System is being used to remove decay heat from the reactor cavity and spent fuel pool, hot water from the reactor core rises to the surface of the reactor cavity, flows into the reactor cavity weir opening and through the transfer canal into the fuel pool, and ultimately flows into the skimmer surge tanks. The water that entered the skimmer surge tanks is cooled and returned to either the reactor vessel or the spent fuel pool. The natural circulation currents cause particulate from the reactor cavity to be transported into the spent fuel pool, degrading overall spent fuel pool water quality. Since none of the particulate is being removed via the RWCU System, corrosion products will continue to accumulate. These particles diminish visibility due to the light from the installed spent fuel pool and reactor cavity lights reflecting off the particles, resulting in the appearance of a haze in the water. The effect worsens as the particle concentration in the reactor vessel increases.

A loss of RBCCW or RWCU does not require suspension of fuel handling activities, and fuel movements would continue, causing more corrosion products to be dispersed in the water. Since the concentration of corrosion product particles will rise in the reactor cavity and spent fuel pool water when RWCU is out of service, and RWCU pumps will trip, visibility will degrade over time without RWCU in service. Per OP-PB-112-101-1011, Fuel Handling Director Shift Turnover Checklist, the LSRO is required to verify once per shift that conditions support error-free operations, including water clarity. This is required due to the possibility that visibility will degrade slowly over a period of time, and requires an LSRO to assess conditions at least once per shift to determine if fuel handling activities may safely continue.

Since the running RWCU pump will trip on high motor winding temperature, and the CRD pump will not trip, reactor cavity/spent fuel pool level will actually rise very slowly. This makes answer “d” incorrect. Also, since the running RWCU pump trips, the clarity of the water in the reactor cavity/spent fuel pool will degrade, making “c” the correct answer.

Therefore, the Facility Licensee requests the correct answer for Question 37 be changed to “c”.

NRC Initial Response

For answer “c” to be correct, your explanation does not address or explain how the loss of RWCU (proposed as an explanation for "D" being incorrect) would affect the Fuel Pool visibility as long as the Fuel Pool Cooling/Cleanup remains in service. Information in the stem is consistent with Fuel Pool Cooling/Cleanup in service and remaining in service for the duration of the event. The "normal" configuration during Core Shuffle Part 1 would be to have Fuel Pool Cooling system in service This system is needed both for decay heat removal and to establish/maintain clarity in the Fuel Pool. The use of RHR Fuel Pool Cooling assist would NOT be the "normal" configuration during Core Shuffle Part 1 since it has a tendency to introduce flow disturbances in the core area. It was also confirmed by Limerick Training that any scheduled outage of Fuel Pool Cooling would NOT normally be done during Core Shuffle Part 1. Rather it would be done during Core Shuffle Part 2 when there is lower decay heat load.

Excerpts from the System Description for the "Fuel Pool Cooling and Cleanup System", P-S-52, Revision 5 (provided by the facility licensee) include the following:

1. The Fuel Pool Cooling and Cleanup System has sufficient cooling capacity to maintain the Spent Fuel Pool water at a temperature at or below 150F for normal decay heat load with two pumps and two heat exchangers operating.

2. If an abnormally large heat load is placed in the spent fuel pool, a cooling train of the RHR System consisting of an RHR pump and heat exchanger, is substituted for the Fuel Pool Cooling pumps and Heat exchangers.for cooling the pool water. The conditions under which cooling of the Spent Fuel Pool water by the RHR System alone would be required include the unloading of a full core of irradiated fuel into the pool.

3. Additional heat removal capability may be needed when full core off-loading occurs and less than three Fuel Pool Cooling Pumps/heat exchangers are available.

4. The service water system shall support operation of the Fuel Pool Cooling System by providing cooling flow at a rate of 800 GPM.
5. Design basis... minimize corrosion product buildup and control water clarity through filtration and demineralization.

6. The Fuel Pool Cooling and Cleanup System performs the decay heat removal function whenever spent fuel is stored in the Spent Fuel Pool, including refueling.

As specified in NUREG -1021, Appendix E. The applicant is directed to ask questions "if you have any questions concerning the intent or the initial conditions of a question". No applicant asked any questions regarding question 37 during the exam. The NUREG goes on to specify that the applicant should "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".

Fuel Pool Cleanup would NOT be lost as a consequence of the loss of RBCCW. It would appear that the RHR System lineup would only be required if there was a Fuel Pool Cooling System partial or full outage. There would be normally NO planned outage of Fuel Pool Cooling during Core Shuffle Part 1. No outage of Fuel Pool Cooling equipment was specified in the stem. Nor was there any mention of a "full core off-load". Therefore, the applicant would have no technical basis for assuming that Fuel Pool Cleanup was either NOT running at the onset, nor for assuming it was lost following the event. Therefore, as presented in your submittal, there appears to be no basis for concluding that the stem conditions would result in degrading Fuel Pool visibility.

Revised and Final Facility Regrade Request:

Change the correct answer to “c.”

Justification:

The question provides that Reactor Building Closed Cooling Water System (RBCCW) is backing up Turbine Building Closed Cooling Water System (TBCCW), and Reactor Water Cleanup (RWCU) is dumping 60 gpm to the main condenser. It also provides that Control Rod Drive System (CRD) is in service, which provides a normal flow of approximately 60 gpm to the reactor vessel. With RWCU dump flow out of the reactor compensating for CRD flow into the reactor, reactor cavity/spent fuel pool level will be stable.

The question then states that both RBCCW pumps trip. The original answer key indicates that the operational implication of this would be that Reactor Cavity and Fuel Pool water level will begin to lower, which is answer “d.” The justification for this on the answer key states that “The CRD pumps will trip after a loss of RBCCW.” If this were the case and the running RWCU pump remains running, then reactor cavity/spent fuel pool level would lower.

However, there is no direct trip of the CRD pump due to a loss of RBCCW flow. The PBAPS Initial Licensed Operator Training lesson plan for Control Rod Drive Hydraulic System, PLOT-5003A states on page 18 of 24, under “Interlocks,” that the pump will trip on low suction pressure and various electrical malfunctions. This is also supported by the Annunciator Response Card for CRD WATER PUMP TRIP (ARC-211, F-1 and G-1) that lists only “Low suction pressure” and “Motor overcurrent” as the automatic trips of the CRD pumps. The lesson plan also states (on page 20 of 24) that “A loss of TBCCW and RBCCW will cause the

CRD pump to overheat." Therefore, the CRD pump will remain running even with a loss of TBCCW and RBCCW.

Since RBCCW also cools the RWCU pump motor coolers (see Design Basis Documents P-S-33 for RBCCW and P-S-36 for RWCU) a loss of RBCCW will result in an automatic trip of the RWCU pump due to high temperature in the RWCU pump motor windings at a setpoint of 149 deg. F. This is supported by ARC-215, A-2 and B-2, which are provided. In addition, the RWCU System Manager at PBAPS, indicated that this trip would occur "fairly soon" after a loss of RBCCW at rated conditions. He also indicated that with the reactor shutdown and cooled down to a temperature typically seen during a refueling outage, the high temperature trip setpoint may take longer to reach due to the absence of heat conduction input from the system, but would still reach the high temperature trip setpoint due to the heat generated due to the motor winding current. This was also confirmed by the alternate RWCU System Manager at PBAPS, who was the previous RWCU System Manager, as well as engineering personnel at LGS, which has similar RWCU pumps. The System Managers also indicated even with the RPV flooded up to normal level for refueling operations, no dump flow would be expected after RWCU pump trip, due to the lack of RPV pressure and the high headloss of the circuitous RWCU dump flowpath (RPV pressure will be approximately 0 psig since the reactor is in Mode 5 with Core Shuffle Part 1 in progress). When the author of this question was asked why RWCU was assumed to remain in service, he responded that he overlooked the high motor winding trip for the RWCU pumps. Since RWCU will trip, and CRD may or may not trip, reactor cavity and spent fuel pool level will not lower, therefore, answer "d" is not correct.

Spent fuel bundles are covered with a loose coating of corrosion products. Some of these corrosion products will easily detach from the bundles when they are moved through the water. With fuel shuffle part 1 in progress, corrosion products will be deposited in the reactor cavity water as the bundles are removed from the core. Documentation of this is seen in the Operations Narrative Logs from PBAPS Refueling Outage 3R14. At 2032 on 9/21/2003, Fuel Shuffle Part 1 commenced. While this log entry does not specify this Fuel Shuffle as being Part 1, subsequent log entries at 2356 on 9/21/2003 and 0430 on 9/22/2003 confirm this as being Shuffle Part 1. The only activities scheduled to be performed in the reactor vessel during Shuffle Part 1 are fuel movements and some in-vessel visual inspection activities. At 0122 on 9/23/2003, RHR Shutdown Cooling was removed from service temporarily for "fuel pool clarity." When RHR is in operation in normal Shutdown Cooling mode or Fuel Pool to Reactor Mode, the discharge of the operating RHR pump is to the bottom head area via the jet pump discharge. The forced flow of water upward through the reactor tends to push corrosion products up, where RWCU has difficulty removing it. Removal of shutdown cooling is one option to allow the corrosion products to be drawn down to the RWCU pump suctions from the recirculation piping and the bottom head drain. This series of log entries shows a degradation in clarity as Fuel Shuffle Part 1 progresses. The effect is obviously worsened if RWCU trips and is out of service, since no removal of corrosion products will occur down in the reactor core area. The RWCU trip results in a reduction in filtration of the reactor cavity water, resulting in a higher concentration of corrosion products, and a degradation of the visibility of the reactor cavity water. Simply put, if the initial conditions assume a given amount of filtration, and some of that filtration is lost, less corrosion particles will be removed, and visibility will degrade. Furthermore, the only filtration system that may still be in service takes water only from the surface of the reactor cavity, spent fuel pool, and equipment pit. Any corrosion products in the water are required to travel to the surface to be removed. These particles diminish visibility due to the light from the installed spent fuel pool and reactor cavity lights reflecting off the particles, resulting in the appearance of a haze in the water. The effect worsens as the particle concentration increases. Since the fuel handling crew on the refuel platform is now required to look through more corrosion products in the water, visibility is degraded.

As shown on PBAPS P&ID M-363, sheet 1, water flows from the reactor cavity, spent fuel pool, and equipment pit to the skimmer surge tanks. Since each of these three bodies of water have four returns to the skimmer surge tanks and are at the same height, an approximately equal amount of water flows from each area to the skimmer surge tanks. The typical fuel pool cooling alignment for Core Shuffle Part 1 is two or three fuel pool cooling pumps, heat exchangers, and demineralizers. Peach Bottom procedure SO 19.1.A-2, Fuel Pool Cooling System Startup and Normal Operations, specifies in step 4.1.15.4 that a maximum flowrate of 550 gpm is permitted through each demineralizer. For example, if two demineralizers are in service, the maximum combined flowrate through the demineralizers is 1100 gpm. If SO 19.7.E-2, Aligning Fuel Pool Cooling System to Reactor Well, is performed, the return flow from the Fuel Pool Cooling System is split between the spent fuel pool and the reactor cavity. Since both the spent fuel pool and reactor cavity each have two 6-inch returns, it can be assumed that approximately the same flow returns to each area. This means that with two Fuel Pool Cooling pumps in service, 550 gpm would return to each area. This was the alignment for Fuel Shuffle Part 1 during the last PBAPS refueling outage, as shown in the attached Operations Narrative Logs from PBAPS Refueling Outage 2R15. The entry at 1135 a.m. on 9/16/2004 shows two Fuel Pool Cooling pumps, heat exchangers, and demins are in service, and aligned for return to both the spent fuel pool and reactor cavity per SO 19.7.E-2.

The RHR system alignment used during the entire Shuffle Part 1 for PBAPS 2R15 was Fuel Pool to Reactor Mode per AO 10.4-2, with a flowrate of 5000 gpm. This is a common mode, and depending on work that must be performed, can be the mode used for the majority of the outage. It is used extensively at both LGS and PBAPS. AO 10.4-2 aligns the operating RHR pump suction from the skimmer surge tanks, and discharges to the reactor vessel via the normal shutdown cooling discharge flowpath. The Narrative Logs show the RHR system was placed in this mode at 0401 a.m. on 9/17/2004, approximately one hour before the start of Shuffle Part 1. RHR was maintained in this alignment until long after Shuffle Part 1 was completed.

Since the RHR pump is drawing 5000 gpm from the skimmer surge tanks, and the Fuel Pool Cooling system is drawing another 1100 gpm from the skimmer surge tanks, a total of 6100 gpm flows into and out of the skimmer surge tanks. Since about one-third of this flow (about 2000 gpm) is coming from the spent fuel pool, and only about 550 gpm of flow returning from the Fuel Pool Cooling system is returning to the spent fuel pool, then about 1450 gpm must flow from the reactor cavity to the spent fuel pool. The assumption that at least one-third of the water flowing into the skimmer surge tanks is from the spent fuel pool is a valid assumption, since the surface area of the spent fuel pool is slightly greater than one-third of the total surface area, and the weir plates will be adjusted to be consistent between the pools. According to Bill Bianco, Outage Services Engineer, surface areas of the three pools of water are as follows:

Spent Fuel Pool – 616 sq. ft.
Reactor Cavity – 550 sq. ft.
Equipment Pit – 602 sq. ft.

It is impossible for all 2000 gpm of the flow out of the spent fuel pool to come strictly from spent fuel pool water. Since only about 550 gpm is returning to the spent fuel pool from the Fuel Pool Cooling system, the only place the other 1450 gpm can come from is from the reactor cavity through the transfer canal. It is also not possible for more water to flow from the reactor cavity into the skimmer surge tanks than from the spent fuel pool. Per AO 10.4-2, step 4.1.4, the fuel pool to skimmer surge tank weir gates and reactor cavity to skimmer surge tank weir gates are in their lowest position. This is also required per SO 19.7.E-2. Since the weir plates in the reactor cavity match the level of the skimmer surge tank weirs on the spent fuel pool side, level would have to be higher in the reactor cavity to have higher flow. Since both bodies of water are connected through the transfer canal, it is not possible for them to be at different heights.

The significant amount of water flowing through the transfer canal from the reactor cavity into the spent fuel pool brings the degraded water from the reactor cavity into the spent fuel pool, causing its water to also degrade. Even when Fuel Pool Cooling is in service, the degradation will slowly worsen over time, as only about 370 gpm of the flow from the spent fuel pool (one-third of 1100) is filtered by the Fuel Pool Cooling demineralizers.

The attached Operations Narrative Logs from the most recent PBAPS Refueling outage (2R15) is provided in support of the above statements.

In summary, since the running RWCU pump <u>will</u> trip on high motor winding temperature, and the CRD pump may or may not trip on overcurrent due to high temperature, reactor cavity/spent fuel pool level will either not change (if CRD trips), or will rise very slowly (if CRD does not trip). In either case, this makes answer “d” incorrect. A loss of RWCU during Shuffle Part 1 will cause degradation of reactor cavity water, and with RHR in Fuel Pool to Reactor mode, which is a typical mode during refueling outages, spent fuel pool water visibility would also degrade. This makes answer “c” the correct answer. Answer “c” must be considered a valid answer, since if the exact same situation had actually occurred at any time during Shuffle Part 1 of the last PBAPS refueling outage, reactor cavity <u>and</u> spent fuel pool visibility would have degraded as a result of the RHR alignment being used, regardless of whether Fuel Pool Cooling was in service or not.

Therefore, the Facility Licensee requests the correct answer for Question 37 be changed to “c.”

References Provided:

- Design Basis Document P-S-09, Residual Heat Removal System
- Design Basis Document P-S-33, Reactor Building Closed Cooling Water System
- Design Basis Document P-S-36, Reactor Water Cleanup System
- Design Basis Document P-S-52, Fuel Pool Cooling and Cleanup System
- MCR ARC-211, G-1
- MCR ARC-215, A-2 and B-2
- PLOT-5003A, Control Rod Drive Hydraulic System Lesson Plan (PBAPS)
- PBAPS P&IDs M-361 sheet 1, M363 sheet 1
- AO 10.4-2, Residual Heat Removal System – Fuel Pool to Reactor Mode
- SO 19.7.E-2, Aligning Fuel Pool Cooling System to Reactor Well
- SO 19.1.A-2, Fuel Pool Cooling System Startup and Normal Operations
- Peach Bottom Archival Operations Narrative Logs for period of 9/20/2003 through 9/23/2003
- Peach Bottom Archival Operations Narrative Logs for period of 9/15/2004 through 9/20/2004

NRC Final Resolution:

The NRC agrees with and accepts the licensees' recommendation to change the correct answer from "d" to "c." The NRC conducted detailed reviews of all references provided. As a result of spent fuel bundles movements some corrosion products will detach from the bundles when they are moved through the water and will be deposited in the reactor cavity. Over time a trip of the RWCU pump will result in a degradation of the visibility of the reactor cavity water due to a reduction in filtration of the reactor cavity water. Furthermore, the Fuel Pool Cooling system takes water only from the surface of the reactor cavity, spent fuel pool, and equipment pit. Any corrosion products in the water are required to travel to the surface to be removed. Even with the Fuel Pool Cooling is in service, the degradation will slowly worsen over time. The system line-ups described in the licensee's submittal would cause a significant flow of water through the transfer canal from the reactor cavity into the spent fuel pool. The degraded water from the reactor cavity would enter the spent fuel pool, therefore, over time the water visibility in both the reactor cavity and the spent fuel pool would be expected to degrade. A loss of RWCU during Shuffle Part 1 will cause degradation of reactor cavity water, and with RHR in Fuel Pool to Reactor mode, which is a typical mode during refueling outages, spent fuel pool water visibility would also degrade. This makes answer “c” the correct answer.

Answer “d” cannot be correct since the CRD pump is expected to continue to run after loss of RBCCW and after the (relatively fast) trip of the RWCU pump. The CRD pump is adding 60 GPM to the reactor cavity (and connected fuel pool). The RWCU pump is required to be running to effect a 60 GPM dump flow to the condenser. In this instance (with CRD pump running and RWCU pump tripped), the Reactor Cavity and Fuel Pool levels would actually increase (making “D” wrong from the onset). In addition, in response to the RWCU pump trip annunciator (ARC # 215), the control room personnel will “Shutdown the RWCU system in accordance with SO 12.2.A-2, Reactor Water Cleanup System Shutdown.”

There is no direct trip of the CRD pump from loss of RBCCW. However, loss of RBCCW will cause the CRD pump to overheat. This overheating condition may (in the long term) cause a pump trip. However, in the worst case, after all RWCU and CRD pumps (eventually) trip, the Reactor Cavity and Fuel Pool levels will remain constant. In NO instance would the level “begin to lower.”

Answer “a” cannot be correct since there is no increase in activity and no mechanism to spread the existing activity to other areas of the plant. With the RWCU pump tripped all activity in the RCS will remain in the RCS and there will be NO “higher than normal” plant dose rates.

Answer “b” cannot be correct since, as stated in the original question explanation, PBAPS Refueling Bridge has an independent self-cooled air compressor that has no connection with the station air system(s) and no connection with RBCCW.

Conclusion

The NRC accepts the licensee’s comment to change the correct answer for Question #37 from “D” to “C.”

Original LSRO Question #26:

A nuclear reactor has been shutdown for one week from long-term power operation and shutdown cooling is in service. Upon loss of cooling water to the shutdown cooling heat exchangers, which one of the following coefficients of reactivity will act first to change core reactivity and determine the effect on Shutdown Margin? (Assume continued forced circulation through the core)

	Coefficient to Act First	Effect on Shutdown Margin
C.	Moderator temperature coefficient	Decrease
D.	Fuel temperature coefficient	Increase
E.	Fuel temperature coefficient	Decrease
F.	Moderator Temperature coefficient	Increase

Originally designated correct answer “A.”

Submitted answer explanation:

Note: Question #26 was significantly modified from the original draft question due to a K/A mismatch problem. The replacement question did not provide any documented justification for the correct answer, nor any explanation about why the three distractors were incorrect. The licensee exam developer confirmed he had intended to change the designated correct answer, after revising the question, but did not follow through with this intent.

LICENSEE’S JUSTIFICATION FOR CHANGE:

Revised answer explanation:

Facility Regrade Request:

Accept “d” as the correct answer.

Justification:

This question was modified from NRC Generic Fundamentals Examination Question Bank question B52.

The question provides a reactor “shutdown for one week from long-term power operation and shutdown cooling in service.” It then provides that cooling water is lost to the shutdown cooling heat exchangers. The candidate is then asked to determine which coefficient of reactivity will act first to change core reactivity and what the effect will be on Shutdown Margin.

The given answer on the answer key provides that moderator temperature coefficient will be the first to act. The Facility Licensee agrees that moderator temperature coefficient will act first, since moderator temperature will rise as a direct result of the loss of cooling to the RHR heat exchangers. The candidate must now decide if this will result in a decrease in Shutdown Margin (choice “a”), or an increase in shutdown margin (choice “d”).

For most of the core life, the reactor is considered to have a negative moderator temperature coefficient, where the effect of increasing moderator temperature will be to add negative reactivity to the core. This is due to the moderator density decreasing as a result of the temperature increase, causing neutrons to travel farther before slowing down to thermal energies, and having a higher probability of resonant absorption. Since more neutrons undergo resonant absorption, fewer neutrons are available for thermal fission, and the effect is to add negative reactivity to the core. This addition of negative reactivity moves the reactor farther from criticality, which increases Shutdown Margin. This would make “d ” the correct answer.

If the assumption is made that the core is at the end of life with low moderator temperature, the reactor could have a positive moderator temperature coefficient, which will result in the addition of positive reactivity as moderator temperature is increased. This occurs because as moderator temperature rises, less moderator atoms are present in the core to compete with the fuel for the thermal neutrons. This causes the thermal utilization factor to increase, resulting in more thermal neutrons available to cause fission in the fuel. The addition of positive reactivity moves the reactor closer to criticality, which decreases Shutdown Margin. This would make “a” the correct answer.

Upon further investigation, information was obtained from LaSalle on reactivity effects of moderator temperature at various points in core life. This information is not normally calculated for Limerick or Peach Bottom, but LaSalle is very similar as a C- lattice plant with 764 fuel bundles and 185 control rods. Core response at Limerick and Peach Bottom would therefore also behave in a similar fashion.

As can be seen in the attached spreadsheets for various times in core life, the moderator temperature coefficient can become positive as fuel exposure increases at low moderator temperatures. This is common for BWR plants and can have operational impacts under these special conditions. However, under “all rod in” conditions, such as during an outage, the moderator temperature coefficient is always negative. This can be seen on the attached spreadsheets since the curve for the ARI condition never crosses the 0.000 reactivity point. This is true for all exposure values calculated and for all temperatures. Based upon this data, answer “a” cannot be correct.

Therefore, the Facility Licensee requests “d” be accepted as the correct answer.

References Provided:

- General Physics BWR Generic Fundamentals Reactor Theory Student Text, Chapter 2 (Neutron Life Cycle)

- General Physics BWR Generic Fundamentals Reactor Theory Student Text, Chapter 4 (Reactivity Coefficients)

- NRC Generic Fundamentals Examination Question Bank – BWR, Questions B52, B948, B1248, B1752, B3652.

LaSalle spreadsheets of reactivity variations with moderator temperature at various times in core life (attached).

Licensee's Conclusion: Change the correct answer from **"A"** to **"D."**

NRC RESOLUTION:

The NRC agrees with and accepts the licensee's recommendation.

The actual training provided to the applicants (page 25 of BWR Reactor Theory, Chapter 2) specifies:

The following parameters or design features affect shutdown reactivity conditions:

- Moderator temperature- An increase inserts negative reactivity, increasing the shutdown margin . . . "

This training justifies "D" as a correct answer. This is also technically correct as explained in the licensee's letter.

The context of this statement in Chapter 2 is based on a reactor being undermoderated and having a negative moderator temperature coefficient of reactivity. This is true for over 90% of the core life. For over-moderated reactors having a positive moderator temperature coefficient of reactivity, the opposite effect (decrease in Shutdown Margin) will be manifested by a loss of cooling to the shutdown cooling heat exchanger. As specified on page 5 of BWR Reactor Theory, Chapter 4, "the potential exists for the occurrence of a positive moderator temperature coefficient of reactivity" This is shown to occur late in core life at temperatures between 100 F and 200 F. Since there was no specified core life in the stem of the question, and in the context of a LSRO (performing refueling outages) it would be reasonable for an applicant to conclude end of core life conditions prevailed.

In this instance (with a positive temperature coefficient at EOL), "A" would be technically justified as the correct answer.

However, it has been demonstrated that the Limerick and Peach Bottom cores would NEVER be in an over-moderated condition with all rods in. Since the stem provided that "the reactor has been shutdown for one week" the applicant should conclude all rods are in. Therefore, answer "A" cannot be correct.

There is a direct training basis for selecting "D" as the correct answer. There is a technically accurate core physics basis for selecting "D" as the correct answer. There is NO technically accurate core physics basis for selecting "A" as a correct answer. Therefore, there is a basis for selecting only "D" as a correct answer.

Conclusion:

Accept the licensee's recommendation to change the correct answer from "A" to "D" for LSRO Question #26. The NRC will change the master grading sheet accordingly and regrade all applicants using the revised grading sheet.